Florida Power
A **Progress Energy** Company

Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.90

August 13, 2002
3F0802-05

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Crystal River Unit 3 - Response to Request for Additional Information Re: Proposed License Amendment Request #270, Revision 0, "Power Uprate to 2568 MWt" (TAC No. MB5289)

References:
1. NRC letter to FPC, Dated August 1, 2002, Crystal River Unit 3 - Request for Additional Information Re: Proposed License Amendment Request #270, Revision 0, "Power Uprate to 2568 MWt" (TAC No. MB5289)
2. FPC letter to NRC, Dated June 5, 2002, Crystal River Unit 3 - License Amendment Request #270, Revision 0, "Power Uprate to 2568 MWt"

Dear Sir:

In Reference 1, the NRC provided a Request for Additional Information (RAI) regarding Florida Power Corporation's (FPC) proposed License Amendment Request #270, "Power Uprate to 2568 MWt" (Reference 2). This letter provides a response to that request.

Attachment A provides the responses to the RAI. Attachment B and Attachment D provide Framatome ANP documents that were requested in Question 5. These documents were listed as references in our submittal (Reference 2).

Attachment D contains information that Framatome ANP considers to be proprietary. Framatome ANP requests that the proprietary information in this response (Attachment D) be withheld from public disclosure in accordance with 10 CFR 9.17(a)(4), 2.790(a)(4) and 2.790(d)(1). An affidavit supporting this request is provided in Attachment C.

This letter makes no new regulatory commitments.

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

A001

Sincerely,

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

DEY/pei

Attachments:

A. Response to Request for Additional Information

B. FRA-ANP 32-5013936-01, "Adjusted Reference Temperature for 32 EFPY for CR-3 Power Uprate"

C. Framatome ANP Affidavit of Proprietary Information

D. FRA-ANP 86-1266133-01, "CR-3 PT Fluence Analysis Report – Cycles 7-10"

xc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager

STATE OF FLORIDA

COUNTY OF CITRUS

Dale E. Young states that he is the Vice President, Crystal River Nuclear Plant for Progress Energy; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 13th day of August, 2002, by Dale E. Young.

Lisa A Morris

Signature of Notary Public
State of Florida

LISA A. MORRIS
Notary Public, State of Florida
My Comm. Exp. Oct. 25, 2003
Comm. No. CC 879691

LISA A MORRIS

(Print, type, or stamp Commissioned Name of Notary Public)

Personally Known X -OR- Produced Identification ______

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT A

Response To Request For Additional Information Re: Proposed License Amendment Request #270, Revision 0, Power Uprate to 2568 MWt

Response to Request for Additional Information

Response to Request for Additional Information

NRC Request:

1. **As seen in Section 4.12.1 of its submittal dated June 5, 2002, Florida Power Corporation (FPC, the licensee) did not perform containment integrity analyses for Crystal River, Unit 3. FPC stated in the submittal that the original calculations that provided the mass and energy release data for the reactor building pressure analysis was performed, but were not located. However, FPC did not explain in the submittal whether they have performed a long-term Loss of Coolant Accident (LOCA) analysis for the uprated power to determine the acceptability of the containment safeguards systems to mitigate the consequences of a hypothetical large break LOCA. The staff needs this information to perform its review. Please provide information for the reactor building integrity analysis. Discuss the assumptions, computer code used, and calculation results for the containment accident pressure responses at the stretch rated power level.**

FPC Response:

No new containment integrity analysis was required to be performed for the power uprate project. The original containment integrity calculation (Gilbert 4203-00-212, December 5, 1973) established that the CR-3 containment was designed to withstand an internal pressure of 55 psig with a 1.5 safety factor. The existing CR-3 Large Break Loss-of-Coolant (LBLOCA) Containment analysis was performed using the CONTEMPT computer code (BAW-10095A, Revision 1) using an input of 102% of 2568 MWt. Decay heat was based on NRC Branch Technical Position APCSB 9-2. The mass and energy releases are listed in FSAR Table 14-49. Other relevant parameters are listed in FSAR Tables 14-44, 14-45 and 14-46. Additional details of the containment design basis analysis are in FSAR Section 14.2.2.5.9. The peak containment temperature was determined to be 278.4 °F and peak pressure 54.2 psig. These values were used in the environmental qualification analysis. No changes to these peak values or for the temperature or pressure profiles were required due to the power uprate.

NRC Request:

2. **Section 4.8.1 of FPC's June 5, 2002, submittal states that the main steam system is not impacted by the uprate. FPC should explain whether the main steam system and its components to operate at 2568 MWt were enveloped by the original design. Please provide information on maximum flow rate and steam pressure corresponding to the thermal power rate originally designed. Explain whether the safety relief valves are capable to support the operation at the stretch power conditions with increased flow rate.**

FPC Response:

The design of the main steam system and components can accommodate operation at 2568 MWt. The total code safety relief valve capacity is over 13.0 million pounds mass per hour (Mlbm/hr), which provides more than the capacity required for the design transient, a turbine trip from 112% of 2568 MWt core power (which requires 11.1 Mlbm/hr). Additional details about the main steam system design are in FSAR Sections 10.2.1.4 and 10.3.4.

NRC Request:

3. **Section 4.8.18 of FPC's June 5, 2002, submittal states that the nuclear services and decay heat seawater (RW) system provides cooling water to the nuclear services closed cycle cooling (SW) system and decay heat closed cycle cooling (DC) system for heat removal during accidents and normal operation. FPC indicated in the submittal that the power uprate has no impact on this system based on its accident analyses. However, at uprated power level, the plant heat discharges to the RW system during normal operation and after accidents increase. The RW system draws seawater from the Gulf of Mexico to cool these safety-related cooling systems. The seawater temperature in the summer for the heat sink is normally higher in the Gulf area. Therefore, at the stretch rated power level, the service water temperature in SW and DC systems will increase from current level. Please explain whether the temperature variation of heat sinks resulting from weather changes was considered in the accident analysis.**

FPC Response:

The Improved Technical Specification (ITS) Limiting Condition for Operation 3.7.11 requires that the UHS temperature be below 95 °F and if not, requires placing the plant in Mode 3 (hot standby) in 6 hours, and in Mode 5 (cold shutdown) in 36 hours. The accident analyses was performed assuming a maximum ultimate heat sink (UHS) (Gulf of Mexico) temperature of 95 °F with reactor power at 102% of 2568 MWt. As long as the UHS remains below this temperature, the RW and SW systems are capable of rejecting the maximum post-accident heat loads to the UHS. Therefore, with the UHS below 95 °F, the accident analysis remains valid and bounds the power uprate. The impact of the increase in heat load during normal operations is bounded by the much larger accident heat loads. No changes to the accident analyses were required for the power uprate.

NRC Request:

4. **FPC did not evaluate the effects of the proposed stretch rated power on equipment qualification to meet 10 CFR 50.49 requirements. Since this is a license amendment for power uprate, the licensee is required to confirm that the temperature profile previously approved for equipment qualification bounds the environmental conditions resulting from LOCA or Main Steam Line Break at the higher power level.**

FPC Response:

Environmental Qualification

Re-evaluation of the electrical equipment important to safety to meet 10CFR50.49 requirements was completed which confirmed that the temperature and radiation profiles (accident analyses), performed at 102 % of 2568 MWt, remains valid and bounds the current requested power uprate effort.

In addition to the effects of temperature, the effects of uprated radiation conditions was also re-evaluated assuming the following bounding conditions:

- Reactor core operates for 700 effective full power days per cycle
- Reactor thermal power is 2619 MWt (102% of 2568 MWt)
- Average fuel burnup is 45,000 MegaWatt days per metric ton of uranium

Fission product activity was calculated using FTI Topical report BAW-1512, "BURPE – A Computer Program for Estimating Fission Product Activity in PWRs," Revision K, June 1993. The airborne source in containment used Regulatory Guide 1.4 assumptions. The waterborne activity in the sump and primary coolant used Regulatory Guide 1.89 assumptions.

This re-evaluation concluded that the assumptions and methods used to meet 10CFR50.49 requirements remain valid and bound the operating conditions proposed by the power uprate.

Margins and Synergistic Effects

The power uprate did not impact how EQ evaluations were performed or any fundamental parameters that were used to determine margins. No new phenomena or considerations were introduced that could impact the evaluation done for synergistic effects used to meet 10CFR50.49 requirements.

Submergence, Aging, and Chemical Effects

The power uprate did not change the chemical or submergence profiles and, therefore, has no effect on these environmental qualification parameters used to meet 10CFR50.49 requirements. In regards to aging, CR-3 uses the Arrhenius calculation methodology and analyses for expected life calculations, which determines the service (qualified) life of electrical equipment important to safety. These qualified life calculations were not affected by the power uprate.

Mechanical Equipment Qualification

In addition to the re-evaluating electrical equipment important to safety (10CFR50.49), mechanical equipment qualification was also addressed. The review concluded that although CR-3 does not have a formal mechanical equipment qualification (MEQ) program, the plant relies upon the original design requirements, design standards (e.g., ASME) and processes, preventative maintenance efforts, periodic inspections, and surveillance testing to ensure that the capability of mechanical equipment will continue to perform its safety function during normal and postulated accident conditions. Therefore, the power uprate effort will not impact mechanical equipment qualification.

It should be noted that CR3 is not required to maintain a formal mechanical equipment qualification (MEQ) program since the plant was licensed prior to the implementation of MEQ program requirements (1980s). MEQ programs were required for some later vintage plants which were licensed to the requirements of 10CFR50, Appendix A, General Design Criteria 4. CR-3 is licensed to the Principal Architectural and Design Criteria listed in FSAR Section 1.4, which do not require a formal MEQ program.

Seismic Qualification

The CR-3 Seismic Design Basis is documented in Section 5.1.2 of the FSAR. Class 1 structures, systems, and components (SSCs) have been seismically qualified and designed in accordance with the requirements of the FSAR. In addition to this original design basis, the NRC Unresolved Safety Issue (USI) A-46 was addressed to ensure all SSCs would perform safe shutdown of the plant following an earthquake. CR-3's Seismic Qualification Utility Group (SQUG) program was developed and implemented to address the USI A-46 concern and confirm that safety-related and non-safety related equipment, considered important to safe shutdown of the plant, is seismically adequate for CR-3 earthquake levels including the current power uprate effort.

__NRC Request:__

5. In its June 5, 2002, submittal FPC stated that it used the following references in its evaluation:

5. **FRA-ANP 86-1266133-01, "ACR-3 PT Fluence Analysis Report - Cycles 7-10"**
6. **FRA-ANP 32-5013936-00, "Adjusted Reference Temperature for 32 EFPY for CR-3 Power Uprate"**

The staff requests these documents in order to complete the evaluation on the changes on pressure vessel fluence.

<u>FPC Response:</u>

FRA-ANP 32-5013936-00 was revised to correct a minor reference error. Therefore, FRA-ANP 32-5013936-01 is included in Attachment B. FRA-ANP 86-1266133-01 is included in Attachment D. The information in Attachment D is considered PROPRIETARY by Framatome ANP as stated in the affidavit in Attachment C.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT B

Response To Request For Additional Information Re: Proposed License Amendment Request #270, Revision 0, Power Uprate to 2568 MWt

FRA-ANP 32-5013936-01, "Adjusted Reference Temperature for 32 EFPY for CR-3 Power Uprate"

FRAMATOME ANP

CALCULATION SUMMARY SHEET (CSS)

Document Identifier 32 - 5013936 - 01

Title ADJUSTED REFERENCE TEMPERATURES FOR 32 EFPY FOR CR-3 POWER UPRATE

PREPARED BY:		REVIEWED BY:	
		METHOD: ☒ DETAILED CHECK ☐ INDEPENDENT CALCULATION	
NAME L. YU		NAME B.R. GRAMBAU	
SIGNATURE		SIGNATURE	
TITLE ENGINEER III	DATE 8/7/02	TITLE SUPV. ENGINEER	DATE 8/7/02
COST CENTER 41020	REF. PAGE(S) 16	TM STATEMENT: REVIEWER INDEPENDENCE	ADM

PURPOSE AND SUMMARY OF RESULTS:

PURPOSE:

The purpose of this analysis is to determine the Florida Power Corporation Crystal River Unit 3 (CR-3) reactor vessel adjusted reference temperature data for 32 effective full power years (32 EFPY) based on increased fluence values in support of CR-3 power uprate to 2619.4 MWt. This information will be used in the preparation of pressure-temperature operating limit curves applicable through 32 EFPY.

SUMMARY OF RESULTS:

The adjusted reference temperature values applicable through 32 EFPY for the CR-3 reactor vessel beltline materials are calculated in accordance with the guidelines outlined in Regulatory Guide 1.99, Revision 2. The controlling values of the adjusted reference temperature for the CR-3 reactor vessel beltline material are 195.7°F at the ¼-thickness (¼T) wall location and 144.1°F at the ¾-thickness (¾T) wall location respectively.

Comments included in Florida Power letter SE01-0195 dated 09/19/2001 were discussed and resolved with S.F. Ulm, Principle Engineer, Systems Engineering on 10/09/2001 via telecon. Words were added to clarify the 7% increase of the current fluence values, which is expected to bound the actual increase in fluence resulting from the power upgrade.

PURPOSE OF REV. 01:
The purpose of this revision is to update the revision numbers of references 2, 3 and 7.

SUMMARY OF RESULTS OF REV. 01:
Same as rev. 00. The revision does not impact the calculation performed in Rev. 00.

THE FOLLOWING COMPUTER CODES HAVE BEEN USED IN THIS DOCUMENT:		THE DOCUMENT CONTAINS ASSUMPTIONS THAT MUST BE VERIFIED PRIOR TO USE ON SAFETY-RELATED WORK
CODE/VERSION/REV	CODE/VERSION/REV	☐ YES ☒ NO

RECORD OF REVISION

Revision	Description
00	Original Release
01	Updated the revision numbers of references 2, 3 and 7. Page affected: Page 1 – Added purpose and summary of results for Rev. 01. Page 2 – Added description for Rev. 01 in record of revision. Page 8 – Updated revision number for BAW-2313. Page 16 – Updated revision numbers for references 2, 3 and 7.

TABLE OF CONTENTS

1.0 INTRODUCTION

The purpose of this analysis is to determine the Florida Power Corporation Crystal River Unit 3 (CR-3) reactor vessel adjusted reference temperature data for 32 effective full power years (32 EFPY) based on increased fluence values in support of the CR-3 power uprate to 2619.4 MWt. This information will be used in the preparation of pressure-temperature operating limit curves applicable through 32 EFPY for the CR-3 power uprate.

2.0 SUMMARY OF RESULTS

The adjusted reference temperature values applicable through 32 EFPY for the CR-3 reactor vessel beltline materials are listed in Table 1 for ¼-thickness (¼T) locations and Table 2 for ¾-thickness (¾T) locations. These values were calculated in accordance with the guidelines outlined in Regulatory Guide 1.99, Revision 2.[1] The controlling values of the adjusted reference temperatures for the CR-3 reactor vessel beltline material are 195.7°F at the ¼T wall location and 144.1°F at the ¾T wall location.

3.0 ASSUMPTIONS

No major assumptions are contained in this report.

4.0 REACTOR VESSEL FLUENCE

4.1 Reactor Vessel Inside Surface Fluence

The extrapolated 32 EFPY inside surface fluences for the CR-3 reactor vessel beltline materials were calculated in the calculation, *CR-3 Power Uprate PTS Evaluation*[2]. They are listed in Table 3. It is noted that the inside surface fluences listed in Table 3 have been increased 7% from previously reported values[3] to account for the power uprate to 2619.4 MWt, which is expected to bound the actual increase in fluence resulting from the power upgrade.

4.2 Attenuation Through Vessel Wall

In accordance with Regulatory Guide 1.99, Revision 2, the neutron fluence at the ¼T and ¾T wall locations in the vessel, f (x 10^{19} n/cm^2, E>1 MeV), is determined as follows:

$$f = f_{surf}\,(e^{-0.24x}) \qquad (1)$$

where f_{surf} (10^{19} n/cm^2, E>1 MeV) is the neutron fluence at the inner wetted surface of the vessel, and x (in inches) is the depth into the vessel wall measured from the vessel inner (wetted) surface. The CR-3 reactor vessel thickness is reported in BAW-1543, Revision 4[4] to be 8.44 inches. The cladding thickness at the reactor vessel inside wall is 0.125" [8]. Hence, the ¼ T and ¾ T wall locations are 2.235" (2.11"+0.125") and 6.455" (6.33"+0.125") from the vessel inner (wetted) surface respectively. The 32 EFPY fluence values for the CR-3 reactor vessel beltline materials at the ¼T and ¾T wall locations are presented in Table 3.

Table 1. Adjusted Reference Temperatures at ¼T Location for Preparation of Pressure-Temperature Limit Curves for Crystal River Unit 3 -- Applicable Through 32 EFPY

Reactor Vessel Beltline Region Location	Material Ident.	Heat Number	Type	Cu	Ni	Estimated Fluence @ 32 EFPY, n/cm^2 (E > 1.0 MeV) IS	1/4T	Adjusted Reference Temperature Evaluation at 1/4T Location: Chem. Factor	Fluence Factor	Initial RT_{NDT}, F	σ_I	σ_Δ	RT_{NDT} Shift, F	Margin	1/4T ART
Regulatory Guide 1.99, Revision 2, Position 1.1															
Lower Nozzle Belt Forging (LNB)	AZJ 94	123V190	SA-508 Cl. 2	0.13	0.72	7.58E+18	4.43E+18	94.0	0.774	3	31.0	17	72.8	70.7	146.5
Upper Shell Plate (US)	C4344-1	C4344-1	SA-533 Gr. B	0.20	0.54	8.45E+18	4.94E+18	141.8	0.803	20	0	17	113.9	34.0	167.9
Upper Shell Plate (US)	C4344-2	C4344-2	SA-533 Gr. B	0.20	0.54	8.45E+18	4.94E+18	141.8	0.803	20	0	17	113.9	34.0	167.9
Lower Shell Plate (LS)	C4347-1	C4347-1	SA-533 Gr. B	0.12	0.58	8.56E+18	5.01E+18	82.6	0.807	-10	0	17	66.7	34.0	90.7
Lower Shell Plate (LS)	C4347-2	C4347-2	SA-533 Gr. B	0.12	0.58	8.56E+18	5.01E+18	82.6	0.807	45	0	17	66.7	34.0	145.7
LNB to US Circ. Weld (ID 40%)	SA-1769	71249	Linde 80 Flux	0.23	0.59	7.58E+18	4.43E+18	167.6	0.774	10	0.0	28	129.7	56.0	195.7
LNB to US Circ. Weld (OD 60%)	WF-169-1	8T1554	Linde 80 Flux	0.16	0.57	7.58E+18	N/A	143.9	N/A	-5	19.7	28	N/A	68.5	N/A
US Longit. Weld (100%)	WF-8	8T1762	Linde 80 Flux	0.19	0.57	7.92E+18	4.63E+18	152.4	0.786	-5	19.7	28	119.8	68.5	183.3
LS Longit. Weld (100%)	WF-18	8T1762	Linde 80 Flux	0.19	0.57	7.92E+18	4.63E+18	152.4	0.786	-5	19.7	28	119.8	68.5	183.3
US to LS Circ. Weld (100%)	WF-70	72105	Linde 80 Flux	0.32	0.58	8.27E+18	4.84E+18	199.3	0.797	-26	0	28	158.8	56.0	188.8
LS Longit. Welds (Both 100%)	SA-1580	8T1762	Linde 80 Flux	0.19	0.57	7.45E+18	4.36E+18	152.4	0.769	-5	19.7	28	117.2	68.5	180.7
Regulatory Guide 1.99, Revision 2, Position 2.1															
Upper Shell Plate (US)	C4344-1	C4344-1	SA-533 Gr. B	0.20	0.54	8.45E+18	4.94E+18	116.3	0.803	20	0.00	8.5	93.4	17.0	130.4

Table 2. Adjusted Reference Temperatures at ¾T Location for Preparation of Pressure-Temperature Limit Curves for Crystal River Unit 3 -- Applicable Through 32 EFPY

Reactor Vessel Beltline Region Location	Material Ident.	Heat Number	Type	Cu	Ni	Estimated Fluence @ 32 EFPY, n/cm^2 (E > 1.0 MeV) IS	Estimated Fluence @ 32 EFPY, n/cm^2 (E > 1.0 MeV) 3/4T	Chem. Factor	Fluence Factor	Initial RT_{NDT}, F	σ_I	σ_Δ	RT_{NDT} Shift, F	Margin	3/4T ART
Regulatory Guide 1.99, Revision 2, Position 1.1															
Lower Nozzle Belt Forging (LNB)	AZJ 94	123V190	SA-508 Cl. 2	0.13	0.72	7.58E+18	1.61E+18	94.0	0.519	3	31.0	17	48.8	70.7	122.5
Upper Shell Plate (US)	C4344-1	C4344-1	SA-533 Gr. B	0.20	0.54	8.45E+18	1.79E+18	141.8	0.544	20	0	17	77.1	34.0	131.1
Upper Shell Plate (US)	C4344-2	C4344-2	SA-533 Gr. B	0.20	0.54	8.45E+18	1.79E+18	141.8	0.544	20	0	17	77.1	34.0	131.1
Lower Shell Plate (LS)	C4347-1	C4347-1	SA-533 Gr. B	0.12	0.58	8.56E+18	1.82E+18	82.6	0.547	-10	0	17	45.2	34.0	69.2
Lower Shell Plate (LS)	C4347-2	C4347-2	SA-533 Gr. B	0.12	0.58	8.56E+18	1.82E+18	82.6	0.547	45	0	17	45.2	34.0	124.2
LNB to US Circ. Weld (ID 40%)	SA-1769	71249	Linde 80 Flux	0.23	0.59	7.58E+18	N/A	167.6	N/A	10	0.0	28	N/A	56.0	N/A
LNB to US Circ. Weld (OD 60%)	WF-169-1	8T1554	Linde 80 Flux	0.16	0.57	7.58E+18	1.61E+18	143.9	0.519	-5	19.7	28	74.7	68.5	138.2
US Longit. Weld (100%)	WF-8	8T1762	Linde 80 Flux	0.19	0.57	7.92E+18	1.68E+18	152.4	0.529	-5	19.7	28	80.6	68.5	**144.1**
LS Longit. Weld (100%)	WF-18	8T1762	Linde 80 Flux	0.19	0.57	7.92E+18	1.68E+18	152.4	0.529	-5	19.7	28	80.6	68.5	**144.1**
US to LS Circ. Weld (100%)	WF-70	72105	Linde 80 Flux	0.32	0.58	8.27E+18	1.76E+18	199.3	0.539	-26	0	28	107.4	56.0	137.4
LS Longit. Welds (Both 100%)	SA-1580	8T1762	Linde 80 Flux	0.19	0.57	7.45E+18	1.58E+18	152.4	0.515	-5	19.7	28	78.5	68.5	142.0
Regulatory Guide 1.99, Revision 2, Position 2.1															
Upper Shell Plate (US)	C4344-1	C4344-1	SA-533 Gr. B	0.20	0.54	8.45E+18	1.79E+18	116.3	0.544	20	0.00	8.5	63.3	17.0	100.3

Table 3. Crystal River Unit 3 Reactor Vessel Beltline Materials Fluence Values at 32 EFPY

Beltline Materials	Material Ident.	32 EFPY Fluence n/cm^2 (E > 1.0 MeV)		
		Inside Surface	¼ T (x = 2.235 in.)	¾ T (x = 6.455 in.)
Lower Nozzle Belt Forging	AZJ 94	7.58E+18	4.43E+18	1.61E+18
Upper Shell Plate	C4344-1	8.45E+18	4.94E+18	1.79E+18
Upper Shell Plate	C4344-2	8.45E+18	4.94E+18	1.79E+18
Lower Shell Plate	C4347-1	8.56E+18	5.01E+18	1.82E+18
Lower Shell Plate	C4347-2	8.56E+18	5.01E+18	1.82E+18
LNB to US Circ. Weld (ID 40%)	SA-1769	7.58E+18	4.43E+18	----
LNB to US Circ. Weld (OD 60%)	WF-169-1	----	----	1.61E+18
US Longit. Weld (100%)	WF-8	7.92E+18	4.63E+18	1.68E+18
US Longit. Weld (100%)	WF-18	7.92E+18	4.63E+18	1.68E+18
US to LS Circ. Weld (100%)	WF-70	8.27E+18	4.84E+18	1.76E+18
LS Longit. Weld (Both 100%)	SA-1580	7.45E+18	4.36E+18	1.58E+18

5.0 ADJUSTED REFERENCE TEMPERATURE CALCULATION WHERE NO SURVEILLANCE DATA IS AVAILABLE

The following information is required for determination of the adjusted reference temperature outlined in Regulatory Guide 1.99, Revision 2.

5.1 Initial RT_{NDT}

The initial RT_{NDT} is the reference temperature for the unirradiated base metal vessel beltline material as defined in Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code.[5] If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

The initial RT_{NDT} of WF-70 is determined using an alternative method based on the material fracture toughness in the transition range. This method is described in BAW-2202.[6]

Table 4 lists the initial RT_{NDT} values for the CR-3 reactor vessel beltline materials and their applicable sources.

Table 4. Crystal River Unit 3 Initial RT_{NDT} Values for Reactor Vessel Beltline Materials

Beltline Materials	Material Ident.	Initial RT_{NDT}, F	Reference Source
Lower Nozzle Belt Forging	AZJ 94	+3	Estimated value (BAW-2313, Rev. 3[7])
Upper Shell Plate	C4344-1	+20	Measured value (BAW-2313, Rev. 3)
Upper Shell Plate	C4344-2	+20	Measured value (BAW-2313, Rev. 3)
Lower Shell Plate	C4347-1	-10	Measured value (BAW-2313, Rev. 3)
Lower Shell Plate	C4347-2	+45	Measured value (BAW-2313, Rev. 3)
LNB to US Circ. Weld (ID 40%)	SA-1769	+10	Measured value (BAW-2313, Rev. 3)
LNB to US Circ. Weld (OD 60%)	WF-169-1	-5	Generic value (BAW-2313, Rev. 3)
US Longit. Weld (100%)	WF-8	-5	Generic value (BAW-2313, Rev. 3)
US Longit. Weld (100%)	WF-18	-5	Generic value (BAW-2313, Rev. 3)
US to LS Circ. Weld (100%)	WF-70	-26	Measured value (BAW-2202[6])
LS Longit. Welds (Both 100%)	SA-1580	-5	Generic value (BAW-2313, Rev. 3)

5.2 ΔRT_{NDT}

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and is calculated as follows:

$$\Delta RT_{NDT} = (CF) * (ff) \qquad (2)$$

where: CF = Chemistry Factor
ff = fluence factor

5.2.1 Chemistry Factor

The chemistry factor (CF) is a function of the material's copper and nickel content. The CF is determined from Table 1 (for weld metals) and Table 2 (for base metals) in Regulatory Guide 1.99, Revision 2. Linear interpolation is permitted. When determining the CF, the "weight percent copper" and "weight percent nickel" are best estimate values for the material, which will normally be the mean of the measured values for the material.

The copper and nickel contents for the beltline materials are reported in BAW-2313, Revision 3 [7].

Using Tables 1 and 2 in Regulatory Guide 1.99, Revision 2, the CF's for the CR-3 reactor vessel beltline region materials are calculated and listed in Table 5.

Table 5. Regulatory Guide 1.99, Revision 2, Chemistry Factors for Crystal River Unit 3 Reactor Vessel Beltline Materials

Beltline Materials	Material Ident.	Cu wt%	Ni wt%	Chemistry Factor
Lower Nozzle Belt Forging	AZJ 94	0.13*	0.72	94.0
Upper Shell Plate	C4344-1	0.20	0.54	141.8
Upper Shell Plate	C4344-2	0.20	0.54	141.8
Lower Shell Plate	C4347-1	0.12	0.58	82.6
Lower Shell Plate	C4347-2	0.12	0.58	82.6
LNB to US Circ. Weld (ID 40%)	SA-1769	0.23	0.59	167.6
LNB to US Circ, Weld (OD 60%)	WF-169-1	0.16	0.57	143.9
US Longit. Weld (100%)	WF-8	0.19	0.57	152.4
US Longit. Weld (100%)	WF-18	0.19	0.57	152.4
US to LS Circ. Weld (100%)	WF-70	0.32	0.58	199.3
LS Longit. Welds (Both 100%)	SA-1580	0.19	0.57	152.4

* Note: this is generic copper content for material AZJ 94[7].

5.2.2 Fluence Factor

In accordance with Regulatory Guide 1.99, Revision 2, the fluence factor (ff) for the ¼T and ¾T wall locations is determined as follows:

$$ff = f^{\,(0.28-0.10\log f)} \qquad (3)$$

Table 6 lists the fluence factors for the ¼T and ¾T wall locations for the CR-3 reactor vessel beltline materials at 32 EFPY.

Table 6. Fluence Factors for the ¼T and ¾T Wall Locations of the Crystal River Unit 3 Reactor Vessel Beltline Region

Beltline Materials	Material Ident.	¼T Location		¾T Location	
		Fluence, n/cm^2 (E > 1.0 MeV) (x 10^{19})	Fluence Factor	Fluence, n/cm^2 (E > 1.0 MeV) (x 10^{19})	Fluence Factor
Lower Nozzle Belt Forging	AZJ 94	0.443	0.774	0.161	0.519
Upper Shell Plate	C4344-1	0.494	0.803	0.179	0.544
Upper Shell Plate	C4344-2	0.494	0.803	0.179	0.544
Lower Shell Plate	C4347-1	0.501	0.807	0.182	0.547
Lower Shell Plate	C4347-2	0.501	0.807	0.182	0.547
LNB to US Circ. Weld (ID 40%)	SA-1769	0.443	0.774	----	----
LNB to US Circ. Weld (OD 60%)	WF-169-1	----	----	0.161	0.519
US Longit. Weld (100%)	WF-8	0.463	0.786	0.168	0.529
US Longit. Weld (100%)	WF-18	0.463	0.786	0.168	0.529
US to LS Circ. Weld (100%)	WF-70	0.484	0.797	0.176	0.539
LS Longit. Welds (Both 100%)	SA-1580	0.436	0.769	0.158	0.515

5.2.3 ΔRT_{NDT} Calculation

The ΔRT_{NDT} values for the CR-3 reactor vessel beltline materials are calculated by multiplying the chemistry factors and fluence factors. The 32 EFPY ΔRT_{NDT} values for the CR-3 reactor vessel beltline materials are presented in Table 7.

Table 7. ΔRT_{NDT} Values for the Crystal River Unit 3 Reactor Vessel Beltline Materials

Beltline Materials	Material Ident.	1/4 T Location			3/4T Location		
		CF	ff	ΔRT_{NDT}	CF	ff	ΔRT_{NDT}
Lower Nozzle Belt Forging	AZJ 94	94.0	0.774	72.8	94.0	0.519	48.8
Upper Shell Plate	C4344-1	141.8	0.803	113.9	141.8	0.544	77.1
Upper Shell Plate	C4344-2	141.8	0.803	113.9	141.8	0.544	77.1
Lower Shell Plate	C4347-1	82.6	0.807	66.7	82.6	0.547	45.2
Lower Shell Plate	C4347-2	82.6	0.807	66.7	82.6	0.547	45.2
LNB to US Circ. Weld (ID 40%)	SA-1769	167.6	0.774	129.7	167.6	----	----
LNB to US Circ. Weld (OD 60%)	WF-169-1	143.9	----	----	143.9	0.519	74.7
US Longit. Weld (100%)	WF-8	152.4	0.786	119.8	152.4	0.529	80.6
US Longit. Weld (100%)	WF-18	152.4	0.786	119.8	152.4	0.529	80.6
US to LS Circ. Weld (100%)	WF-70	199.3	0.797	158.8	199.3	0.539	107.4
LS Longit. Welds (Both 100%)	SA-1580	152.4	0.769	117.2	152.4	0.515	78.5

5.3 Margin

The "margin" is the quantity that is added to obtain conservative, upper-bound values of the adjusted reference temperature. The margin is determined by the following expression:

$$Margin = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2} \quad (4)$$

where σ_I = standard deviation for the initial RT_{NDT}
σ_Δ = standard deviation for ΔRT_{NDT}

If a measured value of initial RT_{NDT} for the material in question is available, σ_I is to be estimated from the precision of the test method. If generic mean values are used, σ_I is the standard deviation obtained from the set of data used to establish the mean.

The standard deviation for the calculated ΔRT_{NDT}, σ_Δ, is 28°F for welds and 17°F for base metals, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}.

Table 8 list the margin values calculated for the CR-3 reactor vessel beltline materials through 32 EFPY.

Table 8. Margin Values for the Crystal River Unit 3 Reactor Vessel Beltline Materials

Beltline Materials	Material Ident.	σ_I	σ_Δ	ΔRT_{NDT} / 2		Margin	
				¼T	¾T	¼T	¾T
Lower Nozzle Belt Forging	AZJ 94	31 [7]	17	36.4	24.4	70.7	70.7
Upper Shell Plate	C4344-1	0 [7]	17	57.0	38.6	34.0	34.0
Upper Shell Plate	C4344-2	0 [7]	17	57.0	38.6	34.0	34.0
Lower Shell Plate	C4347-1	0 [7]	17	33.4	22.6	34.0	34.0
Lower Shell Plate	C4347-2	0 [7]	17	33.4	22.6	34.0	34.0
LNB to US Circ. Weld (ID 40%)	SA-1769	0 [7]	28	64.9	----	56.0	----
LNB to US Circ. Weld (OD 60%)	WF-169-1	19.7 [7]	28	----	37.4	----	68.5
US Longit. Weld (100%)	WF-8	19.7 [7]	28	59.9	40.3	68.5	68.5
US Longit. Weld (100%)	WF-18	19.7 [7]	28	59.9	40.3	68.5	68.5
US to LS Circ. Weld (100%)	WF-70	0 [6]	28	79.4	53.7	56.0	56.0
LS Longit. Welds (Both 100%)	SA-1580	19.7 [7]	28	58.6	39.3	68.5	68.5

5.4 Calculation of Adjusted Reference Temperature

The adjusted reference temperature (ART) is given by the following expression:

$$ART = Initial\ RT_{NDT} + \Delta RT_{NDT} + Margin \quad (5)$$

Table 9 lists the ¼T and ¾T ART values calculated for the CR-3 reactor vessel beltline materials through 32 EFPY.

Table 9. ¼T and ¾T ART Values for the Crystal River Unit 3 Reactor Vessel Beltline Materials

Beltline Materials	Material Ident.	Adjusted Reference Temperature, F	
		T/4	3/4T
Lower Nozzle Belt Forging	AZJ 94	146.5	122.5
Upper Shell Plate	C4344-1	167.9	131.1
Upper Shell Plate	C4344-2	167.9	131.1
Lower Shell Plate	C4347-1	90.7	69.2
Lower Shell Plate	C4347-2	145.7	124.2
LNB to US Circ. Weld (ID 40%)	SA-1769	**195.7**	----
LNB to US Circ. Weld (OD 60%)	WF-169-1	----	138.2
US Longit. Weld (100%)	WF-8	183.3	**144.1**
US Longit. Weld (100%)	WF-18	183.3	**144.1**
US to LS Circ. Weld (100%)	WF-70	188.8	137.4
LS Longit. Welds (Both 100%)	SA-1580	180.7	142.0

6.0 ADJUSTED REFERENCE TEMPERATURE CALCULATION WHERE SURVEILLANCE DATA IS AVAILABLE

Results from plant specific surveillance programs may be integrated into the adjusted reference temperature estimate if the surveillance data have been deemed credible as judged by the following criteria:

1. Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement according to the recommendations of Regulatory Guide 1.99, Revision 2.

2. Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature unambiguously.

3. When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Guide 1.99, Revision 2, Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values.

4. The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within ±25°F.

5. The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the database for that material.

When two or more credible surveillance data sets are available, these data may be used to determine the adjusted reference temperature of the reactor vessel beltline materials as follows:

First, if there is clear evidence that the copper or nickel content of the surveillance weld differs from that of the reactor vessel weld, the measured values of ΔRT_{NDT} should be adjusted by multiplying the values by the ratio of the chemistry factor for the reactor vessel weld to that for the surveillance weld. Second, using the ΔRT_{NDT} and its corresponding fluence, the chemistry factor may be calculated by multiplying each adjusted ΔRT_{NDT} by the corresponding fluence factor, summing the products, and dividing by the sum of the squares of the fluence factors:

$$CF = \frac{\sum \Delta RT_{NDT} * ff}{\sum ff^2} \tag{6}$$

The CR-3 plant specific reactor vessel surveillance program (RVSP) provides data for predicting the reference temperature shift for the base metal plate heat number C4344-1. In addition, the Master Integrated Reactor Vessel Surveillance Program (MIRVP) described in BAW-1543, Revision 4,[4] provides surveillance data for weld metals SA-1769 and WF-70 on predicting the reference temperature shift.

The calculation, *CR-3 Power Uprate PTS Evaluation*[2], evaluated the credibility of these three materials. It was found that the surveillance data were credible for base metal plate of heat number C4344-1, but the data were not credible for weld metals SA-1769 heat number 71249 and WF-70 heat number 72105.

Hence, the adjusted reference temperature based on surveillance data needs to be calculated for base metal plate heat number C4344-1.

6.1 The Chemistry Factor Using Surveillance Data

Appendix A of Reference 2 made an assessment of base metal C4344-1 surveillance data. The surveillance data were found to be credible per Regulatory Guide 1.99, Rev. 2. The chemistry factor was calculated as 116.3.

6.2 Adjusted Reference Temperature Calculated Using Surveillance Data

6.2.1 Initial RT_{NDT}

The initial RT_{NDT} for base metal plate of heat number C4344-1 is the same as specified in Section 5.1.

Beltline Materials	Initial RT_{NDT}, F
Upper Shell Plate (C4344-1)	+20

6.2.2 ΔRT_{NDT} Calculation

The ¼T and ¾T ΔRT_{NDT} values for the beltline region materials where surveillance data are available are calculated in accordance with Equation 2 using the chemistry factor determined in Section 6.1 and the corresponding fluence factors in Table 6. The 32 EFPY ¼T and ¾T ΔRT_{NDT} values for the CR-3 reactor vessel beltline region materials using credible surveillance data are presented in Table 10.

Table 10. ΔRT_{NDT} Values at 32 EFPY for the CR-3 Reactor Vessel Beltline Materials Using Surveillance Data

Beltline Materials	Material Ident.	¼T Location			¾T Location		
		CF	ff	ΔRT_{NDT}	CF	ff	ΔRT_{NDT}
Upper Shell Plate	C4344-1	116.3	0.803	93.4	116.3	0.544	63.3

6.2.3 Margin

To calculate the margin values for beltline materials with surveillance data available, Equation 4 is used. Table 11 lists the margin values calculated for the CR-3 vessel beltline materials where surveillance data available.

Table 11. Margin Values for the CR-3 Reactor Vessel Beltline Materials Using Surveillance Data

Beltline Materials	Material Ident.	σ_I	σ_Δ	Margin	
				¼T	¾T
Upper Shell Plate	C4344-1	0	8.5	17	17

6.2.4 Calculation of Adjusted Reference Temperature

The ART is calculated using Equation 5. Table 12 lists the ¼T and ¾T ART values through 32 EFPY calculated for the CR-3 reactor vessel beltline materials where surveillance data available.

Table 12. ¼T and ¾T ART Values Through 32 EFPY for the CR-3 Reactor Vessel Beltline Materials Using Surveillance Data

Beltline Materials	Material Ident.	Adjusted Reference Temperature, F	
		¼T	¾T
Upper Shell Plate	C4344-1	130.4	100.3

7.0 REFERENCES

1. U.S. Nuclear Regulatory Commission, *"Radiation Damage to Reactor Vessel Material,"* Regulatory Guide 1.99, Revision 2, May 1988.

2. FRA-ANP Document 32-5013892-02, *"CR-3 Power Uprate PTS Evaluation,"* August 2002.

3. FRA-ANP Document 86-1266133-01, *"CR-3 PT Fluence Analysis Report Cycles 7 – 10,"* August 1998.

4. L. S. Harbison, *"Master Integrated Reactor Vessel Surveillance Program,"* BAW-1543, Revision 4, B&W Nuclear Technologies, Inc., Lynchburg, Virginia, February 1993.

5. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, *Nuclear Power Plant Components,* Subsection NB, Class 1 Components.

6. K. K. Yoon, *"Fracture Toughness Characterization of WF-70 Weld Metal,"* BAW-2202, B&W Nuclear Technologies, Inc., Lynchburg, Virginia, September 1993.

7. J.B. Hall, "*B&WOG Reactor Vessel Working Group Reactor Vessel Materials and Surveillance Data Information, Volume 1 and 2*," BAW-2313, Revision 3, Framatome ANP, Inc., Lynchburg, Virginia, May 2002.

8. B&W Drawing 02-135538E-05, "Shell Assy and Head Details," March 1971.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT C

Response To Request For Additional Information Re: Proposed License Amendment Request #270, Revision 0, Power Uprate to 2568 MWt

Framatome ANP Affidavit of Proprietary Information

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is James F. Mallay. I am Director, Regulatory Affairs, for Framatome ANP ("FRA-ANP"), and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by FRA-ANP to determine whether certain FRA-ANP information is proprietary. I am familiar with the policies established by FRA-ANP to ensure the proper application of these criteria.

3. I am familiar with the information contained in a fluence analysis report, which is designated as 86-1266133-01, provided to the NRC by Florida Power Corp. and referred to herein as "Document." Information contained in this Document has been classified by FRA-ANP as proprietary in accordance with the policies established by FRA-ANP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by FRA-ANP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in the Document be withheld from public disclosure.

6. The following criteria are customarily applied by FRA-ANP to determine whether information should be classified as proprietary:

(a) The information reveals details of FRA-ANP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for FRA-ANP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for FRA-ANP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by FRA-ANP, would be helpful to competitors to FRA-ANP, and would likely cause substantial harm to the competitive position of FRA-ANP.

7. In accordance with FRA-ANP's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside FRA-ANP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. FRA-ANP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 12th

day of August, 2002.

Ella F. Carr-Payne

Ella F. Carr-Payne
NOTARY PUBLIC, STATE OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/05

ELLA F. CARR-PAYNE
Notary Public
Commonwealth of Virginia
My Commission Expa. Aug. 31, 2005