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MIRAGLIA, F. J. Division of Pressurized Water Reactor Licensing - B (post 8

SUBJECT: Notifies that max RT (pressurized thermal shock) of 230 F will be experienced by lower shell longitudinal seams at end of OL, per 10CFR50.61 (b)(1). Value consistent w/screening criterion for longitudinal welds.

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FLORIDA POWER & LIGHT COMPANY

JAN 2 3 1986

L-86-20

Mr. Frank J. Miraglia, Director
Division of PWR Licensing - B
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Miraglia:

Re: **St. Lucie Unit 1**
Docket No 50-335
<u>10 CFR 50.61 (b)(1) Report</u>

10 CFR 50.61 (b)(1) requires the submittal of projected values of RT (PTS) for the end of the operating license for St. Lucie Unit 1. This requirement arises from the concern over reactor vessel embrittlement and pressurized thermal shock.

As part of the FPL effort associated with the St. Lucie Unit 1 thermal shield removal in 1983, we have previously submitted the majority of the required data. At that time, it was concluded that St. Lucie Unit 1 could safely operate without restriction from PTS considerations.

Since that time we have refined our fluence predictions and analyzed a surveillance capsule. It is our determination therefore, that a maximum RT (PTS) of 230^{o}F will be experienced by the lower shell longitudinal seams at the termination of the plant operating license (July, 2010). This value compares favorably to the screening criterion of 270^{o}F for longitudinal welds.

Should you or your staff have any questions on this information, please contact us.

Very truly yours,

C. O. Woody
Group Vice President
Nuclear Energy Department

SAC/cac

Attachment

cc: Dr. J. Nelson Grace - Region II, USNRC
Harold F. Reis, Esquire
PNS-LI-86-14

Add:

AD - D. CRUTCHFIELD (LTR ONLY)
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ATTACHMENT 1

Reactor Vessel Beltline Materials Description

The St. Lucie Unit 1 reactor pressure vessel was designed and fabricated by Combustion Engineering, Inc. The reactor vessel beltline, as defined by 10 CFR 50, Appendix H, consists of the six plates used to form the lower and intermediate shell courses in the vessel, the included longitudinal seam welds and the lower to intermediate shell girth seam weld. The plates were manufactured from SA533 Grade B Class 1 quenched and tempered plate. The heat treatment consisted of austenization at 1600 ± 50F for four hours, water quenching and tempering at 1225 ± 25F for four hours. The ASME Code qualification test plates were stress relieved at 1150 ± 25F for forty hours, and furnace cooled to 600F. The longitudinal and girth seam welds were fabricated using E8018-C3 manual arc electrodes and Mil B-4 submerged arc weld wire with Linde 124, 0091 and 1092 flux. The post weld heat treatment consisted of a forty hour 1150 ± 25F stress relief heat treatment followed by furnace cooling to 600F. The beltline materials are identified in Tables 1 and 2. Copper, Nickel and initial RT NDT are also listed in these tables.

Calculations

Calculations for present and end of license RT (PTS) are based on equation 1 of 10 CFR 50.61 (b)(2) which gave a lower value than equation 2. All materials data used for these calculations are given in Tables 1 and 2. Source material for these data is supplied in the references.

Generic values for "I" are used for all materials with the associated 59°F "M" value except for a lower shell plate identified as heat number C-5935-3 and the intermediate to lower girth seam where actual values were used with the associated 48°F "M" value

Fluences are projected to end of license based on cycle specific predicted fluence and extrapolated to the end of license based on calculated power distributions. The models used to predict fluence have been reported in reference 4.

TABLE 1

REACTOR VESSEL

BELTLINE PLATES

ST. LUCIE UNIT 1

LOCATION	HEAT NO.	SUPPLIER	% Cu(2)	% Ni(2)	$RT(NDT)_o$(1)
Intermediate Shell	A-4567-1	Lukens	0.11	0.64	0°F
Intermediate Shell	B-9427-1	Lukens	0.11	0.64	0°F
Intermediate Shell	A-4567-2	Lukens	0.11	0.58	10°F
Lower Shell	C-5935-1	Lukens	0.15	0.56	20°F
Lower Shell	C-5935-2	Lukens	0.15	0.57	20°F(3)
Lower Shell	C-5935-3	Lukens	0.12	0.58	0°F

TABLE 2

REACTOR VESSEL BELTLINE WELDS

ST. LUCIE UNIT 1

Location	Weld Seam No.	Wire Heat No.	Flux Type	Flux Batch	% Cu[1]	% Ni[1]	RT_{NDT_o}[1]
Intermediate Shell Longitudinal Seam	2-203A	A-8746/34B009 M/A IAGI*	Linde 124	3878 & 3688	0.12	0.20	-56
Intermediate Shell Longitudinal Seam	2-203B	A-8746/34B009 M/A IAGI* M/A CBB*	Linde 124	3878 & 3688	0.12	0.20	-56
Intermediate Shell Longitudinal Seam	2-203C	A-8746/34B009	Linde 124	3878 & 3668	0.12	0.20	-56
Lower Shell Longitudinal Seam	3-203A	34B009 M/A KBEJ*	Linde 1092	3889	0.30	0.64	-56
Lower Shell Longitudinal Seam	3-203B	34B009 M/A KBEJ*	Linde 1092	3889	0.30	0.64	-56
Lower Shell Longitudinal Seam	3-203C	34B009	Linde 1092	3889	0.30	0.64	-56
Intermediate to Lower Girth Seam	9-203	90136 M/A ABEA* M/A FOAA*	Linde 0091	3999	0.23	0.11	-60[3]

*Manual shielded metal arc electrode (all others automatic submerged arc wire).

TABLE 3

FLUENCE AND RTPTS

1986 and 2010

Location	Fluence n/cm^2 Jan 1986	RT PTS °F JAN 1986	Fluence n/cm^2 JULY 2010	RT PTS °F JULY 2010
Intermediate Shell	3.39×10^{18}	109	1.68×10^{19}	135
Intermediate Shell	3.39×10^{18}	109	1.68×10^{19}	135
Intermediate Shell	3.39×10^{18}	117	1.68×10^{19}	143
Lower Shell	3.39×10^{18}	146	1.68×10^{19}	182
Lower Shell	3.39×10^{18}	136	1.68×10^{19}	172
Lower Shell	3.39×10^{18}	112	1.68×10^{19}	140
Intermediate Shell Longi-tudinal Seam	3.39×10^{18}	44	1.68×10^{19}	66
Intermediate Shell Longi-tudinal Seam	3.39×10^{18}	44	1.68×10^{19}	66
Intermediate Shell Longi-tudinal Seam	3.39×10^{18}	44	1.68×10^{19}	66
Lower shell Longitudinal Seam	3.39×10^{18}	151	1.68×10^{19}	230
Lower Shell Longitudinal Seam	3.39×10^{18}	151	1.68×10^{19}	230
Lower Shell Longitudinal Seam	3.39×10^{18}	151	1.68×10^{19}	230
Intermediate to Lower Girth Seam	3.39×10^{18}	68	1.68×10^{19}	111

References

1. FPL Letter, L-83-263, St. Lucie Unit 1 Reactor Vessel Internals and Thermal Shield, Plant Recovery Program, R. E. Uhrig to R. A. Clark, April 27, 1983

2. Combustion Engineering Report, F-NLM-007 Program For Irradiation Surveillance of Hutchinson Island Plant (PSL-1) Reactor Vessel Materials, September 15, 1970

3. Combustion Engineering Report, TR-F-MCM-005 Evaluation of Baseline Specimens, FPL St. Lucie Unit 1

4. FPL Letter, L-84-29, Reactor Vessel Internals and Thermal Shield Conclusions and Findings, St. Lucie Unit 1, J. W. Williams, Jr. to J. Miller, February 10, 1984