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SUBJECT: Forwards response to RAI re pressurized thermal shock evaluations.

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

FPL

January 14, 1997

L-97-10
10 CFR 50.4
10 CFR 50.61

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

RE: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
NRC TAC Nos. M95484 and M95485
Request for Additional Information (RAI) - Response
10 CFR 50.61 - Pressurized Thermal Shock Evaluation

By letter dated October 15, 1996, the NRC staff requested additional information needed to complete their assessment of the pressurized thermal shock (PTS) evaluations for St. Lucie Units 1 and 2. The response to your RAI is attached.

The 10 CFR 50.61(b)(1) PTS evaluations were submitted by Florida Power and Light Company (FPL) letter, L-96-112, on May 14, 1996, and supplemented by FPL letter, L-96-233, on September 23, 1996. The evaluations determined that the projected reference temperature (RT_{PTS}) at the end of license (EOL) for the reactor vessel beltline materials of each reactor vessel. The EOL RT_{PTS} values were compared against the regulatory limits and determined to be acceptably below those limits. FPL originally requested NRC approval of the PTS evaluations by April 1, 1997, to support design of the fuel for St. Lucie Unit 1 Cycle 15. Due to outage schedule changes for Cycle 15, FPL now requests approval by April 1, 1998, to support fuel design for St. Lucie Unit 1 Cycle 16.

This letter does not contain any new regulatory commitments. Please contact us should you need any additional information to support your assessment.

Very truly yours,

J. A. Stall
Vice President
St. Lucie Plant

JAS/GRM 210049

A049

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

Background

The NRC request for additional information generated from the Florida Power and Light Company pressurized thermal shock evaluations[1] is restated below:

The staff requires additional information in order to complete its review of the Florida Power and Light Company pressurized thermal shock (PTS) evaluations. The submittal consisted of three parts:

- "St. Lucie Unit 1 Reactor Vessel Beltline Materials End of License Assessment of RT_{PTS}"
- "St. Lucie Unit 2 Reactor Vessel Beltline Materials End of License Assessment of RT_{PTS}"
- Proprietary and non-proprietary copies of the Combustion Engineering Owner's Group (CEOG) report CEN-405-P, Revision 3, "Application of Reactor Vessel Surveillance Data for Embrittlement Management"

Below are the four NRC questions with the FPL response provided after each question:

NRC Request 1:

Provide the actual cold leg temperatures for each cycle for both St. Lucie Unit 1 (SL-1) and Beaver Valley Unit 1 (BV-1) in support of the integrated surveillance approach. Page A-7 of the submittal discusses "nominal inlet temperatures" as listed in the updated final safety analysis report (UFSAR) and "assumed cold leg temperatures." The actual cold leg temperatures for each cycle are needed for both units in order for the staff to determine the applicability of the BV-1 surveillance data to the SL-1 reactor vessel.

FPL Response 1:

The Beaver Valley Unit 1 (BV-1) surveillance specimens from capsules V, U and W were removed from the BV-1 reactor vessel at the end of cycle 1, cycle 4, and cycle 6 respectively. In order to compare the surveillance specimen temperature environment, the actual cold leg temperature data from the BV-1 control room logs were reviewed. The logs from cycle 4 through cycle 7 were made available by Duquesne Light Co. to Florida Power and Light. The logs were reviewed with minimum, maximum and bulk (majority of the data) temperature entries with greater than 95% power recorded for each cycle. The minimum and maximum temperatures for each cycle were averaged to further reduce the data for comparison. Although cycle 7 was after the BV-1 surveillance capsules were removed, it was evaluated to show uniformity with the T-cold data. The results are provided in the following table.

Beaver Valley Unit 1 Cold Leg (Inlet) Temperature Data				
Cycle	Bulk of data centered around	Minimum Cold Leg Temperature	Maximum Cold Leg Temperature	Average of Min.& Max. Temperature
4	544°F	541.5°F	546.0°F	543.8°F
5	544°F	541.8°F	546.2°F	544.0°F
6	545°F	543.5°F	547.8°F	545.7°F
7	545°F	542.8°F	547.4°F	545.1°F
Average 4-7				544.7°F

The Beaver Valley cycle 1 through cycle 3 data were not available. However, Beaver Valley plant personnel indicated that the BV-1 cold leg temperature had remained essentially unchanged from startup through the cycle 7, and that the data from cycle 4 through cycle 7 above should be representative for cycle 1 through cycle 3[2]. Therefore, based on the actual temperature data from cycle 4 through cycle 7, the BV-1 cold leg inlet temperatures representative of cycle 1 through cycle 7 were assumed to be 544.7°F.

The St. Lucie Unit 1 actual cold leg temperature data from the control room logs were reviewed from the end of cycle 4 through cycle 13. Since the temperature data were uniform and there was a large amount of data, full power steady state temperatures were representative of the beginning and end of each cycle. The actual cycle specific data are presented in the following table.

St. Lucie Unit 1 Cold Leg (Inlet) Temperature Data				
Date	Cycle	A Cold Leg Temp (°F)	B Cold Leg Temp (°F)	Average Cold Leg Temp. (°F)
*7/30/81 (EOC)	4	541.4	541.7	541.6
*4/1/82 (BOC)	5	541.4	541.3	541.4
9/12/82 (BOC)	5	548.3	547.4	547.9
2/10/83 (EOC)	5	548.8	548.3	548.6
6/14/84 (BOC)	6	548.5	548.4	548.5
9/14/85 (EOC)	6	548.8	548.6	548.7
1/10/86 (BOC)	7	549.4	548.3	548.9
1/20/87 (EOC)	7	548.9	548.7	548.8
5/14/87 (BOC)	8	549.3	548.2	548.8
7/8/88 (EOC)	8	549.0	548.4	548.7
9/12/88 (BOC)	9	549.0	548.5	548.8
1/9/90 (EOC)	9	549.5	547.9	548.7
5/17/90 (BOC)	10	549.2	548.5	548.8
10/17/91 (EOC)	10	549.2	548.4	548.8
1/7/92 (BOC)	11	548.9	548.7	548.8
3/27/93 (EOC)	11	549.1	548.3	548.7
7/4/93 (BOC)	12	549.0	548.7	548.8
10/24/94 (EOC)	12	549.2	548.8	549.0
12/20/94 (BOC)	13	549.2	548.8	549.0
4/10/96 (EOC)	13	548.3	548.1	548.2
Average - 2700 MWt thermal	5-13			548.7
* Average - 2560 MWt thermal	4, start of 5			541.5

* Prior to up rating to 2700 MWt on 4-10-82
BOC- Near the Beginning of the cycle EOC- Near the End of the cycle

The temperature data for St. Lucie Unit 1 cycle 1 through cycle 3 and the early part of cycle 4 were not readily available. From cycle 1 through cycle 4, St. Lucie Unit 1 operated with a 2560 MWt thermal core. Plant personnel indicated that vessel inlet temperature for the end of cycle 4 and the beginning of cycle 5 (prior to the power up rate on April 10, 1982) should be representative of cycle 1 through cycle 4 vessel inlet temperature. Therefore, the average reactor vessel inlet temperature for St. Lucie Unit 1 for cycle 1 through cycle 4 was 541.5°F and for cycle 5 through cycle 13 was 548.7°F.

The time weighted average inlet temperature is determined as follows. Using a cold leg inlet temperature of 541.5°F for 32,800 hours (Cycle 1-4) and 548.7°F for the remaining 85,600 hours (Cycle 5-13) the time weighted average SL-1 nominal cold leg inlet temperature is 546.7°F which is almost identical to the 544.7°F BV-1 cold leg data.

Since the BV-1 surveillance capsules were irradiated at a similar, but slightly lower cold leg temperature compared to the St. Lucie Unit 1 cold leg temperature, the BV-1 surveillance data does not require any temperature correction for use. The inlet temperature comparison between the BV-1 surveillance capsule specimens and the SL-1 reactor vessel meets the credibility criteria of ± 25°F irradiation temperature requirement stated in 10 CFR 50.61.

NRC Request 2:

Determine if there are other initial test data (Charpy, Drop Weight and $RT_{NDT(U)}$ values) for heats 305424, 90136, 83642, 83637 and 3P7317. Provide a listing of each initial reference temperature value, including the source of the data. If other data are available, justify the initial reference temperature values that appear in Table 4 (page A-14) and Table 2 (page B-5) of the submittal for the heats listed above.

FPL Response 2:

To determine if there are other initial test data for the above listed weld wire heats applicable to the St. Lucie Units 1 and 2, the Electric Power Research Institute (EPRI) RPVDATA (Version 1.3) database was reviewed. The EPRI RPVDATA database contains an NRC Best Estimate Data screen of the results of the NRC RVID database from the Generic Letter 92-01 responses.

St. Lucie Unit 1 was designed and built to ASME Section III 1965 Edition, Winter of 1967 Addenda when drop weight testing of weld metal was not required. At that time it was the practice for Combustion Engineering (CE), the nuclear steam supply system vendor, to perform weld material drop weight testing on the weld material in the reactor vessel surveillance program. As a result, there is usually a generic value or only one initial RT_{NDT} test value for each weld wire heat. This vessel was built at a time when the weld wires were copper coated, and nickel was alloyed with some of the wires. Therefore, the welds are the limiting material and the initial RT_{NDT} values are critical in the determination of embrittlement predictions such as PTS.

St. Lucie Unit 2 was designed and built to ASME Section III 1971 Edition, Summer of 1972 Addenda, when both drop weight and Charpy testing was required for all pressure boundary materials. Therefore, there is measured initial RT_{NDT} data for every heat of weld material and in some cases, multiple data exist. The St. Lucie Unit 2 weld heat initial properties are not critical

for embrittlement predictions since these low copper, low nickel welds are not the limiting vessel material and their end of license shift in RT_{NDT} is less than 60°F.

The St. Lucie Unit 2 vessel also has some welds that used multiple wire heats, in one case as a result of a repair. In cases with multiple heats of wire, FPL conservatively chose the higher of the initial RT_{NDT} values and assigned it to the weld seam. When this weld seam was reported in a database, the higher "conservative" value was sometimes assigned to both heats separately. This can add to the variability of the data outside of the normal variability of the measured test data. Table 1 and Table 2 now list each heat separately with the specific initial reference test value reported.

The results for each heat are discussed below.

St. Lucie Unit 1 Weld Wire Heats:

Heat 305424/Linde 1092/Lot 3889: This limiting weld heat was used in the fabrication of the St. Lucie Unit 1 vessel lower shell axial welds, 3-203A, B & C. The database search indicated there are three plants that have this wire heat in combination with Linde 1092 flux, all with the same flux lot. The initial RT_{NDT} of -60°F is based on actual drop weight and Charpy results from the Beaver Valley Unit 1 surveillance program[3]. The only other initial RT_{NDT} values reported are a generic value of -56°F and a plant specific -50°F from LaSalle Unit 1. The -50°F LaSalle value is not a measured value based on drop weight testing. It is a calculated result based on +10°F Charpy data as identified in the LaSalle Unit 1 response to Generic Letter 92-01[4] and was submitted as a licensed value specific for LaSalle. Therefore, the initial RT_{NDT} value of -60°F is justified as the only measured value that meets the requirements of ASME Section III NB 2331.

Heat 90136/Linde 0091/Lot 3999: This non limiting weld heat was used in the fabrication of the St. Lucie Unit 1 vessel intermediate to lower shell weld, 9-203, and the surveillance weld. The database search indicated there are 4 plants that have this wire heat in combination with different lots of Linde 0091 flux. These plants have either reported plant specific values of -60°F or the generic value if they have no data. The St. Lucie Unit 1 value of -60°F is justified as a measured value that meets the requirements of ASME Section III NB 2331 with all other measured value in agreement.

St. Lucie Unit 2 Weld Wire Heats:

Heat 83642/Linde 0091/Lot 3536: This non limiting weld heat was used in the fabrication of the St. Lucie Unit 2 vessel intermediate shell axial welds, 101-124 A, B, & C. All three seams used this wire/flux/lot combination, however, the C seam was repaired with another wire heat identified below. The database search indicated one other plant that has this wire heat in combination with Linde 0091 flux. St. Lucie Unit 2 reported an initial RT_{NDT} of -80°F based on the CE weld material qualification report[5]. Beaver Valley Unit 2 reported an initial RT_{NDT} of -30°F based on a plant specific surveillance weldment tests. Both of these test values are valid data that meet the requirements of ASME Section III NB 2331 and are within the ranges of reported

data that were used to determine the generic RT_{NDT} value of -56°F for Linde 0091 flux welds[6]. This variation should be of little concern for this low copper, low nickel weld, since it is more than 250°F below the 10 CFR 50.61 screening limit and more than 140°F below the most limiting St. Lucie Unit 2 beltline material at the end of license. Even substituting the generic value of -56°F or -30°F this material will not become limiting. Since no guidance exists for treatment of multiple initial RT_{NDT} data, and this is a non limiting weld, the St. Lucie Unit 2 value of -80°F initial RT_{NDT} should be applicable for the intermediate axial shell welds 101-124 A & B since it meets the requirements of ASME Section III NB 2331.

Heat 83637/Linde 0091/Lot 1122: This non limiting weld heat was used in the fabrication of the St. Lucie Unit 2 vessel lower shell axial welds 101-142A, B & C and the partial repair of the intermediate shell axial weld 101-124C only. The database search indicated one other plant that has this wire heat in combination with Linde 0091 flux. St. Lucie Unit 2 reported an initial RT_{NDT} of -50°F based on the CE weld material qualification report[7]. This value of -50°F is in agreement with the value reported for the other plant for this heat in combination with Linde 0091 flux. An error was noted in the EPRI RPVDATA database for the flux lot and initial RT_{NDT} value reported for the intermediate shell axial weld 101-124C. The correct flux lot is 1122 and the correct initial RT_{NDT} of -50°F will be reported to the organization tasked with the EPRI RPVDATA database update. Since the intermediate shell axial weld 101-124C was fabricated with a single arc, with both wire heat 83642/Linde 0091/lot 3536 with an initial RT_{NDT} of -80°F, and heat 83637/Linde 0091/lot 1122 with an initial RT_{NDT} of -50°F, the conservative higher value of -50°F is the more limiting for this weld for embrittlement predictions.

The St. Lucie Unit 2 value of -50°F initial RT_{NDT} for the lower shell axial welds 101-142A, B, & C and the intermediate shell axial weld 101-124C is justified as a measured initial RT_{NDT} value that meets the requirements of ASME Section III NB 2331, with all other measured values in agreement.

Heat 83637/Linde 124/Lot 0951: This non limiting weld heat was used in the partial fabrication of the St. Lucie Unit 2 vessel intermediate to lower shell girth weld, 101-171, and all of the surveillance weld. The database search indicated that this wire heat in combination with Linde 124 flux was not used at any other plant. The initial RT_{NDT} value reported by CE was -70°F. The same heat and flux combination was used to fabricate the surveillance weld. The initial RT_{NDT} determined from the surveillance baseline specimen weld was -50°F[8] and meets the requirements of ASME Section III NB 2331. This higher value of -50°F is applicable for the intermediate to lower shell weld, 101-171 and is corrected in Table 1 and 2 from Attachment B pages B-4 and B-5[1].

Heat 3P7317/Linde 124/0951: This non limiting weld heat was used in the partial fabrication of the St. Lucie Unit 2 vessel intermediate to lower shell girth weld, 101-171. The database search indicated that this wire heat in combination with Linde 124 flux was also used to fabricate one other plant. Even though the flux lot is different, both plants have plant specific values of -80°F reported for the initial RT_{NDT}. However, since this heat was also used to fabricate the intermediate to lower shell weld with wire heat 83637/Linde 124/lot 0951 flux, the higher value

of -50°F is more limiting for determining embrittlement predictions for the intermediate to lower shell girth weld , 101-171.

NRC Request 3:

Provide the neutron flux for the BV-1 surveillance capsule and the SL-1 vessel to clarify the conclusion that the neutron flux values are within an order of magnitude of each other.

FPL Response 3:

Table 3 of Attachment A, page A-13 of the subject PTS submittal[1] has been revised to indicate the last column as flux (E> 1MeV). The fluence at the St. Lucie Unit 1 limiting weld and vessel maximums has also been updated. Comparison of the neutron flux data for the Beaver Valley Unit 1 surveillance capsules and the St. Lucie Unit 1 limiting weld shows that both are within the same order of magnitude.

NRC Request 4:

Provide details on the equivalence of the BV-1 neutron spectra to that of SL-1 to quantify the conclusion on page A-6 of the submittal that "the irradiation behavior of the BV-1 surveillance specimen [is] therefore comparable to the SL-1 limiting weld material."

FPL Response 4:

The equivalence of the BV-1 neutron spectra to the SL-1 neutron spectra can be determined by comparison of the neutron spectra analyses for the last surveillance capsule removed from both units. The capsule analyses represent the most realistic/accurate comparison since they are measured data points. Each capsule report identifies the flux and D.A. at the capsules and transports the measured capsule analysis to the reactor vessel base metal clad interface. The BV-1 capsule spectra as compared to the SL-1 vessel wall at the peak and critical weld locations are provided in the following table:

[illegible]

Plant (ref. noted)	Location	Φ Flux, n/cm²-sec (E> 1 MeV)	D.A./sec	Damage Ratio, D.A./sec/Flux
BV-1 [9]	Center of Capsule W	5.08E10 (Avg. cycle 1-6)	8.38E-11	1.65E-21
BV-1 [9]	Vessel @ 0° (max flux)	4.60E10 (Avg. cycle 1-6)	7.49E-11	1.63E-21
SL-1 [10]	Center of Capsule 104°	3.00E10 (Avg. cycle 1-9)	4.52E-11	1.51E-21
SL-1 [10]	Vessel @ 15° (critical weld)	1.97E10 (Avg. cycle 1-9)	3.00E-11	1.52E-21
SL-1 [10]	Vessel @ 0° (max flux)	3.14E10 (Avg. cycle 1-9)	4.71E-11	1.50E-21

Regardless of the different nuclear data used in the discrete ordinate transport analyses and different reactor internals, it can be concluded that the irradiation behaviors of the BV-1 capsule, which contains the SL-1 critical weld material and the SL-1 vessel at the critical weld location (1.65E-21/1.52E-21) are similar within 9%. Therefore, the BV-1 surveillance data are appropriately applicable to the SL-1 vessel data.

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
Inter. Shell Plate (M-605-1)	A-8490-2	NA	0.11	0.61	+30
Inter. Shell Plate (M-605-2)	B-3416-2	NA	0.13	0.62	+10
Inter. Shell Plate (M-605-3)	A-8490-1	NA	0.11	0.61	0
Inter. Shell Axial Welds (101-124 A,B,C)	83642	Linde 0091/3536	0.04	0.07	-80
Inter. Shell Axial Welds (101-124C)	83637	Linde 0091/1122	0.04	0.07	-50
Lower Shell Axial (101-142 A,B,C)	83637	Linde 0091/1122	0.05	0.10	-50
Inter. to Lower Shell Girth Weld (101-171)	**3P7317**	**Linde 124/0951**	**0.07**	**0.08**	**-80**
Inter. to Lower Shell Girth Weld (101-171)	**83637**	**Linde 124/0951**	**0.07**	**0.08**	**-70**
Surveillance Weld represents Weld (101-171)	**83637**	**Linde 124/0951**	**0.07**	**0.08**	**-50***

* This most limiting value applicable for weld 101-171.

Changes to the original submittal[1] table are shown in bold

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/cm²	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
Inter. 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
Inter. 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
Inter. 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
Inter. Shell Axial Welds (101-124 A,B,C)	83642 Linde 0091 Lot 3536	0.04	0.07	30.7	-80°F	56°F	2.76E19	39°F	15°F	270°F
Inter. Shell Axial Weld (101-124C)	83637 Linde 0091 Lot 1122	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 Linde 0091 Lot 1122	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 Linde 124 Lot 0951**	**0.07**	**0.08**	**41.2**	**-70°F**	**56°F**	**2.76E19**	**52°F**	**38°F**	**300°F**
Inter. to Lower Shell Girth Weld(101-171)	**3P7317 Linde 124 Lot 0951**	**0.07**	**0.08**	**41.2**	**-80°F**	**56°F**	**2.76E19**	**52°F**	**28°F**	**300°F**
Inter. to Lower Shell Surveillance weld	**83637 Linde 124 Lot 0951**	**0.07**	**0.08**	**41.2**	**-50°F****	**56°F**	**2.76E19**	**52°F**	**58°F**	**300°F**

**** Limiting property for the inter. to lower shell girth weld 101-171.   Changes to the original submittal[1] table are shown in bold.**

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/cm²	EFPY	EFPS	Fluence, f/EFPY	Φ Flux, n/cm²-s
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 maximum	1.72 E19	14.84	4.68 E08	1.16 E18	3.68 E10
SL-1	Vessel limiting welds	1.06 E19	14.84	4.68 E08	7.14 E17	2.26 E10

EFPY= Effective full power years
EFPS= Effective full power seconds
Changes to the original submittal[1] table are shown in bold.

Endnotes

1. FPL Letter, L-96-112, "10 CFR 50.61 Evaluation of Pressurizer Thermal Shock of Reactor Vessel Beltline Materials," W. H. Bohlke to NRC, May 14, 1996.

2. Duquesne Light Letter, ND1MNE:7589, "Reactor Vessel Inlet Temperature for Beaver Valley Unit 1," L.R. Freeland to R. S. Boggs, FPL, November 7, 1996.

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

4. Commonwealth Edison Letter, ZNLD/1790, "Dresden..., QuadCities..., LaSalle County Units 1 and 2 .... Response to Generic Letter (GL) 92-01, Reactor Vessel Structural Integrity, Revision 1," Marcia A. Jackson to Dr. Thomas E. Murley, Director, USNRC, July 1, 1992.

5. "Atypical Weld Material In Reactor Pressure Vessel Welds," CE Response to I&E Bulletin 78-12, June 8, 1979, Section VIII, Page 61.

6. "Evaluation of PTS Effects Due to Small Break LOCA's with Loss of Feedwater for the Combustion Engineering NSSS," Section 6.4, CE Owners Group, December 1981, CEN-189.

7. "Atypical Weld Material In Reactor Pressure Vessel Welds" CE Response to I&E Bulletin 78-12, June 8, 1979, Section VIII, Page 69.

8. "Analysis of Capsule W-83, Florida Power and Light Co. St. Lucie Unit No. 2", Babcock & Wilcox, September 1985, BAW-1880.

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, Table 6.1.

10. "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.