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SUBJECT: Forwards revised response to final rule 10CFR50.61 re pressurized thermal shock. Analysis results demonstrate that reactor vessel will not exceed established screening criteria utilizing existing flux reduction program.

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Director of Nuclear Reactor Regulation
Attention: Mr. Lester S. Rubenstein, Director
PWR Project Directorate #2
Division of PWR Licensing - A
United States Nuclear Regulatory Commission
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
PRESSURIZE THERMAL SHOCK
CORRECTION TO RESPONSE TO FINAL RULE 10CFR50.61

Dear Mr. Rubenstein:

On January 22, 1986, Carolina Power & Light Company submitted its response to the final rule 10CFR50.61 concerning Pressurized Thermal Shock values for the H. B. Robinson Steam Electric Plant, Unit No. 2. Certain mathematical symbols were omitted from that original submittal. The copy of the report transmitted herein is identical to that previously submitted except that the notations have been added where appropriate. This revision does not affect any of the conclusions of the originally transmitted version.

Please replace all copies of the originally transmitted report with the version transmitted herein. If you have any questions concerning this subject, please contact Mr. Jan Kozyra at (919) 836-7924.

Yours very truly,

S. R. Zimmerman

S. R. Zimmerman
Manager
Nuclear Licensing Section

MDM/ljs (3346MDM)

Attachment

cc: Dr. J. Nelson Grace (NRC-RII)
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H. B. ROBINSON - UNIT NO. 2

PRESSURIZED THERMAL SHOCK

RESPONSE TO FINAL RULE 10CFR50.61

(0592NEL)

TABLE OF CONTENTS

1.0 INTRODUCTION

Pressurized Thermal Shock (PTS) has been under intensive study by CP&L since it was identified as a potential safety concern. CP&L has already implemented a flux reduction program at HBR2 using Partial Length Shield Assemblies (PLSA) to reduce flux in the most limiting regions of the vessel. Details of this program were previously provided to the NRC via a transmittal from A. B. Cutter to H. R. Denton, dated September 30, 1983 (Reference 1). The analysis presented herein assumes continued implementation of that program.

This report fulfills the requirements of the Final Rule, 10CFR50.61, as published in the Federal Register, Volume 50, No. 141, July 23, 1985. Most of the information provided has previously been reviewed by the NRC in prior correspondence on this subject. This report extracts the pertinent information from those documents and compiles it in a format appropriate for response to the rule. In each case, the source of the previously docketed information is referenced within the report to provide a more detailed discussion of the basis for the data used to perform the analysis. Calculations have been refined, and the current status has been included.

2.0 SUMMARY OF RESULTS

The results of the analysis demonstrate that with the existing flux reduction program, the HBR2 vessel will not exceed the established screening criteria within the period of its license to operate. The most limiting region is the Upper Circumferential Weld which has been calculated to reach a PTS Reference Temperature (RT_{PTS}) of 293.8°F by the end of license (EOL). This is within the 300°F screening criterion established in the final rule.

3.0 CORE REGION MATERIAL CHEMISTRIES

3.1 Plate Materials to Requirements of ASTM A302

- Manufacture and Chemical Analyses by Lukens Steel Company.

Plate	Melt	Nickel %	Copper %	Phosphorus %
W9807-3	B0650-1	.10	.12	.012
-5	A5891-1	.10	.15	.012
-9	P1444-1	.15	.14	.015
W10201-1	A6623-1	.11	.13	.010
-2	A6520-1	.25	.15	.009
-3	B1255-1	.08	.11	.006
-4	A6604-1	.09	.12	.007
-5	B1256-1	.12	.10	.010
-6	B1250-1	.09	.09	.010

3.2 Weld Materials

3.2.1 Longitudinal Welds

- Weld Identification Nos. 1-273A, B, C; 2-273A, B, C; and 2-273A, B, C (see Figure 1 for weld locations).

- Made as part of Malibu reactor (cancelled) in early 1960s when nickel was not added to Combustion Engineering commercial reactor vessel welds.

- No analyses of these welds were made.

- Made with RACO 3 ARCOS B5 Flux Heat 86054B.

- Through its review of the welds similar to HBR2, CP&L was able to determine that both the HBR2 longitudinal welds and the surveillance weld at Connecticut Yankee were both Heat 86054B, RACO 3 welds. Analysis of the Connecticut Yankee surveillance weld is documented in WCAP 10433 which has been previously submitted to the NRC on several dockets. The weld chemistry of the Connecticut Yankee surveillance welds is a copper content of .22% and a nickel content of .054%. See Reference 2.

3.2.2 Lower Circumferential Weld

- Weld Identification No. 11-273 (see Figure 1 for weld location).

- The chemical analysis of this weld has been determined by analyzing a weld in the head of the HBR2 reactor vessel which also was made with RACO 3, Heat 34B009, with 1/16" nickel wire addition and Linde 1092 flux. The weld sample was taken by a procedure agreed to by the NRC staff. The samples were excised by Westinghouse Nuclear Division personnel, and chemical analyses were performed with the best techniques found by Westinghouse Research. See References 2 and 3.

- The copper and nickel content was established at 0.20% and 0.80%, respectively. See References 4 and 5.

3.2.3 Upper Circumferential Weld

- Weld Identification No. 10-273 (see Figure 1 for weld location).

- Made with RACO 3, Heat W5214, and Linde 1092 flux.

- HBR2 vessel surveillance block made from same materials.

- Official chemistry performed on surveillance block by Spectrochemical Laboratories, Inc., Pittsburgh, PA, November 14-23, 1973, for Westinghouse in percentage composition was:

C	Si	Mo	Cu	Ni	Mn	Cr	V	S	P
.16	.34	.46	.32	.66	.98	.024	.001	.014	.021

- A second analysis performed by Spectrochemical Laboratories to verify copper content resulted in a copper value of 0.34. This showed acceptable correlation, and the more conservative 0.34 value was used in the calculation.

Use of Manual Metallic Arc Welding

- These materials were employed to some extent in all of the above welds. Their copper content was very low. Inclusion of their analyses into the weld calculations would produce less conservative results.

4.0 HISTORY OF REACTOR OPERATION IN EFFECTIVE FULL POWER YEARS (EFPY) at 2300 Mw THERMAL TO END OF 1985

Cycles 1 through 8	7.271 EFPY
Cycle 9	.857 EFPY
Cycle 10	.824 EFPY

5.0 INITIAL REFERENCE TEMPERATURES FOR BELTLINE MATERIALS (REFERENCE 6)

Plate No.	NDTT °F	Minimum 50 ft lb/35 mil Temp °F: Parallel to Major Working	Normal to Major Working	RT_{NDT} °F
W10201-1	-30	57	129*	69
-2	-10	46	90*	30
-3	-10	40	96*	36
-4	-30	45	80*	20
-5	-20	55	80*	20
-6	-30	70	105*	45
W9807-3	-20	60	110*	50
-5	-20	48	93*	33
-9	-30	55	69*	9

*Estimated 77 ft/lb temperature from longitudinal data

Weld	°RT_{NDT}
Longitudinal	-56°F **
Lower Circumferential	-56°F **
Upper Circumferential	-56°F **

** In accordance with PTS rule.

6.0 FAST NEUTRON FLUENCE

The fast neutron (> 1 Mev) flux for the HBR2 reactor vessel core region inner vessel surface was calculated by TEC (Reference 1). Their calculations have been checked and found conservative by Brookhaven National Laboratory (Reference 7) and Westinghouse (Reference 8).

After actual copper and nickel contents were established for Weld 11-273 (References 2 and 4), the upper circumferential weld, 10-273, became controlling for PTS. The maximum fast neutron ($>$ 1 Mev) flux for this weld is:

Cycles 1 through 8	3.62×10^{10} n/cm^{2}sec
Cycle 9	1.81×10^{10} n/cm^{2}sec
Cycle 10	1.47×10^{10} n/cm^{2}sec

Cycles (after Cycle 10) plan to employ the same low leakage core loading pattern with the PLSA fuel element configuration; therefore, CP&L projects the same fast flux for those cycles.

The neutron flux is greatest at the center of core flats. The maximum longitudinal weld flux (Weld 2-273A) occurs at an azimuthal reduction of .7777 for Cycles 1-9 and .8185 for Cycle 10 forward.

The fast neutron ($>$ 1 Mev) fluence at the end of 1985 and end of license is as follows for each core region material:

Plate No.	End of 1985	End of License
W10201-1	9.18×10^{18} n/cm^2	1.76×10^{19} n/cm^2
-2	9.18×10^{18}	1.76×10^{19}
-3	9.18×10^{18}	1.76×10^{19}
-4	1.83×10^{19}	4.20×10^{19}
-5	1.83×10^{19}	4.20×10^{19}
-6	1.83×10^{19}	4.20×10^{19}
W9807-3	1.57×10^{19} n/cm^2	1.91×10^{19} n/cm^2
-5	1.57×10^{19}	1.91×10^{19}
-9	1.57×10^{19}	1.91×10^{19}

Weld No.	End of 1985	End of License
1-273A	$< 9.18 \times 10^{18}$ n/cm^2	$< 1.76 \times 10^{19}$ n/cm^2
1-273B	$< 9.18 \times 10^{18}$	$< 1.76 \times 10^{19}$
1-273C	$< 9.18 \times 10^{18}$	$< 1.76 \times 10^{19}$
2-273A	1.36×10^{19}	3.37×10^{19}
2-273B	$< 1.36 \times 10^{19}$	$< 3.37 \times 10^{19}$
2-273C	$< 1.36 \times 10^{19}$	$< 3.37 \times 10^{19}$
3-273A	1.57×10^{19}	1.91×10^{19}
3-273B	$< 1.57 \times 10^{19}$	$< 1.91 \times 10^{19}$
3-273C	$< 1.57 \times 10^{19}$	$< 1.91 \times 10^{19}$
10-273	9.18×10^{18}	1.76×10^{19}
11-273	1.57×10^{19}	1.91×10^{19}

Notes:

1. All longitudinal welds were made from one heat of submerged arc welding wire. Only the highest flux locations were calculated individually. Lower fast neutron flux locations are noted with a less than sign.

2. Figure 1 is provided to display EOL fluences for the various materials and locations.

7.0 RT_{PTS} CALCULATIONS

7.1 Upper Circumferential Weld

$$RT_{PTS} = I + M + 238\ f^{0.194}$$

Where:

$I = -56°F$ for Linde 1092 flux

$M = +34°F$ for Generic "I" values

$f = 1.76$ at upper circumferential weld

Then:

$$RT_{PTS} = (-56°F) + 34°F + 283°F\ (1.76)^{0.194}$$

$$RT_{PTS} = 293.8°F \quad (EOL)$$

7.2 Lower Circumferential Weld   Cu = .20   Ni = .80

$$RT_{PTS} = I + M + [-10 + 470\ Cu + 350\ Cu\ Ni]f^{0.270}$$

Where:

$I = -56°F$

$M = 59°F$

$Cu = 0.20$

$Ni = 0.80$

$f = 1.91$

Then:

$$RT_{PTS} = (-56°F) + 59°F + [-10 + 470\ (0.20) + 350\ (0.20 \times 0.80)\ 1.91^{.270}$$

$$RT_{PTS} = 170°F\ (EOL)$$

7.3 RT_{PTS} Values for Other Plates and Welds

The following values were obtained using fluences reported in the preceding section and the method of paragraph 50.61(b) of the PTS Rule:

Plate No.	RT_{PTS}(°F) End of 1985	RT_{PTS}(°F) End of License
W10201-1	172	182
-2	150	164
-3	128	136
-4	127	141
-5	116	128
-6	134	144
W9807-3	156	159
-5	156	160
-9	128	132

Weld No.	RT_{PTS}(°F) End of 1985	RT_{PTS}(°F) End of License
1-273A	98	117
1-273B	98	117
1-273C	98	117
2-273A	109	139
2-273B	109	139
2-273C	109	139
3-273A	113	120
3-273B	113	120
3-273C	113	120

8.0 FIGURE

IDENTIFICATION, LOCATION AND END OF LICENSE FLUENCE OF BELTLINE REGION MATERIAL FOR THE H. B. ROBINSON UNIT NO. 2 REACTOR VESSEL

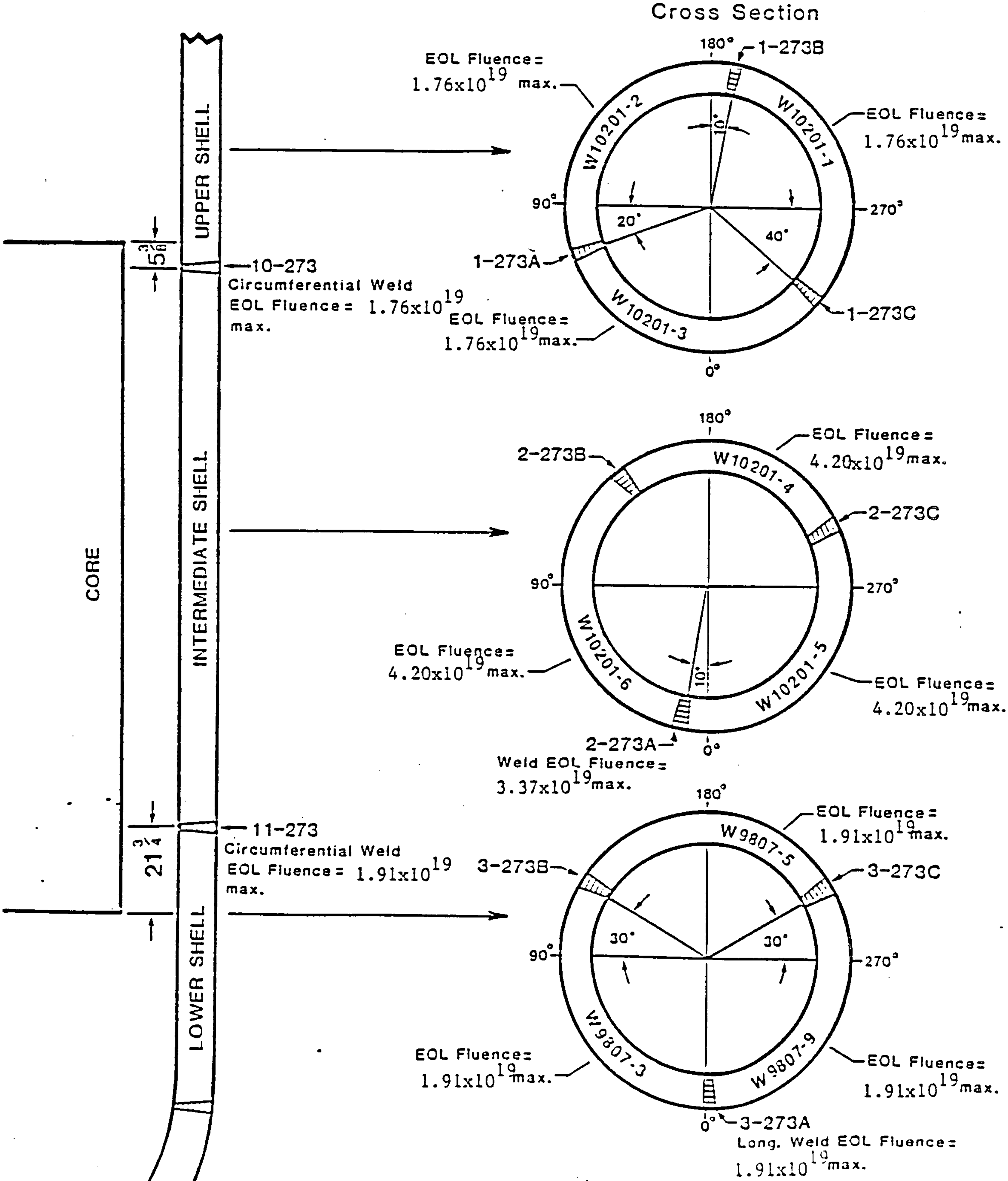

FIGURE 1

9.0 REFERENCES

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2. H. B. Robinson Steam Electric Plant, Unit No. 2, Pressurized Thermal Shock - Material Properties, A. B. Cutter (CP&L) to H. R. Denton (NRC), June 29, 1984.

3. H. B. Robinson Steam Electric Plant, Unit No. 2, Reactor Vessel Materials Properties, S. A. Varga (NRC) to E. E. Utley (CP&L), September 11, 1984.

4. H. B. Robinson Steam Electric Plant, Unit No. 2, Pressurized Thermal Shock - Reactor Vessel Materials, M. A. McDuffie (CP&L) to S. A. Varga (NRC), October 30, 1984.

5. H. B. Robinson Steam Electric Plant, Unit No. 2, Reactor Vessel Material Properties - PTS, S. A. Varga (NRC) to E. E. Utley (CP&L), March 11, 1985.

6. Analysis of Capsule S from the H. B. Robinson, Unit No. 2, Reactor Vessel Surveillance Program, S. E. Yanichko, et. al., December 18, 1973.

7. H. B. Robinson Steam Electric Plant, Unit No. 2, Evaluation of Flux Reduction Achievable by the Use of Partial Length Shield Assemblies, S. A. Varga (NRC) to E. E. Utley (CP&L), November 6, 1984.

8. Westinghouse Evaluation of TEC Report, V. A. Perone, FSD-RSA-84/2134, July 5, 1984.