Florida Power
A **Progress Energy** Company

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

Ref: 10 CFR 50.90

September 30, 2002
3F0902-07

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. FPC to NRC letter, dated June 5, 2002, Crystal River Unit 3 – License Amendment Request #270, Revision 0, "Power Uprate to 2568 MWt"
2. NRC to FPC letter, dated September 18, 2002, Crystal River Unit 3 - Request for Additional Information Re: Proposed License Amendment on Power Uprate to 2568 MWt (TAC No. MB5289)

Dear Sir:

During discussions with the NRC staff on September 3, through several electronic mail transmittals, and by letter dated September 18, 2002 (Reference 2), the NRC staff has requested additional information regarding Florida Power Corporation's (FPC) proposed License Amendment Request #270, "Power Uprate to 2568 MWt." This letter provides the response to these requests.

Attachment A provides the questions and responses. Attachment C provides a Framatome ANP document that was requested for review, and Attachment C contains information that Framatome ANP considers to be proprietary. Framatome ANP requests that the proprietary information in this response (Attachment C) 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 B.

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.

Sincerely,

Dale E Young

Dale E. Young

DEY/pei

Attachments:

A. Response to Request for Additional Information
B. Framatome ANP Affidavit of Proprietary Information
C. FRA-ANP 51-5015662-01, "FIV Development, Qualification and Clarification for TMI"

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

A001

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 30th day of September, 2002, by Dale E. Young.

Janet Schroeder

Signature of Notary Public
State of Florida

JANET SCHROEDER
MY COMMISSION # DD 128063
EXPIRES: June 20, 2006
Bonded Thru Notary Public Underwriters

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

Personally Known ✓ -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 (email dated August 5, 2002):

1. **With respect to Section 4.7.14.5, OTSG Flow-Induced Vibration, of the licensee's submittal, dated June 5, 2002, explain how the minimum margin of about 57 percent against excessive turbulence-induced stress in the stabilized tube was determined. What is the wear rate due to tube-to-TSP interaction under this condition. What are the minimum margins against excessive turbulence-induced stress and wear rate in the other types of OTSG tubes (e.g., virgin tube and other types of repaired OTSG tubes)?**

FPC Response

As identified in Section 3.3 of Reference 1, the maximum stress for any tube configuration (i.e., a virgin tube, a tube stabilized with various designs and hypothetical tube sever locations) is 2187 psi,rms (pounds per square inch, root mean square). The fatigue endurance limit for flow induced vibration (FIV) stress is 3450 psi,rms at 10^{11} cycles. The 57% percent margin against excessive turbulence-induced stress was determined from the relation relative to this margin (3450/2187 - 1) = 57%. The fatigue usage factor associated with this stress would be approximately 0.09 for a tube flaw with a SCF=1.5, at a crossing frequency of 28.9 Hertz (Hz) for 40 effective full power years (EFPY).

The maximum turbulent induced stress of 2187 psi,rms for the cable stabilizer design envelopes the maximum stress for the other stabilizer designs and also the virgin tube Therefore, the evaluation of other stabilizer designs and the virgin tube is not explicitly performed in Reference 1, but is considered to be enveloped by the evaluation for the cable stabilizer. If calculated, the margin against excessive turbulence-induced stress for a virgin tube would be equal to 453% (3450/624 – 1) (Reference 1, Table 2).

As stated in Section 3.4 of Reference 1, the wear determined for the worst located virgin tube (tube location 75-1) is approximately 5.4 mils of its wall thickness after five years of effective full power operation or a wear rate of 1.08 mils per year. This wear rate was determined for a secondary mass flow rate of 5.4E06 pounds mass per hour (lbm/hr) which is the correct flow rate for Crystal River Unit 3 (CR-3) before the 1% increase in secondary side flow rate to account for 20% plugging.

The wear rate will increase slightly for the 1% increase in flow rate associated with the 20% plugging. For small differences in the mean cross flow gap velocities, the wear rate varies approximately by the fourth power of the cross flow gap velocity. The wear rate associated with a secondary side flow rate of 5.454E06 lbm/hr can be estimated as $(1.08)(1.01)^4 = 1.12$ mils/year for the worst located virgin tube.

The wear in a tube at tube support locations for the stabilized tube condition was not explicitly determined in Reference 1. The potential for increased wear or a larger wear rate resulting from the installation of a stabilizer in the tube is not a critical issue as long as the installed stabilizer spans the tube support locations at which the wear rate has increased thereby stabilizing or capturing the potentially severed tube.

However, the wear rate at a tube support location can be relatively evaluated between a stabilized tube and a virgin tube by the product of the reaction load at the support locations and the mid-span tube displacements. The wear at the tube support locations is a function of the turbulent flow induced reaction load at the tube support location and the cumulative sliding distance of the tube at the support location. Since the axial movement of the tube at the support location is proportional to the mid-span displacement of the tube, it is appropriate that the wear rate can be determined through the product of the reaction load and the mid-span displacements. The reaction loads at the tube support locations and mid-span displacements for the stabilized tube and a virgin tube are tabulated below in Table 1. The ratio of the relative wear rate index is also reported.

For the same stabilized tube configuration in which the maximum stress was reported above (a cable stabilizer installed in top span with tube sever at secondary face of tubesheet), the change in wear rate associated with the tube-to-TSP interaction as a result of installing the cable stabilizer would increase approximately 118% at TSP 15 (Table 1). This value is determined for a power rating of 2568 MWt and does not include the approximately 1% increase of secondary side flow associated with 20% of the tube bundle being plugged. The wear rate of the tube at tube support plate 15 with a cable stabilizer installed in consideration of the 1% increase in secondary side flow would be (2.18)(1.12) = 2.44 mils/year if the stabilizer was installed in the worst located tube location within the OTSG tube bundle.

Table 1: Comparison of Reaction Loads at Tube Support Locations [Reference 2, Section 4.3]

Support Location	Reaction Load for Virgin Tube (lbs,rms)	Mid-Span Displacement (inch)	Reaction Load for Cable Stabilizer (lbs,rms)	Mid-Span Displacement (inch)	Relative Wear Rate Index Ratio
LTS	1.01	1.3144E-02	1.07	1.4173E-02	1.14
TSP1	0.82	9.7645E-03	0.71	9.9351E-03	0.88
TSP15	0.97	1.2747E-02	1.96	1.3755E-02	2.18
UTS	1.58	2.1307E-02	2.13	2.9940E-02	1.89

On page 1 of Reference 1, the wear rate on a virgin tube at the top tube support plate is stated to increase about 43% based on a increase of 8% in the secondary side flow rate. This wear rate was determined based on the 1% increase in the OTSG secondary flow rate for 20% plugging and the additional 8% of margin. As stated above, the wear rate is proportional to the fourth power of the cross flow gap velocity. The change in wear rate was determined as follows:

$(1.09)^4 - 1 = 41\%$.

This value differs slightly from the value documented in Reference 1 (43%). However, as stated in Section 6 of Reference 1, the increase in wear rate of 43% is a conservative estimate of this phenomenon and bounds the 41% calculation above.

NRC Request (email dated August 5, 2002):

2. **It is stated, in Section 3.4 of Reference 13, that the stabilized tubes have been plugged and removed from service. Therefore, wear is not a concern for the stabilized tubes. Explain what is the relevance of having the minimum margin against excessive turbulence-induced stress in the stabilized tube as stated in Section 4.7.14.5 of the June 5, 2002, submittal.**

FPC Response:

Random turbulent-induced vibration analysis is performed for the stabilized tube with various hypothetical tube severs assumed in the structural models to evaluate the potential for the stabilized tube to impact an adjacent in-service tube. Additionally, the stress in the out-of-service tube is evaluated to ensure that the response of the out-of-service tube with the addition of the stabilizer does not create a high cycle stress/fatigue condition such that a circumferential tube sever is developed by fatigue at a tube location that is not captured by the stabilizer.

NRC Request (per phone call):

The following two statements are taken from the Safety Evaluation for License Amendment 204. The NRC staff requested that FPC address these statements as part of the power uprate RAI questions concerning flow induced vibration.

"Although Attachments B and D [CR-3 and TMI power uprate FIV evaluations] to the licensee's June 5, 2002, letter are not the subject of this review, the staff noticed that some parameters used in the FIV analysis of the OTSG tubes may not be appropriate and may require detailed justification in future reviews. In particular, if the high damping ratios (> 3 percent) of the steam generator tubes continue to be used in any future analysis, the staff will require detailed justification and review of the test data."

"If increased feedwater flow exceeding the licensed limit is needed for any reason with up to 20-percent tube plugging, the licensee [Florida Power Corporation] should provide the FIV with sufficient details for staff review to demonstrate the functional integrity of the steam generator tubes for up to 20-percent tube plugging under symmetric and worst-case asymmetric plugging distributions."

FPC Response:

The FIV methodologies used by Framatome ANP are outlined in Reference 4 (included in Attachment C) and have been employed for power uprate FIV analysis at CR-3. Currently, Framatome ANP performs fluid-elastic instability analysis of the OTSG tubes and stabilizers employing a Connors' constant of 3.3 and 3% damping value for a loosely supported tube. The

3% damping value is used only for the fluid-elastic instability analysis where the tube or stabilized tube experiences large amplitude vibrations. The nominal tube-to-tube support plate diametrical clearance is 18 mils. It is worth while to mention that Framatome ANP does not account for squeeze film damping in our fluid-elastic instability analysis. This is a conservative approach. For random turbulence-induced vibrations where the displacements are smaller, 2% viscous damping is used.

In the initial developmental stages of the OTSG design, which occurred during the late 1960's, Babcock & Wilcox performed numerous tests to assess the heat transfer characteristics and structural integrity of the OTSG shell and tube bundle. The mockup of the OTSG design was similar in length and other pertinent design considerations to that which was constructed for commercial operation with the exception of the number of tubes. One of the objectives of this program was to study, experimentally, the effect of various operating and physical parameters on the stability characteristics of the OTSG tube. Vibration "pluck" testing of tubes on the actual fabricated commercial OTSG was performed in order to demonstrate the production unit's vibratory response and the damping ratio. In the single tube mockup test, the average percent of critical damping was 5% to 6% in air. In the production unit test, the average percentage of damping was about 6% in air (Reference 4, Section 2.0).

The flow test of the OTSG tube bundle performed by Chalk River Laboratories in June of 1994 and the analysis of the test data performed in Reference 3, suggest a Connors' constant of 2.4 with a damping of 5%. The FIV results documented in Reference 3 employ a Connors' constant of 2.4 and 5% damping. These two parameters (Connors' constant and damping) are directly proportional the Fluid-elastic Stability Margin (FSM) through the relation;

FSM ~ $\beta(\zeta)^{1/2}$, where: β is defined as the Connors' constant and
ζ is defined as the damping.

The fluid-elastic stability margins computed with the two combinations of Connors' constant and damping are performed below.

FSM ~ $2.4(0.05)^{1/2} = 0.536$
FSM ~ $3.3(0.03)^{1/2} = 0.571$
% Difference = (0.571 – 0.536)/0.536 = 6.5%

Therefore, the FSM computed with a Connors' constant of 3.3 and 3% damping would show an additional 6.5% margin relative to the FSM computed with a Connors' constant of 2.4 and 5% damping. Since a Connors' constant of 2.4 is considered a lower bound value for fluid-elastic instability analysis and the 5% damping is considered a upper bound value for the damping, Framatome ANP believes the combination of a Connors' constant of 3.3 and tube-to-tube support plate interaction damping of 3% to be appropriate. However, the FSMs results determined with Connors' constant of 2.4 and 5% damping provide about the same results and therefore, these results are considered by Framatome ANP and FPC to also be accurate.

Table 1 of Reference 1 is a summary of the FSMs computed from several FIV analyses of various stabilizers. The values reported were computed with both combinations of Connors' constants and damping. The results are considered to be equivalent based on the relatively small percent difference in the product of Connors' constant and damping shown above.

NRC Request (email dated August 13, 2002):

1. Accidents and Transients Analyses

a. Discuss your licensing basis for accidents and transients analyses. Clarify that all accidents and transients original analyses of record were analyzed with a 2 % uncertainty.

FPC Response:

As summarized in Table 3 of Attachment A of LAR #270, the majority of the events have been analyzed at a power level of 2619 MWt (102% of 2568) or greater. For those events not analyzed at 2619 MWt or greater, operation at a maximum error adjusted power level of 2619 MWt is supported as discussed in the response to question 1c.

NRC Request (email dated August 13, 2002):

b. In attachment 1, page 2, it is stated, "All of the revised analyses were performed considering a maximum power output of 2568 MWt or higher .... All of these analyses were approved by the NRC or were performed using methods or processes that were approved by the NRC." Please provide a complete documentation of these analyses and the NRC approval letters. Similarly, expand Table 3 (page 26 of Attachment A) matrix to include NRC's previous approvals references.

FPC Response:

The following analyses were performed after the initial licensing of CR-3. For all other events, the latest analyses are the original Final Safety Analysis Report (FSAR) cases and were approved during initial licensing of CR-3 (SER in letter from Roger Boyd (NRC) to J. T. Rogers (FPC) dated December 3, 1976). The revised analyses were performed using NRC approved computer codes and the methodology as defined in the FSAR or approved topical reports as listed. Framatome-ANP maintains NRC-approved codes and implementation methodology, applies them to the calculations, and confirms that the analyses meet the limitations and restrictions of the tools and methodology topical reports.

Startup Accident

BAW-10164P-A, "RELAP5MOD2/B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," October 1992. Approved for use for B&W plants in letter from Gary M. Holahan (NRC) to J. H. Taylor (B&W), dated March 14, 1995 and SER for BAW-10193P-A.

BAW-10193P-A, "RELAP/MOD2-B&W for Safety Analysis of B&W-Designed PWRs." Approved in letter from S. Bailey (USNRC) to J. J. Kelly (Framatome Technologies), "Safety Evaluation Report for RELAP5MOD2/B&W for Safety Analysis of B&W-Designed PWRs." October 15, 1999.

Moderator Dilution at Power

BAW-10098, "CADDS Computer Application to Direct Digital Simulation of Transients in PWRs With or Without SCRAM," January 1975, following FSAR methodology. CADDS was one of the original codes used to analyze this event for initial licensing. BAW-10098 was approved in a letter from Cecil O. Thomas (NRC) to James H. Taylor (B&W), dated August 21, 1983. A code-to-code comparison of CADDS to RELAP5/MOD2-B&W can be found in the SER for BAW-10193P-A. CADDS is referred to as being previously approved by the NRC in the SER for BAW-10193P-A.

Locked Pump Rotor

BAW-10164P-A (RELAP5/MOD2-B&W) and follows BAW-10193P-A.

Station Blackout

BAW-10164P-A (RELAP5/MOD2-B&W), NUMARC 87-00 methodology, SER and Supplemental SERs in letters from Harley Silver (NRC) to Percy Beard (FPC), dated August 23, 1990, May 6, 1991 and May 29, 1992.

Main Steam Line Break

BAW-10128, "TRAP2 -FORTRAN Program for Digital Simulations of the Transient Behavior of the Once-Through Steam Generator and Associated Reactor Coolant System," Rev. 0, August 1976, following FSAR methodology. BAW-10128 was approved in a letter from Cecil O. Thomas (NRC) to James H. Taylor (B&W), dated August 3, 1983. TRAP2 was used for predicting transient system response. TRAP2 is referred to as being NRC approved in the SER for BAW-10193P-A.

BAW-10095A Rev. 1, "B&W's Revisions to CONTEMPT – Computer Program for Predicting Response to a Loss of Coolant Accident," April 1978. CONTEMPT was used for predicting containment response to the mass/energy release. The CONTEMPT code was approved in a letter from Steven Varga (NRC) to James H. Taylor (B&W) dated February 13, 1978. CONTEMPT is also listed as an approved methodology in the SER for BAW-10192P-A.

Large Break and Small Break LOCAs

BAW-10164-A (RELAP5MOD2/B&W) was used for predicting transient system response.

BAW-10192P-A, "BWNT LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," June 1998. BAW-10192P-A contains the overall method for demonstrating ECCS performance. BAW-10192P-A was approved in a letter from James E. Lyons (NRC) to J. H. Taylor (Framatome Technologies), dated February 18, 1997.

The adoption of BAW-10192P-A as the analysis of record for CR-3 was submitted in FPC letter, 3F0199-02, dated January 7, 1999, "Change in Analysis of Record for Small Break Loss of Coolant Accident and 10 CFR 50.46 Notification," and FPC letter, 3F1199-01, dated November 10, 1999, "Notification of Change in Peak Clad Temperature for Small Break Loss of Coolant Accident in Accordance with 10 CFR 50.46(a)(3) and Change in the Analysis of Record for Large Break Loss of Coolant Accident."

Letdown Line Break

RELAP5/MOD2-B&W following BAW-10192P-A methodology, approved for use at CR-3 in License Amendment #173, dated April 13, 1999. This analysis was performed for dose calculations only, not for demonstrating ECCS performance.

Loss of Main Feedwater

RELAP5/MOD2-B&W and follows BAW-10193P-A.

Feedwater Line Break

RELAP5/MOD2-B&W and follows BAW-10193P-A.

Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC)

BAW-10098, Revision 1.

Radiological Analyses

Radiological analyses were originally done using BURPE (BAW-1512, Rev. K), LOCACL (NPGD-TM-438, Rev.1) and RELOAD (NPGD-TM-537, Rev.C). License Amendment #199, dated September 17, 2001, approved the use of the Alternative Source Term per 10 CFR 50.67. The SER for this amendment approved the use of the Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," methodology and the use of RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation. The revised methodology was used for all radiological analysis which were performed assuming a power level 2619 MWt.

The above information is summarized in the table below using the same format as Table 3 in the original LAR #270 submittal:

TABLE 2: Accident Analysis Methodology			
FSAR Section	**Accident**	**Topical Methodology (Code)**	**SER Date**
14.1.2.1	Uncompensated Operating Reactivity Changes	NPGD-TM-52 (Star7)	12/03/76
14.1.2.2	Startup Accident	BAW-10068 (KAPPB) BAW-10193P-A (RELAP5)	12/03/76 10/15/99
14.1.2.3	Rod Withdrawal at Power	BAW-10068 (KAPPB)	12/03/76
14.1.2.4	Moderator Dilution From Full Power	BAW-10098 (CADDS) BAW-10193P-A (RELAP5)	08/21/83 10/15/99
14.1.2.4	Moderator Dilution Accident During Refueling	NPGD-TM-52 (Star7) BAW-10098 (CADDS)	12/03/76 08/21/83
14.1.2.5	Cold Water Accident	BAW-10068 (KAPPB)	12/03/76
14.1.2.6	Single Pump Coastdown	Bounded by Locked Rotor	N/A
14.1.2.6	Locked Rotor	BAW-10098 (CADDS) BAW-10193P-A (RELAP5)	08/21/83 10/15/99
14.1.2.6	Four-Pump Coastdown	BAW-10098 (CADDS)	08/21/83
14.1.2.7	Stuck-Out, Stuck-In, or Dropped Control Rod Accident	BAW-10068 (KAPPB)	12/03/76
14.1.2.8	Load Rejection/Turbine Trip	BAW-10068 (KAPPB) BAW–10070 (POWERTRAIN)	12/03/76 12/03/76
14.1.2.9	Station Blackout	BAW-10164P-A (RELAP5) NUMARC 87-00	03/14/95 05/29/92
14.1.2.9	Loss of AC Power	BAW-10068 (KAPPB) BAW–10070 (POWERTRAIN)	12/03/76 12/03/76
14.2.2.1	Steam Line Failure Accident	BAW-10164P-A (RELAP5) BAW-10128 (TRAP2) BAW-10095A, Rev.1 (CONTEMPT) BAW-10193P-A (RELAP5)	03/14/95 08/03/83 02/13/78 10/15/99
14.2.2.2	Steam Generator Tube Rupture	Alternative Source Term	09/17/01
14.2.2.3	Fuel Handling Accident	Alternative Source Term	09/17/01
14.2.2.4	Hot Zero Power Rod Ejection	BAW-10068 (KAPPB)	12/03/76
14.2.2.4	Full Power Rod Ejection	BAW-10068 (KAPPB)	12/03/76
14.2.2.5	Loss-of-Coolant Accidents	BAW-10164P-A (RELAP5)	03/14/95
14.2.2.6	Makeup System Letdown Line Failure	BAW-10164P-A (RELAP5)	03/14/95
14.2.2.7	Maximum Hypothetical Accident	Bounded by LOCA	N/A
14.2.2.8	Waste Gas Tank Rupture Accident	Alternative Source Term	09/17/01
14.2.2.9	Loss of Main Feedwater	BAW-10164P-A (RELAP5) BAW-10193P-A (RELAP5)	03/14/95 10/15/99
14.2.2.9	Total Loss of Feedwater Accident	BAW-10092P-A (CRAFT2)	02/18/97
14.2.2.9	Feedwater Line Break	BAW-10164P-A (RELAP5)	03/14/95
N/A	ATWS (LOFW)	BAW-10098 (CADDS)	08/21/83
N/A	AMSAC	BAW-10098 (CADDS)	08/21/83

NRC Request (email dated August 13, 2002):

c. In Attachment 1, page 26 Table 3, "Accident Analysis Summary" has listed all accidents and transients analyses as record of analyses. Most of the analyses were performed at 2619 MWt and some analyses were performed at 2568 MWt. Justify those analyses which were performed at 2568 MWt from DNBR and over pressure protection point of view that they meet uprated conditions. For these transients, justify how the uncertainty is considered for uprated condition.

FPC Response:

Table 3 identified six event analyses performed at 2568 MWt. These are: Rod Withdrawal at Power (FSAR Section 14.1.2.3), Stuck-Out, Stuck-In, or Dropped Control Rod Accident (FSAR Section 14.1.2.7), Loss of AC Power (FSAR Section 14.1.2.9), Steam Line Failure Accident (FSAR Section 14.2.2.1), Steam Generator Tube Rupture (FSAR Section 14.2.2.2), and Full Power Rod Ejection (FSAR Section 14.2.2.4). The first three events are classified as ANS Condition II while the last three are classified as ANS Condition IV. Condition II events are events of moderate frequency and shall not result in fuel rod failures or reactor coolant system or secondary system overpressure. Condition IV events are limiting faults which are not expected to take place, but are postulated because their consequence would include the potential for the release of significant amounts of radioactivity. Condition IV event analysis may involve fuel cladding failures and reactor coolant system (RCS) pressure boundary breaches. These failures are acceptable if the resultant dose consequences are acceptable. Justification for performing the six identified events at 2568 MWt versus 2619 MWt is provided below:

1. Rod Withdrawal at Power (FSAR Section 14.1.2.3 – ANS Condition II)

The information provided below supplements the discussion provided in Section 4.11.3 of Attachment A to LAR#270.

The initial core power level for the Rod Withdrawal at Power accident analyses is 2568 MWt. The high flux trip setpoint used in the analysis was 112 percent of 2568 MWt. The 112 percent high flux trip setpoint includes a 2 percent heat balance error. Including the heat balance error in the high flux trip setpoint, versus starting from an initial power level of 102 percent of 2568 MWt, results in the same core thermal power at the time of reactor trip. In addition, starting with an initial power level of 2568 MWt and including the heat balance error in the high flux trip setpoint maximizes the increase in core power from the initial power level to the power level at the time of trip. Maximizing the increase in core power between event initiation and the time of reactor trip results in a larger energy mismatch between core heat generation and secondary heat removal. This produces a higher peak RCS pressure with all other boundary conditions remaining the same. Therefore, the Rod Withdrawal at Power accident DNBR and over pressure protection analyses performed at 2568 MWt support CR-3 operation at a maximum error-adjusted power level of 2619 MWt.

2. Stuck-Out, Stuck-In, or Dropped Control Rod Accident (FSAR Section 14.1.2.7 – ANS Condition II)

The information provided below supplements the discussion provided in Section 4.11.8 of Attachment A to LAR#270.

The Stuck-Out and Stuck-In events are control rod misalignments. They are evaluated to assure sufficient reactivity margins are maintained. They are not dependent on the initial core power level.

The dropped control rod accident is evaluated to demonstrate the minimum DNBR and peak RCS acceptance criteria are met.

The original dropped rod analysis was performed at 2568 MWt. The system response to a dropped rod is not affected by the initial power level since the heat generated in the core is being removed by the steam generators. An increase in the initial analyzed power level from 2568 MWt to 2619 MWt will not affect the evolution of primary coolant temperature, coolant flow rate, or RCS pressure. The evolution of these parameters is determined by the worth of the dropped control rod and core reactivity feedback.

The normalized power response along with coolant temperature, coolant flow rate, and RCS pressure are provided as input to the cycle-specific core DNBR analysis. The cycle-specific analyses use NRC approved statistical methods and apply the appropriate conservatism for core power, i.e., accounts for a 2 percent full power heat balance error. Therefore, the Stuck-Out, Stuck-In, or Dropped Rod accident DNBR and over pressure protection analyses performed at 2568 MWt support CR-3 operation at a maximum error-adjusted power level of 2619 MWt.

3. Loss of AC Power (FSAR Section 14.1.2.9 – ANS Condition II)

The information provided below supplements the discussion provided in Section 4.11.10 of Attachment A to LAR#270.

The concerns related to a Loss of AC Power event are peak RCS pressure, minimum DNBR, and decay heat removal capability under natural circulation. The Loss of AC Power event is not the limiting transient with respect to peak RCS pressure or minimum DNBR. The consequences with respect to peak RCS pressure are bounded by the Loss of Main Feedwater event. The Loss of Main Feedwater event results in a more severe reduction in primary-to-secondary heat transfer than the Loss of AC Power event. The Loss of Main Feedwater event was analyzed at 102 percent of 2568 MWt and the peak RCS pressure was below the acceptance criterion limit. The Single-Pump Coastdown event results in a more severe transient for minimum DNBR than the Loss of AC Power event. The Single-Pump Coastdown event is evaluated every reload to ensure that the current analysis bounds fuel cycle operation. The evaluation includes DNB analysis at 102 percent of 2568 MWt. As a result, the Loss of AC Power event consequences with respect to peak RCS pressure and minimum DNBR are bounded by the Loss of Main Feedwater event and the Single-Pump Coastdown event which are analyzed at 102 percent of 2568 MWt.

The remaining issue to address is the ability to remove decay heat with natural circulation. On loss of forced reactor coolant flow, emergency feedwater (EFW) will be started and the liquid level in the steam generators will be raised to the natural circulation control setpoint. The transition to natural circulation will be governed by the ability of EFW to absorb decay heat. The Loss of Main Feedwater event establishes the minimum EFW flow requirement. Since the Loss of Main Feedwater event was analyzed with an initial power of 102 percent of 2568 MWt, adequate EFW flow will also be available to remove decay heat for the Loss of AC Power transient. Therefore, the current Loss of AC Power event analysis supports CR-3 operation at a maximum error-adjusted power level of 2619 MWt.

4. Steam Line Failure Accident (FSAR Section 14.2.2.1 – ANS Condition IV)

The information provided below supplements the discussion provided in Section 4.11.11 of Attachment A to LAR#270 and FSAR 14.2.2.1.5.a and b.

The Steam Line Failure accident was analyzed at 2568 MWt. This power level is consistent with the guidance given in BAW-10193P-A and is conservative for calculating core response to the steam line break. It is conservative, relative to 102 percent of 2568 MWt, as it tends to minimize core decay heat and maximize overcooling of the RCS. The initial mass inventory of each steam generator is inflated to correspond to 102 percent of 2568 MWt to also maximize the overcooling of the RCS.

A Steam Line Failure accident analysis for determining mass and energy releases to containment was performed at 102 percent of 2568 MWt. These mass and energy releases were used to calculate the containment pressure response which was acceptable.

Therefore, the Steam Line Failure analyses for the core power response and mass and energy releases for containment pressure response support CR-3 operation at a maximum error-adjusted power level of 2619 MWt.

5. Steam Generator Tube Rupture (FSAR Section 14.2.2.2 – ANS Condition IV)

The information provided below supplements the discussion provided in Section 4.11.12 of Attachment A to LAR#270.

The steam generator tube rupture (SGTR) analysis of record is a hand calculation of primary to secondary leakage based on a constant leak rate of 435 gpm. The leak flow is the combined critical flow from each end of the ruptured tube. This leak rate was assumed constant until the plant was cooled down to the temperature at which the decay heat removal system can be placed in service. This leak flow rate is conservative because it does not credit the decrease in the leakage rate with RCS depressurization or the secondary side pressurization following reactor trip and turbine trip nor the hydraulic losses through the tube. The SGTR calculation is independent of power level based on the analytical methods used. The core and coolant source terms for the offsite dose release evaluation are based on cycle-specific calculations and account for the power uprate. Therefore, the SGTR analysis supports CR-3 operation at a maximum error-adjusted power level of 2619 MWt.

6. Full Power Rod Ejection (FSAR Section 14.2.2.4 – ANS Condition IV)

The information provided below supplements the discussion provided in Section 4.11.14.2 of Attachment A to LAR#270.

The Rod Ejection from Full Power is analyzed at 2568 MWt. Assuming actual power is at the maximum of the 2 percent power uncertainty (2619 MWt) will increase the initial fuel enthalpy slightly, thereby reducing the existing margin to the 280 calorie/gram limit. The analysis shows that there is approximately 80 calorie/gram margin for rod worths of 0.7 %Δk/k, which is sufficient to bound the 2 percent power uncertainty. Current core designs limit the maximum worth of a control rod to less than 0.65 %Δk/k with a 15% margin. Therefore, the Rod Ejection from Full Power analysis supports CR-3 operation at a maximum error-adjusted power level of 2619 MWt.

The dose assessment for the Rod Ejection from Full Power is based on core inventories of radioactive nuclides. The radioactive nuclide inventories are based on plant power level and fuel irradiation history and the expected number of DNBR fuel failures based on a maximum ejected rod worth of 0.65 %Δk/k. The power level utilized in determining radioactive nuclide inventories was 102 percent of 2568 MWt. Therefore, the dose consequences of the Rod Ejection from Full Power supports CR-3 operation at a maximum error-adjusted power level of 2619 MWt.

NRC Request (email dated August 13, 2002):

d. On page one of the attachment A, the second from the end of the first paragraph, alludes to the fact that the actual power trip will occur at a higher absolute power. Did CR-3 investigate the impact of this trip at the higher power level ? What is the impact of the trip setting at the higher power level on other pertinent systems ?

FPC Response:

The referenced statement was provided in the introduction, as an example, to illustrate that trip set points expressed in percent of rated power are unchanged, but will occur at a higher absolute power due to the increase in maximum power level from 2544 MWt to 2568 MWt. The impact of this and other trips occurring at a higher absolute power has been considered in the uprate evaluation documented in Section 4.0, "Technical Analysis," of Attachment A of LAR #270. The uprate evaluation addressed the following categories: NSSS performance parameters, design transients, systems, components, accidents, and nuclear fuel as well as interfaces between the NSSS and balance-of-plant (BOP) systems. The results of all of the analyses and evaluations performed demonstrate that all acceptance criteria continue to be met and that the plant requires no design changes other than calibrations and setpoint changes to safely operate at the uprated conditions.

NRC Request (email dated August 13, 2002):

2. Section 14.2 Core Thermal-Hydraulic Design

a. Provide the fuel type to be used for Cycle 13 operation and identify the difference if not an equilibrium core.

FPC Response:

The Cycle 13 core consists entirely of the Mark-B10 fuel assembly design. Since the Cycle 13 core consists entirely of the Mark-B10 fuel assembly design, there is no mixed core condition and a mixed core penalty was not applied to the DNBR evaluations. The detailed evaluation of the CR-3 Cycle 13 core was provided in BAW-2391, "Crystal River Unit 3 Cycle 13 Reload Report," Revision 1, September 2001. BAW-2391 documents that the Cycle 13 analyses were performed in accordance with the approved methodology described in topical BAW-10179P-A, Rev. 3.

NRC Request (email dated August 13, 2002):

b. Provide detailed justification that the BWC DNB (or CHF) correlation is still valid for Cycle 13 operation including all available data bases to support the approved BWC CHF correlation if fuel type other than approved fuel type specified in the approved topical report. Also, describe the mixed core CHF calculation method for Cycle 13 operation if mixed core condition is applicable to Cycle 13 operation.

FPC Response:

The BWC CHF correlation was developed and approved for the Mark-B series zircaloy intermediate spacer grid design. Over the last 20 years, the grid design has incurred only insignificant changes that do not influence the application of the BWC CHF correlation. As a result, no additional CHF data has been obtained for the grid. The data base for the Mark-B series application of BWC is contained in BAW-10143P-A, "BWC Correlation of Critical Heat Flux," April 1985.

The approval for the application of the BWC CHF correlation with the LYNXT thermal-hydraulic code is documented in the LYNXT Safety Evaluation Report in BAW-10156-A, Revision 1, "LYNXT Core Transient Thermal-Hydraulic Program," August 1993. Since there have been no significant design changes in the grid since the CHF tests, and the calculational tool, LYNXT, has been approved for providing BWC CHF predictions, the BWC correlation remains valid for the Mark-B10 fuel design contained in the CR-3 Cycle 13 core. The Reload Report (BAW-2391, Revision 1, Crystal River Unit 3 – Cycle 13 Reload Report) documents the applicability of the BWC CHF correlation in Section 6, "Thermal-Hydraulic Design."

NRC Request (from September 3, 2002 Meeting):

1. In Section 4.9.7, the licensee stated that the small load increase to the 500 KV switchyard is acceptable and will have no significant impact on grid stability. What is the basis for this conclusion?

FPC Response:

FPC performed a grid stability study in June 2001. This study used the CR-3 generator output of 890 MWe. The power uprate will enable CR-3 to operate at approximately 903 MWe. FPC Transmission Planning has reviewed the uprate and has determined that the small increase in generation will have no appreciable effect on the grid stability study. No revision was required to the grid stability study due to the small size of the uprate.

NRC Request (from September 3, 2002 Meeting):

2. **What impact will the power uprate have on the generator power factor? How will the uprate impact reactive load capability?**

FPC Response:

Figure 1 shows the CR-3 generator Reactive Capability Curve. CR-3 will continue to operate within the limits of this curve after the power uprate. The CR-3 generator is capable of 989.4 MVA with a power factor of 1.0 (zero megavar output). With the current thermal power output, CR-3 can operate at approximately 895 MWe. The power uprate will enable CR-3 to operate at approximately 903 MWe. As MWe output is increased, the capability to carry reactive load decreases. CR-3 typically supplies between 50 to 250 megavars to the FPC grid. This small power uprate will not affect the ability of CR-3 to provide this reactive load.

CR-3's ability to carry reactive load is limited by the step up transformers and not the main generator. The step up transformers have a nominal rating of 950 MVA, lower that the main generator rating of 989.4 MVA. Operating at 903 MWe and a power factor of 0.95 would limit reactive load to approximately 295 megavars. This reactive load is greater than that generally required. If greater megavar output is needed, the MWe output would have to be decreased.

The power output values described assume a hydrogen cooling pressure of 60 psig. Lower hydrogen cooling pressure decreases generator capability. Note that the CR-3 generator is cooled by hydrogen only. The hydrogen is cooled by secondary cooling water. Section 4.9.1 of the original submittal incorrectly states that the hydrogen cooling is supplemented with cooling water for the stator.

NRC Request (from September 3, 2002 Meeting):

3. **The increased power level will increase some cooling water temperatures, what impact will this have on the cooling capability and rating of the emergency diesel generators?**

FPC Response:

The CR-3 emergency diesel generators (EDGs) have a self-contained jacket cooling system with radiators which are air-cooled by an engine driven cooling fan. Therefore, the cooling capability and electrical rating of the EDGs are not affected by the power uprate. The EDG cooling system is discussed in FSAR Section 8.2.3.1.3.c.

NRC Request (letter dated September 18, 2002):

1) Section 4.7.2.4 of Attachment A to the amendment request indicates the RT_{PTS} value at end of license (EOL) for the limiting beltline material, upper shell longitudinal welds WF-18 and WF-8 is 206.0°F. The attachment also indicates that neutron embrittlement analyses were performed using a 7-percent increase in neutron fluence and the chemical composition reported in Reference 7, BAW-2325, Revision 1, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," January 1999. This report identifies the EOL RT_{PTS} value for the upper shell longitudinal weld as 215.7°F, which is greater than the value reported in this amendment. Since the neutron fluence was assumed to increase in this evaluation, the RT_{PTS} should have increased.

Explain this discrepancy and provide the following information:

FPC Response:

Information provided in Table 1 of BAW-2325 (Reference 5) contained updated chemistry information while using the current docketed fluence values for each plant. At that time (May 1998), BAW-2049 (Reference 6) was the current fluence analysis on the docket for CR-3. Framatome ANP Document 86-1266133-01 (Reference 7) is now the current fluence report for CR-3 utilizing NRC approved fluence methodology, BAW-2241-P (Reference 11). EOL fluence values in Reference 7 were increased by 7% and used in the RT_{PTS} (Reference Temperature – Pressurized Thermal Shock) calculation prepared for the CR-3 power uprate (Reference 8). The calculated EOL fluences in Reference 7 are less than those reported in References 5 and 6, due to implementation of low leakage cores and improved neutron transport calculations. The uprated fluence values used in Reference 8 are less than those reported in References 5 and 6, even with the 7% increase. Therefore, the RT_{PTS} of the limiting beltline locations at CR-3 is lower than that given in BAW-2325 (Reference 5).

NRC Request (letter dated September 18, 2002):

a) The neutron fluence at the clad-base metal interface for the upper shell longitudinal weld at EOL,

FPC Response:

The neutron fluence (32 Effective Full Power Years, EOL, inside wetted surface) for the CR-3 upper shell longitudinal welds is 7.92E+18 neutrons/cm^2 (Reference 8). Attenuation through the nominal 0.125" clad is minimal, the RT_{PTS} calculation utilized this conservative fluence.

NRC Request (letter dated September 18, 2002):

b) The amounts of copper and nickel for the upper shell longitudinal weld,

The CR-3 upper shell longitudinal welds have 0.19 percent by weight (wt%) copper and 0.57 wt% nickel (References 5 and 8).

NRC Request (letter dated September 18, 2002):

c) The unirradiated reference temperature for the upper shell longitudinal weld,

FPC Response:

The unirradiated RT_{NDT} for the CR-3 upper shell longitudinal welds is –5°F (References 5 and 8).

NRC Request (letter dated September 18, 2002):

d) The margin value utilized in the RT_{PTS} evaluation for the upper shell longitudinal weld.

FPC Response:

The margin value for the CR-3 upper shell longitudinal welds is 68.5 °F (References 5 and 8).

NRC Request (letter dated September 18, 2002):

2) In order to satisfy the reactor vessel Charpy upper-shelf energy requirements of Appendix G, 10 CFR Part 50, Florida Power Corporation has performed equivalent margins analyses for the beltline welds. What is the impact of the power uprate on the equivalent margins analyses? Provide the neutron fluence at the 1/4 thickness at EOL for the beltline welds and describe the impact of this neutron fluence on the equivalent margins analyses.

FPC Response:

CR-3 EOL 1/4 thickness uprated fluences for the limiting circumferential and longitudinal welds identified in Reference 6 are 4.84E+18 neutrons/cm^2 and 4.63E+18 neutrons/cm^2, respectively (Reference 9). The power uprate has a very small impact on these margins and the existing equivalent margins analyses in place continue to bound the uprated fluences listed above (Reference 10).

References:

1. Framatome ANP Document 51-5000475-01, "CR-3 OTSG FIV Margins," dated October 1, 2001 (non-Proprietary - submitted in letter dated June 5, 2002).

2. Framatome ANP Document 32-1225188-00, "OTSG Cable Stabilizer – FIV," dated September 1993.

3. Framatome ANP Document 32-1257514-01, "Flow-Induced Vibration Analysis of TMI OTSG Tube due to Power Up-rate," dated June 23, 1997 (Proprietary - submitted in letter dated June 5, 2002).

4. Framatome ANP Document 51-5015662-01, "FIV Development, Qualification and Clarification for TMI," dated December 7, 2001 (Proprietary – included in Attachment C).

5. Framatome ANP Document BAW-2325, Revision 1, "Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel Integrity," January 1999.

6. Framatome ANP Document BAW-2049, "Analysis of Capsule CR3-F Florida Power Corporation Crystal River Unit-3," September 1988.

7. Framatome ANP Document 86-1266133-01, "CR-3 PT Fluence Analysis Report Cycles 7-10," August 1998 (included in letter dated August 13, 2002 - Proprietary).

8. Framatome ANP Document 32-5013892-02, "CR-3 Power Uprate PTS Evaluation," August 2002.

9. Framatome ANP Document 32-5013936-01, "Adjusted Reference Temperatures for 32 EFPY for CR-3 Power Uprate," August 2002 (included in letter dated August 13, 2002).

10. Framatome ANP Document 32-1245770-01, "Low Upper-Shelf Toughness Fracture Analysis – Levels A, B, C, D," January 1998.

11. Framatome ANP Document BAW-2241-P, "Acceptance for Referencing of Licensing Topical Report Methodologies," dated February 18, 1999.

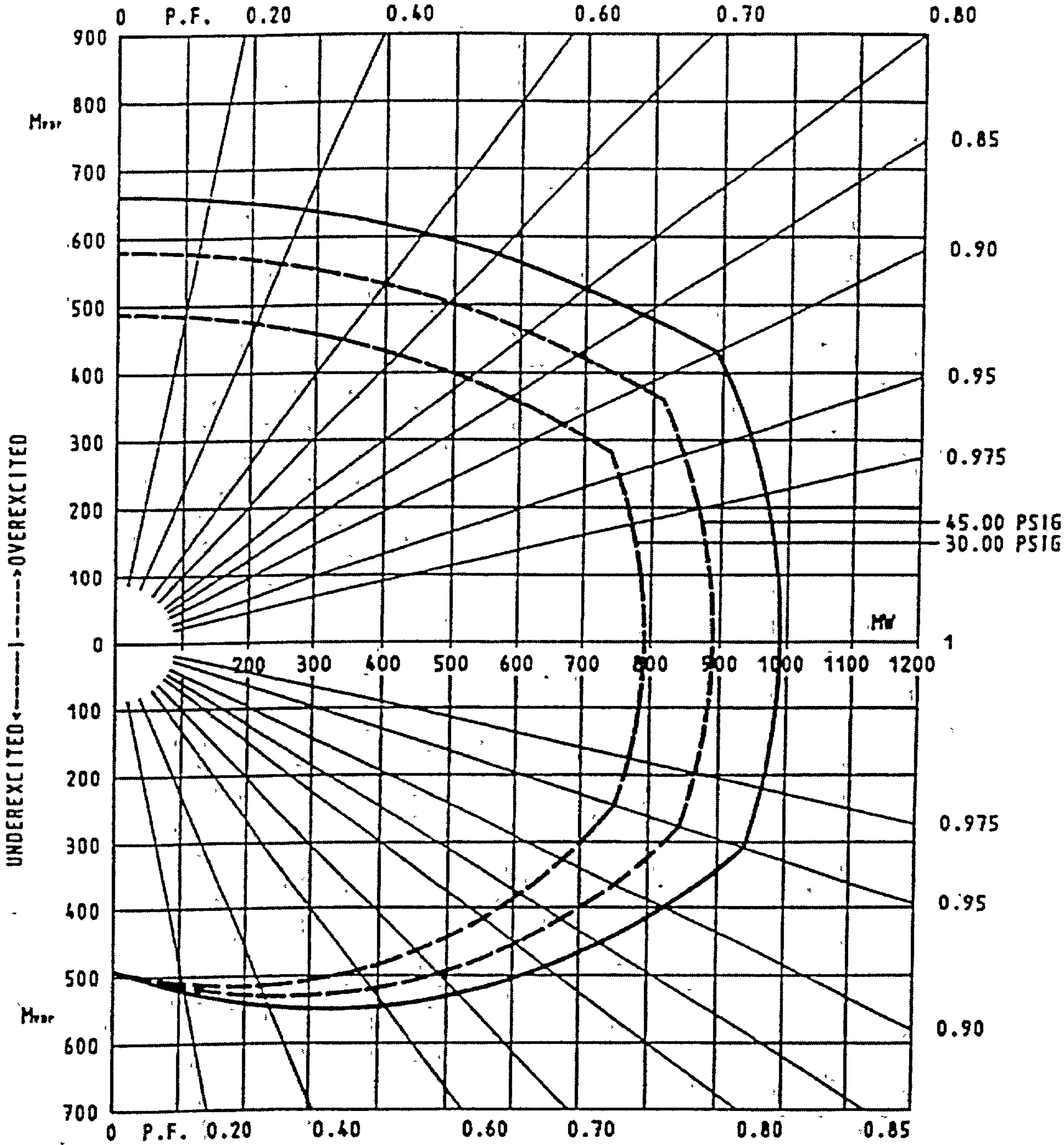

Figure 1
Generator Reactive Capability Curve

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

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.

James E. Malloy

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 Exps. Aug. 31, 2005