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SUBJECT: Forwards 10CFR50.61 evaluation of PTS of RV beltline matls, completing commitment in 951124 ltr.Proprietary & nonproprietary versions of CEOG rept, "Application of RV Surveillance Data..." also encl.Proprietary rept withheld.

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Florida Power & Light Company, P.O. Box 128, Fort Pierce, FL 34954-0128

FPL

May 14, 1996

L-96-112
10 CFR 50.4
10 CFR 50.61
10 CFR 2.790

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

RE: St. Lucie Units 1 and 2
Docket No. 50-335 and 50-389
10 CFR 50.61 Evaluation of Pressurized Thermal
Shock of Reactor Vessel Beltline Materials

Attachments A and B provide the 10 CFR 50.61(b)(1) pressurized thermal shock submittals for the St. Lucie Units 1 and 2 reactor vessel beltline materials. The evaluations determined the projected reference temperature (RT_{PTS}) at end of license (EOL) for the reactor vessel beltline materials of each reactor vessel. The EOL RT_{PTS} values were compared against the regulatory limit of 270 degrees Fahrenheit (°F) and determined to be acceptably below the limit. This submittal completes the commitment in our letter, L-95-315, dated November 24, 1995.

In addition, one (1) copy each of the proprietary and non-proprietary versions of Combustion Engineering Owners Group report, *Application of Reactor Vessel Surveillance Data for Embrittlement Management*, is enclosed. CEN-405-P Revision 2 is the proprietary version and CEN-405-NP is the non-proprietary version of the report which is referenced in our submittal. CEN-405-P contains proprietary information, the disclosure of which would compromise trade secrets and commercial information considered by Combustion Engineering, Inc. as privileged or confidential. Pursuant to 10 CFR 2.790(a)(4), FPL requests that the enclosed report CEN-405-P be withheld from public disclosure. The affidavit, required by 2.790(b), supporting this request and executed by an authorized representative of Combustion Engineering, Inc., is provided as Attachment C.

NRC approval of this material is requested by April 1, 1997, to support design of the fuel for St. Lucie Unit 1 Cycle 15. Please contact us if there are any questions.

Very truly yours,

W.H. Bohlke
Vice President
St. Lucie Plant

A001 1/1

WHB/GRM

cc: Stewart D. Ebneter, Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant

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ST. LUCIE UNIT 1
REACTOR VESSEL BELTLINE MATERIALS
END OF LICENSE ASSESSMENT OF RT-PTS

I. Discussion

The purpose of this assessment is to show that the St. Lucie Unit 1 projected reference temperature (RT_{PTS}) at end of license (EOL) remains below the PTS screening criteria when using the methodology in the most recent 10 CFR 50.61[1]. The revised rule states that "licensees shall consider plant specific information that could affect the level of embrittlement including data from any related surveillance program results." This submittal provides updated projections of RT_{PTS} for the St. Lucie Unit 1 reactor vessel beltline materials with improved accuracy due to incorporation of available surveillance data from the St. Lucie Unit 1 and Duquesne Light Co. Beaver Valley Unit 1 surveillance programs corresponding to these materials. Using this surveillance data it will be shown that the St. Lucie Unit 1 reactor vessel limiting beltline material, axial welds, 3-203 A, B and C, EOL RT_{PTS} is 213°F which is 57°F below the 270°F screening limit. All other materials are more than 100°F below their respective screening limits.

II. Introduction

The initial RT_{NDT}, chemistry values and end of license RT_{PTS} values have been reported for the St. Lucie Unit 1 reactor vessel beltline materials[2,3]. The NRC has incorporated these initial values into its Reactor Vessel Integrity Database (RVID). The St. Lucie Unit 1 initial RT_{NDT} and chemical composition are summarized in Table 1 for the beltline plate and weld materials. For projections of EOL RT_{PTS} there exist additional information for the St. Lucie Unit 1 reactor beltline controlling weld, girth weld and controlling plate material from reactor vessel material surveillance programs.

St. Lucie Unit 1 Surveillance Program

The St. Lucie Unit 1 (SL-1) reactor vessel and surveillance program materials were fabricated by Combustion Engineering (CE). The surveillance program was designed to meet ASTM E185-73. Two capsules have been removed and tested as part of the original surveillance program[4,5]. The data from these capsules provides information on the embrittlement behavior of the SL-1 reactor vessel lower shell beltline plate (Heat C-5935-2, Code C-8-2) and the intermediate to lower shell girth weld 9-203 (heat 90136). The data indicates that these materials embrittle in a manner that is within $1\sigma_{\Delta}$ of their calculated best fit projections as outlined in

10 CFR 50.61. Therefore, we propose using this surveillance data to determine ΔRT_{NDT} and RT_{PTS} in accordance with 10 CFR 50.61.

Surveillance data for the controlling lower shell axial weld seams 3-203A, B and C (heat 305424) is not available in the SL-1 surveillance program. However, in addition to fabricating vessels for its Nuclear Steam Supply System (NSSS), CE also fabricated vessels for Westinghouse (W) NSSS designed vessels. Therefore, surveillance materials from CE fabricated, W designed NSSS vessels may represent additional data applicable for the CE fabricated St. Lucie Unit 1 vessel. The Duquesne Light Company, Beaver Valley Unit 1 is a W designed NSSS with a CE fabricated reactor vessel. The Beaver Valley Unit 1 (BV-1) and SL-1 vessels were both fabricated during the same period by CE in Chattanooga, Tennessee. The weld selected by W for the BV-1 surveillance program is identical to the SL-1 lower shell axial welds. Both welds were fabricated using the submerged arc weld process with weld wire heat 305424 and Linde 1092 flux lot 3889. Use of the BV-1 surveillance data for the SL-1 welds 3-203A, B and C will permit a more accurate assessment of ΔRT_{NDT} and RT_{PTS}.

Beaver Valley Unit 1 Surveillance Program

The Duquesne Light Co. Beaver Valley Unit 1 reactor vessel and surveillance weld were fabricated by Combustion Engineering (CE). The BV-1 surveillance program was designed to meet ASTM E185-73[6]. Three capsules have been removed and tested as part of the original surveillance program[7,8,9]. Updated fluence values reported with the most recent capsule analysis[9] will be used with the BV-1 capsule data. The data from these capsules provides information on the embrittlement behavior of the SL-1 reactor vessel lower shell axial weld seams 3-203A,B and C (Heat 305424). The data indicates that this material embrittles in a manner that is within $1\sigma_\Delta$ of their calculated best fit projections as outlined in 10 CFR 50.61. Therefore, we proposes to use this surveillance data to determine ΔRT_{NDT} and RT_{PTS} in accordance with 10 CFR 50.61.

III. Method of Calculation of RT_{PTS}

Calculation equations from 10 CFR 50.61 for determining limiting adjusted reference temperatures (RT_{PTS}) are as follows:

Equation 1: Determination of RT_{PTS}

$$RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + Margin$$

$RT_{NDT(U)}$ = Reference temperature (°F) for the unirradiated material or initial RT_{NDT}.

Equation 2: Determination of margin value to apply to RT_{PTS}

$$Margin = 2[(\sigma_U^2 + \sigma_\Delta^2)]^{0.5}$$

Margin = The quantity (°F) to be added to obtain a conservative upper bound value of RT_{PTS}

σ_U = standard deviation for $RT_{NDT(U)}$. °F for measured test values, 17°F for generic data.

σ_Δ = standard deviation for ΔRT_{NDT}. 17°F for plate and 28°F for welds.

Equation 3: Determination of transition temperature shift in RT_{PTS} due to irradiation.

$$\Delta RT_{PTS} = (CF) f^{(0.28 - 0.10 \log f)}$$

CF = Chemistry factor (°F) is a function of Cu and Ni content of the weld or base material and is determined from tables in 10 CFR 50.61. It can also be calculated from actual surveillance data (Equation 4).

$f^{(0.28 - 0.10 \log f)}$ is referred as the fluence factor (ff)

f = Best estimate peak fluence in units of 10E19 n/cm^2(E > 1MeV), at the clad base metal interface on the vessel ID surface of the material being evaluated.

Equation 4: Sum of the squares method for determining CF with surveillance data.

$$CF = \frac{\sum A_i \times ff_i}{\sum ff_i^2}$$

ff_i= Fluence factor - $f^{(0.28 - 0.10 \log f)}$ of the actual surveillance capsule result.

A_i= Measured ΔRT_{NDT} of the actual surveillance capsule data.

Note: Were the Cu and Ni values of the surveillance material differ from the vessel material, A_i must be adjusted by the ratio of the tabulated CF values.

IV. Criteria to Include Surveillance Data into Determination of Calculation of RT_{PTS}

In order to determine if the surveillance material from either of these programs are deemed credible for integrating into RT_{PTS}

estimates, the criteria stated in 10 CFR 50.61 paragraph (c)(2)(i) sections (A) through (E) must be addressed. The five criteria are as follows:

(A) The material in the surveillance capsules must be those which are the controlling materials with regard to radiation embrittlement.
(B) Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions must be small enough to permit the determination of the 30 ft-pound temperature unambiguously.
(C) Where there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values must be less than 28°F for welds and 17°F for base metal. Even if the range in the capsule fluence is large (two or more orders of magnitude), the scatter may not exceed twice those values.
(D) The irradiation temperature of the Charpy specimens in the capsule must equal the vessel wall temperature at the cladding/base metal interface within ± 25°F.
(E) The surveillance data for the correlation monitor material in the capsule, if present, must fall within the scatter band of the data base for the material.

In addition, the CE Owners Group (CEOG) produced a report[10] that provides guidance on the five credibility criteria above and developed additional criteria for using surveillance material from a host reactor of a different design. The report also compares 146 irradiated surveillance data measurement, representing 27 reactor vessels and 52 capsules from CE and Westinghouse NSSS's and concluded that no significant bias exist between CE and Westinghouse surveillance data for CE fabricated vessels. This report was submitted to the NRC and had received partial review[11]. The criteria above with the additional recommendations of the CEOG report[10] will be used to determine the credibility of the BV-1 surveillance weld material for incorporation into the SL-1 surveillance program and determination of RT_{PTS}.

V. Incorporation of All Applicable Surveillance Data to Calculate Chemistry Factor.

The St. Lucie Unit 1 reactor vessel surveillance program contains material samples from the lower shell plate C-8-2, heat C-5935-2 and the intermediate to lower shell girth weld seam, heat 90136. Although these materials are not controlling, incorporation of the surveillance data will improve the accuracy of RT_{PTS} estimates. In addition, the Duquesne Light Co. Beaver Valley Unit 1 surveillance program has been identified as having the same weld heat (305424) and flux type/lot (Linde 1092/3889) as the SL-1 controlling lower shell axial seam welds (3-203 A, B, C).

The determination of the applicability of these surveillance materials and the determination of the chemistry factor for each material is provided below:

A) **Plate C-8-2, heat C-5935-2,** Cu = 0.15%, Ni = 0.57%, tabulated Chemistry Factor (CF) = 108.35°F.

The C-8-2 plate irradiation data is from the St Lucie Unit 1 surveillance program and the results are shown in Table 2. This surveillance material is considered applicable since it is an extension of the vessel shell and was irradiated in the SL-1 reactor vessel under the same conditions (temperature and fluence) as the vessel. The 30 ft-lb temperature was determined unambiguously. The scatter of this ΔRT_{NDT} plate data is within 17°F ($1\sigma_{\Delta}$) of the best fit of the surveillance data and the one correlation monitor plate or standard reference material (SRM) from the SL-1 surveillance program is within the scatter band ($2\sigma_{\Delta}$) for plate material. Therefore this surveillance material meets the credibility requirements stated in 10 CFR 50.61 section (c)(2)(i) in all respects except that this plate material is not the limiting vessel material. **It is proposed that the calculated CF of 79.53°F be used in determinations of RT_{NDT} and RT_{PTS}.**

B) **Plate C-8-1, heat C-5935-1,** Cu = 0.15%, Ni = 0.56%, tabulated Chemistry Factor (CF) = 107.8°F.

This material is the same heat (C-5935) as the SL-1 surveillance plate. The extension (-1) indicates that the heat of plate was sectioned into multiple pieces. Using the ratio procedure to adjust A_1 in Eq. 4 by 107.8°F/108.35°F the calculated value of CF for plate C-8-1, heat C-5935-1 is 79.13°F. **It is proposed that this calculated CF of 79.13°F be used in determinations of RT_{NDT} and RT_{PTS}.**

C) **Plate C-8-3, heat C-5935-3,** Cu = 0.12%, Ni = 0.58%, tabulated Chemistry Factor (CF) = 82.6°F.

This material is the same heat (C-5935) as the SL-1 surveillance plate. The extension (-3) indicates that the heat of plate was sectioned into multiple pieces. Using the ratio procedure to adjust A_1 in Eq. 4 by 82.6°F/108.35°F the calculated value of CF for plate C-8-3, heat C-5935-3 is 60.63°F. **It is proposed that this calculated CF of 60.63°F be used in determinations of RT_{NDT} and RT_{PTS}.**

D) **Weld heat 90136/girth weld 9-203,** Cu = 0.23%, Ni = 0.11%, tabulated Chemistry Factor (CF) = 109.8°F.

The 90136 weld irradiation data is from the St Lucie Unit 1 surveillance program and the results are shown in Table 2.

This surveillance material is considered applicable since it is identical to the vessel girth weld (heat 90136/code 9-203), and was irradiated in the SL-1 reactor vessel under the same conditions (temperature and fluence) as the vessel. The 30 ft-lb temperature was determined unambiguously. The scatter of this ΔRT_{NDT} weld data is within 28°F ($1\sigma_{\Delta}$) of the best fit of the surveillance data and the one correlation monitor plate or standard reference material (SRM) from the SL-1 surveillance program is within the scatter band ($2\sigma_{\Delta}$) for plate material. Therefore, this surveillance material meets the credibility requirements stated in 10 CFR 50.61 section (c)(2)(i) in all respects except that this weld material is not the limiting vessel material. **It is proposed that the calculated CF of 84.35°F be used in determinations of RT_{NDT} and RT_{PTS}.**

E) **Weld heat 305424/welds 3-203A, B & C, flux type/lot Linde 1092/3889 (controlling material for embrittlement),** Cu = 0.28%, Ni = 0.63%, tabulated Chemistry Factor (CF) = 191.65°F.

This weld heat 305424 irradiation data is from the Beaver Valley Unit 1 surveillance program and the results of the 3 capsules are shown in Table 2. This weld material is considered applicable to the SL-1 since it was made by the same submerged arc weld process, by the same fabricator (CE), with the same heat of weld wire and flux lot as the SL-1 controlling lower shell axial welds (3-203 A, B & C). The 30 ft-lb temperature was determined unambiguously. The scatter of this ΔRT_{NDT} weld data is within 28°F ($1\sigma_{\Delta}$) of the best fit of the surveillance data and the calculated CF is 191.33°F.

The BV-1 surveillance program does not have correlation monitor plate, however the CEOG report[10] recommends the host vessel (BV-1) surveillance plate be compared to be within the scatter band ($2\sigma_{\Delta}$) for plate material. The BV-1 surveillance plate material is shown in Table 2 and meets this 34°F ($2\sigma_{\Delta}$) test for credible surveillance program data.

The CEOG report[10] recommends that the fast neutron fluence be compared between the BV-1 surveillance capsules and the SL-1 capsule and vessel. The CEOG report[10] concluded that differences within a factor of 10 will result in comparable irradiation behavior. Table 3 shows the fast neutron fluence of the BV-1 capsules to be within the a factor of 10 of the fluence received by SL-1 capsules and the limiting SL-1 weld that the capsule material represents. The irradiation behavior of the BV-1 surveillance specimen are therefore comparable to the SL-1 limiting weld material.

The CEOG report[10] recommends irradiation temperatures of the surveillance specimens be compared by evaluating the

temperature monitors inside the surveillance capsule and the vessel cold leg temperatures to meet the ± 25°F criteria. The BV-1 capsules are mounted on the thermal shield. The SL-1 capsules are mounted on the vessel wall. Both capsules are subjected to the inlet temperature water during operations. The BV-1 surveillance capsules have thermal monitor melt wires with temperatures of 579°F and 590°F, none of which were reported melted in the 3 surveillance capsule reports[7,8,9]. The SL-1 surveillance capsules also have thermal monitor melt wires, with temperatures of 536°F, 558°F, 580°F and 590°F. The two capsules that have been removed and tested in the SL-1 surveillance program had melted 536°F monitors and deformed 558°F monitors indicating temperatures had exceeded 536°F and were below 558°F.

The BV-1 design nominal inlet temperature was listed in Table 3.1-1 of the Beaver Valley Power Station Unit 1 Updated FSAR as 542.5°F. This nominal temperature has been unchanged since Revision 0 (1/82) through Revision 8 (1/90) which covers the period through the last BV-1 capsule removal date. Review of the inlet temperature data from available BV-1 control room logs[12] from cycles 4-7 indicate that the inlet temperature at ≥ 95% power, ranged from 541.5°F to 547.8°F (544.6°F mathematical average) with the majority of the data clustered around 544°F and 545°F.

By comparison, the St. Lucie Unit 1 design nominal cold leg (inlet) temperature was listed in Table 5.1-1 in the St. Lucie Updated FSAR as 548.4°F from initial issue through the current Amendment 14. However, Technical Specification DNB Margin Limits for cold leg temperatures were set at ≤ 542°F from start up (Amendment 2) through cycle 4 (Amendment 27) which consisted of approximately 32.8 khrs of operation. The St. Lucie Unit 2 Updated FSAR compares several plants design data and lists the SL-1 cycle 1 nominal inlet temperature as 538.9°F. From cycle 5 to present (cycle 13) the SL-1 Technical Specification DNB Margin Limits for cold leg temperatures (Amendment 48 and 130) have been ≤ 549°F. Examination of available control room cold leg temperature logs from cycle 8 to through the current operating cycle 13, indicates the nominal inlet temperature to be 548.8°F with individual channel readings ranging between 547.9°F and 549.5°F. This nominal inlet temperature should also be comparable for cycles 5-7. Assuming a cold leg temperature of 538.9°F for 32.8 khrs (Cycle 1-4) and 548.8°F for the remaining 85.6 khrs (Cycle 5-12) the time weighted average SL-1 nominal cold leg inlet temperature is 546.1°F which is almost identical to the 544.6°F BV-1 cold leg data and should not require any temperature correction for use. The inlet temperature comparison between the BV-1 surveillance capsule specimens and the SL-1 reactor vessel therefore meets the credibility criteria for ± 25°F irradiation temperature requirement stated in 10 CFR 50.61.

The BV-1 surveillance capsule specimen chemistry is a single measurement from the surveillance capsule program[6] of .26% Cu and .62% Ni which correspond to a CF determined from the table of 183.2°F. The average chemistry value for the same weld in the SL-1 and BV-1 vessel is reported as .28% Cu and .63% Ni with a CF of 191.65°F. Using a ratio of CF (191.65°F/183.2°F) and Eq. 4 above the calculated CF applicable to the SL-1 reactor vessel lower shell axial welds (3-203 A, B & C) is 200.15°F.

The BV-1 weld surveillance material meets the credibility requirements stated in 10 CFR 50.61 section (c)(2)(i) in all respects and it is the controlling material for the SL-1 reactor vessel. **It is proposed that the calculated CF of 200.15°F be used in determinations of RT_{NDT} and RT_{PTS}.**

VI. Fluence Projections.

The cumulative vessel maximum fluence at the end of license (EOL) of March 1, 2016, is 3.415 E19 neutrons/cm^2 (E $\geq$1.0 MeV) and is applicable to all the beltline plates and the girth weld (9-203). The azimuthally adjusted maximum EOL fluence for the axial weld is 2.272 E19 neutrons/cm^2 (E $\geq$1.0 MeV). These best estimate fluence projections are based on accumulated fluence to date and projections based on current core loading patterns and a capacity factor of 89.7% from the approved operating schedule. This capacity factor should result in a conservative EOL fluence projection since it is higher than the average capacity factor of the past cycles.

VII. EOL Calculation of RT_{PTS}.

The EOL RT_{PTS} values for the St. Lucie Unit 1 reactor vessel beltline materials are shown in Table 4. The calculation were determined using Eq. 1 through 4 above. Actual surveillance data is used in the determination of chemistry factor and RT_{PTS} for: the limiting axial welds 3-203 A, B and C; the girth weld, 9-203; and the lower shell plates C-8-1, -2 and -3. The margin term for these materials with credible surveillance data is adjusted by a factor of 2 as specified in 10 CFR 50.61.

Table 4 shows that the St. Lucie Unit 1 reactor vessel limiting beltline material, axial welds, 3-203 A, B and C, EOL RT_{PTS} is 213°F which is 57°F below the 270°F screening limit. All other materials are more than 100°F below their respective screening limits.

VIII. REFERENCES

1. Title 10 Code of Federal Regulations, Section 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", December 19, 1995, Federal Register Vol. 60, No. 243.

2. FPL Letter, L-86-20, "St. Lucie Unit 1 Docket No. 50-335, 10 CFR 50.61 (b) (1) Report", C. O. Woody to NRC, January 23, 1986.

3. FPL Letter, L-94-169, "St. Lucie Units 1 and 2, Docket No. 50-335 and 50-389, Generic Letter 92-01 Revision 1 Response to Request for Additional Information", D. A. Sager to NRC, July 1, 1994

4. "Florida Power & Light Co. St. Lucie Unit 1 Post Irradiation Evaluation of Reactor Vessel Surveillance Capsule W-97", Combustion Engineering, Inc., December 1983, TR-F-MCM-004

5. "Analysis of the Capsule at 104° from the FPL St. Lucie Unit 1 Reactor Vessel Radiation Surveillance Program", Westinghouse Electric Corp., November 1990, WCAP-12751.

6. "Duquesne Light Co. Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program", Westinghouse Electric Corp., October 1974, WCAP-8457.

7. "Analysis of Capsule V from Duquesne Light Co. Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program", Westinghouse Electric Corp., January 1981, WCAP-9860.

8. "Analysis of Capsule U from Duquesne Light Co. Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program", Westinghouse Electric Corp., September 1985, WCAP-10867.

9. "Analysis of Capsule W from Duquesne Light Co. Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program", Westinghouse Electric Corp., November 1988, WCAP-12005.

10. "Application of Reactor Vessel Surveillance Data for Embrittlement Management", Combustion Engineering Owners Group, July 1993, CEN-405-P Revision 2.

11. CEOG Letter, CEOG-93-252, "Submittal of CEN-405-P, Revision 2, Application of Reactor Vessel Surveillance Data for Embrittlement Management", R. F. Burski to NRC, August 6, 1993.

12. Duquesne Light Letter, ND1DMS:0370, "Operating Temperature Data for Beaver Valley Unit 1", R. A. Hruby Jr. to R. S. Boggs, FPL, April 4, 1996.

TABLE 1: St. Lucie Unit 1 Reactor Vessel Beltline Material Initial Properties

MATERIAL LOCATION & (CODE NO.)	HEAT NO	FLUX TYPE/LOT	% Cu	% Ni	INITIAL RT_{NDT} (°F)
Intermediate Shell Plate (C-7-1)	A4567-1	NA	0.11	0.64	0
Intermediate Shell Plate (C-7-2)	B9427-1	NA	0.11	0.64	-10
Intermediate Shell Plate (C-7-3)	A4567-2	NA	0.11	0.58	+10
Lower Shell Plate (C-8-1)	C5935-1	NA	0.15	0.56	+20
Lower Shell Plate (C-8-2) *	C5935-2	NA	0.15	0.57	+20
Lower Shell Plate (C-8-3)	C5935-3	NA	0.12	0.58	0
Intermediate Shell Axial Seam Welds(2-203 A, B, C)	A8746/ 34B009	Linde 124/ 3878&3688	0.19	0.10	-56 (Generic)
Lower Shell Axial Seam (3-203 A,B,C)**	305424	Linde 1092/3889	0.28	0.63	-60
Intermediate to Lower Shell Girth Seam (9-203)*	90136	Linde 0091/3999	0.23	0.11	-60

* Monitored in the St. Lucie Unit 1 reactor vessel surveillance program.

** Monitored in the Duquesne Light Company Beaver Valley Unit 1 reactor vessel surveillance program.

Table 2: Surveillance Data for St.Lucie Unit 1 Beltline Material

Plant/ Capsule	Matl /Code	Measured ΔRT_{NDT}	Capsule Fluence (n/cm^2)	Calculated Best Fit CF (Eq. 4)	Best fit CF x ff (Eq 3)	Measured ΔRT_{NDT} minus Best fit
SL-1 /97°	Weld/ 90136	74°F	5.50 E18	84.35°F	58.5°F	15.5°F
SL-1 /104°	Weld/ 90136	73°F	7.16 E18		69.3°F	3.7°F
SL-1 /97°	C-8-2 Trans.	70°F	5.50 E18	79.53°F	55.2°F	14.8°F
SL-1 /97°	C-8-2 Long.	68°F	5.50 E18		55.2°F	12.8°F
SL-1 /104°	C-8-2 Long.	67°F	7.16 E18		65.3°F	1.7°F
SL-1 /104°	SRM-HSST-01MY	110°F	7.16 E18	136.1°F (Table .18 Cu,.66 Ni)	123.3°F	-13.3°F
BV-1 /V	Weld/ 305424	150°F	2.91 E18	191.33°F	126.8°F	23.2°F
BV-1 /U	Weld/ 305424	155°F	6.54 E18		168.6°F	-13.6°F
BV-1 /W	Weld/ 305424	185°F	9.49 E18		188.5°F	- 3.5°F
BV-1 /V	plate /long	130°F	2.91 E18	167.8°F	111.2°F	18.8°F
BV-1 /V	plate /trans	140°F	2.91 E18		111.2°F	28.8°F
BV-1 /U	plate /long	120°F	6.54 E18		147.8°F	-27.8°F
BV-1 /U	plate /trans	135°F	6.54 E18		147.8°F	-12.8°F
BV-1 /W	plate /long	150°F	9.49 E18		165.4°F	-15.4°F
BV-1 /W	plate /trans	185°F	9.49 E18		165.4°F	19.6°F

SL = St. Lucie
BV = Beaver Valley
Trans. = transverse orientation
Long. = longitudinal orientation
ff= fluence factor from Eq. 3

Table 3: Beaver Valley Unit 1 Surveillance Capsule and St. Lucie Unit 1 Vessel Fast Neutron Fluence (E> 1 MeV) Comparison

Plant	Capsule	Fluence (f), n/cm^2	EFPY	EFPS	Fluence, f/EFPY	Fluence, f/EFPS
BV-1	V	2.91 E18	1.16	3.66 E07	2.51 E18	7.95 E10
BV-1	U	6.54 E18	3.59	1.13 E08	1.82 E18	5.78 E10
BV-1	W	9.49 E18	5.89	1.86 E08	1.61 E18	5.11 E10
SL-1	97°	5.50 E18	4.67	1.47 E08	1.18 E18	3.73 E10
SL-1	104°	7.16 E18	9.515	3.00 E08	7.52 E17	2.39 E10
SL-1	Vessel limiting welds	1.20 E19	11.27	3.55 E08	1.07 E18	3.39 E10

EFPY= Effective full power years
EFPS= Effective full power seconds

Table 4: St. Lucie Unit 1 Reactor Vessel Beltline Material EOL RT_{PTS} Values.

LOCATION	Heat ID #	Cu%	Ni%	CALCULATED. CF	TABLE CF	INITIAL RTndt	MARGIN	EOL PEAK FLEUNCE, E19 n/cm^2	FLUENCE FACTOR (ff)	Delta RTpts	EOL RTpts	PTS LIMIT
Lower shell plate (C-8-1)	C-5935-1	0.15	0.56	79.13		20 °F	17 °F	3.42E+19	1.32	105 °F	142 °F	270 °F
Lower shell plate (C-8-2)	C-5935-2	0.15	0.57	79.53		20 °F	17 °F	3.42E+19	1.32	105 °F	142 °F	270 °F
Lower shell plate (C-8-3)	C-5935-3	0.12	0.58	60.63		0 °F	17 °F	3.42E+19	1.32	80 °F	97 °F	270 °F
Int. shell plate (C-7-1)	A-4567-1	0.11	0.64		74.6	0 °F	34 °F	3.42E+19	1.32	99 °F	133 °F	270 °F
Int. shell plate (C-7-2)	B-9427-1	0.11	0.64		74.6	-10 °F	34 °F	3.42E+19	1.32	99 °F	123 °F	270 °F
Int. shell plate (C-7-3)	A-4567-2	0.11	0.58		73.8	10 °F	34 °F	3.42E+19	1.32	97 °F	141 °F	270 °F
Lower shell axial welds (3-203A,B,C)	305424	0.28	0.63	200.15		-60 °F	28.0 °F	2.27E+19	1.22	245 °F	213 °F	270 °F
Int. shell axial welds (2-203A,B,C)	A-8746 & 34B009	0.19	0.10		91.5	-56 °F	65.5 °F	2.27E+19	1.22	112 °F	121 °F	270 °F
Int to Lower girth welds (9-203)	90136	0.23	0.11	84.35		-60 °F	28 °F	3.42E+19	1.32	111 °F	79 °F	300 °F

ST. LUCIE UNIT 2
REACTOR VESSEL BELTLINE MATERIALS
END OF LICENSE ASSESSMENT OF RT-PTS

I. Discussion

The purpose of this assessment is to show that the St. Lucie Unit 2 projected reference temperature (RT_{PTS}) at end of license (EOL) remains below the PTS screening criteria when using the methodology in the most recent 10 CFR 50.61[1]. This submittal provides updated projections of RT_{PTS} for the St. Lucie Unit 2 reactor vessel beltline materials. The St. Lucie reactor vessel beltline is plate limited. The beltline materials are low in copper and nickel, and therefore, are relatively more resistant to embrittlement. Sufficient surveillance capsule data is not yet available, therefore projections of EOL RT_{PTS} will be made based on initial chemistry values and the projection methodology of 10 CFR 50.61. The St. Lucie Unit 2 reactor vessel limiting beltline material, plate, M-605-2 has an EOL RT_{PTS} of 160°F which is 110°F below the 270°F screening limit. All other beltline materials are more than 110°F below their respective screening limits.

II. Introduction

The initial RT_{NDT}, chemistry values and previous end of license RT_{PTS} values have been reported for the St. Lucie Unit 2 reactor vessel beltline materials[2,3]. The NRC has incorporated these initial values into its Reactor Vessel Integrity Database (RVID). The St. Lucie Unit 2 initial RT_{NDT} and chemical composition are summarized in Table 1 for the beltline plate and weld materials.

III. Method of Calculation of RT_{PTS}

Calculation equations from 10 CFR 50.61 for determining limiting adjusted reference temperatures (RT_{PTS}) are as follows:

Equation 1: Determination of RT_{PTS}

$$RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$$

$RT_{NDT(U)}$ = Reference temperature (°F) for the unirradiated material or initial RT_{NDT}.

Equation 2: Determination of margin value to apply to RT_{PTS}

Margin = $2[(\sigma_U^2 + \sigma_\Delta^2)]^{0.5}$

Margin = The quantity (°F) to be added to obtain a conservative upper bound value of RT_{PTS}

σ_U = standard deviation for $RT_{NDT(U)}$. 0°F for measured test values, 17°F for generic data.

σ_Δ = standard deviation for ΔRT_{NDT}. 17°F for plate and 28°F for welds

Equation 3: Determination of transition temperature shift in RT_{PTS} due to irradiation.

$$\Delta RT_{PTS} = (CF)\, f^{(0.28 - 0.10 \log f)}$$

CF = Chemistry factor (°F) is a function of Cu and Ni content of the weld or base material and is determined from tables in 10 CFR 50.61.

$f^{(0.28 - 0.10 \log f)}$ is referred as the fluence factor (ff)

f = Best estimate peak fluence in units of 10E19 n/cm^2(E > 1MeV), at the clad base metal interface on the vessel ID surface of the material being evaluated.

IV. Fluence Projections.

The cumulative vessel maximum fluence at the end of license (EOL) of April 6, 2023 is 2.76 E19 neutrons/cm^2 (E $\geq$1.0 MeV) and is conservatively applied to all the beltline plates and welds. The fluence projections are based on current fluence to date and projections based on current core loading patterns. These best estimate fluence projections are based on accumulated fluence to date and projections based on current core loading patterns and a capacity factor of 96.0% from the approved operating schedule. This capacity factor should result in a conservative EOL fluence projection since it is higher than the average capacity factor of the past cycles.

V. EOL Calculation of RT_{PTS}.

The EOL RT_{PTS} values for the St. Lucie Unit 2 reactor vessel beltline materials are shown in Table 2. The calculation were determined using Eq. 1 through 3 above.

Table 2 shows that the St. Lucie Unit 2 reactor vessel limiting beltline material, lower shell plate M-605-2, EOL RT_{PTS} is 160°F which is 110°F below the 270°F screening limit. All other materials are in excess of 110°F below their respective screening limits.

VI. REFERENCES

1. Title 10 Code of Federal Regulations, Section 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", December 19, 1995, Federal Register Vol. 60, No. 243.

2. FPL Letter, L-86-25, "St. Lucie Unit 2 Docket No. 50-389, 10 CFR 50.61 (b) (1) Report", C. O. Woody to NRC, January 23, 1986.

3. FPL Letter, L-94-169, "St. Lucie Units 1 and 2, Docket No. 50-335 and 50-389, Generic Letter 92-01 Revision 1 Response to Request for Additional Information", D. A. Sager to NRC, July 1, 1994

TABLE 1: St. Lucie Unit 2 Reactor Vessel Beltline Material Initial Properties

MATERIAL LOCATION & (CODE NO.)	HEAT NO	FLUX TYPE/LOT	% Cu	% Ni	INITIAL RT_{NDT} (°F)
Lower Shell Plate(M-4116-1)	B-8307-2	NA	0.06	0.57	+20
Lower Shell Plate(M-4116-2)	A-3131-1	NA	0.07	0.60	+20
Lower Shell Plate(M-4116-3)	A-3131-2	NA	0.07	0.60	+20
Intermediate Shell Plate (M-605-1)	A-8490-2	NA	0.11	0.61	+30
Intermediate Shell Plate (M-605-2)	B-3416-2	NA	0.13	0.62	+10
Intermediate Shell Plate (M-605-3)	A-8490-1	NA	0.11	0.61	0
Intermediate Shell Axial Seam Welds (101-124 A,B)	83642	Linde 0091 /3536	0.04	0.07	-80
Intermediate Shell Axial Seam Welds (101-124C)	83642 /83637	Linde 0091/ 3536&1122	0.04	0.07	-50
Lower Shell Axial Seam (101-142 A,B,C)	83637	Linde 0091/1122	0.05	0.10	-50
Intermediate to Lower Shell Girth Seam (101-171)	83637 /3P7317	Linde 124/0951	0.07	0.08	-70

Table 2: St. Lucie Unit 2 Reactor Vessel Beltline Material EOL RT_{PTS} Values.

MATERIAL LOCATION & (CODE NO.)	HEAT NO	% Cu	% Ni	Chemistry Factor (CF)	Initial RT_{NDT}	Margin	EOL Fluence n/cm^2	ΔRT_{PTS}	EOL RT_{PTS}	PTS Limit
Lower Shell Plate(M-4116-1)	B-8307-2	0.06	0.57	37.0	+20°F	34°F	2.76E19	47°F	101°F	270°F
Lower Shell Plate(M-4116-2)	A-3131-1	0.07	0.60	44.0	+20°F	34°F	2.76E19	56°F	110°F	270°F
Lower Shell Plate(M-4116-3)	A-3131-2	0.07	0.60	44.0	+20°F	34°F	2.76E19	56°F	110°F	270°F
Intermediate Shell Plate (M-605-1)	A-8490-2	0.11	0.61	74.15	+30°F	34°F	2.76E19	94°F	158°F	270°F
Intermediate Shell Plate (M-605-2)	B-3416-2	0.13	0.62	91.5	+10°F	34°F	2.76E19	116°F	160°F	270°F
Intermediate Shell Plate (M-605-3)	A-8490-1	0.11	0.61	74.15	0°F	34°F	2.76E19	94°F	128°F	270°F
Interm. Shell Axial Welds (101-124 A,B)	83642	0.04	0.07	30.7	-80°F	56°F	2.76E19	39°F	15°F	270°F
Interm. Shell Axial Welds (101-124C)	83642 /83637	0.04	0.07	30.7	-50°F	56°F	2.76E19	39°F	45°F	270°F
Lower Shell Axial Welds (101-142 A,B,C)	83637	0.05	0.10	37.5	-50°F	56°F	2.76E19	48°F	54°F	270°F
Inter. to Lower Shell Girth Weld(101-171)	83637 /3P7317	0.07	0.08	41.2	-70°F	56°F	2.76E19	52°F	38°F	300°F

ORIGINAL

L-96-112
Attachment C

AFFIDAVIT PURSUANT

TO 10 CFR 2.790

I, I.C. Rickard, depose and say that I am the Director, Operations Licensing of Combustion Engineering, Inc., duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is contained in the following documents:

1) CEN-405-P Revision 2, "Application of Reactor Vessel Surveillance Data for Embrittlement Management," July 1993.
2) Enclosure II to CEOG-93-252, Formal Response to NRC Request for Additional Information on CEN-405-P, "Application of Reactor Vessel Surveillance Data for Embrittlement Management," August 1993.

These documents have been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Combustion Engineering in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1. The information sought to be withheld from public disclosure, is owned and has been held in confidence by Combustion Engineering. It consists of the details concerning the fabrication process, material properties, and surveillance data used to develop an approach to ascertain the embrittlement of reactor vessels.
2. The information consists of test data or other similar data concerning a process, method or component, the application of which results in substantial competitive advantage to Combustion Engineering.
3. The information is of a type customarily held in confidence by Combustion Engineering and not customarily disclosed to the public. Combustion Engineering has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The details of the aforementioned system were provided to the Nuclear Regulatory Commission via letter DP-537 from F. M. Stern to Frank Schroeder dated December 2, 1974. This system was applied in determining that the subject document herein is proprietary.
4. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.
5. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.
6. Public disclosure of the information is likely to cause substantial harm to the competitive position of Combustion Engineering because:

a. A similar product is manufactured and sold by major pressurized water reactor competitors of Combustion Engineering.

b. Development of this information by Combustion Engineering required hundreds of thousand of dollars and thousands of manhours of effort. A competitor would have to undergo similar expense in generating equivalent information.

c. In order to acquire such information, a competitor would also require considerable time and inconvenience to develop a similar approach to ascertain the embrittlement of reactor vessels given detailed knowledge of the fabrication process, material properties, and surveillance data.

d. The information consists of the details concerning the fabrication process, material properties, and surveillance data used to develop an approach to ascertain the embrittlement of reactor vessels, the application of which provides a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Combustion Engineering, take marketing or other actions to improve their product's position or impair the position of Combustion Engineering's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

e. In pricing Combustion Engineering's products and services, significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included. The ability of Combustion Engineering's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

f. Use of the information by competitors in the international marketplace would increase their ability to market nuclear steam supply systems by reducing the costs associated with their technology development. In addition, disclosure would have an adverse economic impact on Combustion Engineering's potential for obtaining or maintaining foreign licensees.

Further the deponent sayeth not.

I. C. Rickard

~~I. C. Rickard~~

Director, Operations Licensing

Sworn to before me

this 22nd day of April, 1996

Laurie J. White

Notary Public

My commission expires: 8/31/99