Duke Energy®
Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

August 30, 2012
3F0812-06

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

Subject: Crystal River Unit 3 – Response to Second Request for Additional Information to Support NRC Mechanical and Civil Engineering Branch (EMCB) Technical Review of the CR-3 Extended Power Uprate LAR (TAC No. ME6527)

References:
1. FPC to NRC letter dated June 15, 2011, "Crystal River Unit 3 – License Amendment Request #309, Revision 0, Extended Power Uprate" (ADAMS Accession No. ML112070659)
2. Email from S. Lingam (NRC) to D. Westcott (CR-3) dated May 1, 2012, "Crystal River, Unit 3 EPU LAR Draft RAIs from Power Ascension and Testing Plan (EMCB, Formerly Under IQVB)"
3. NRC to FPC letter dated August 1, 2012, "Crystal River Unit 3 Nuclear Generating Plant – Request For Additional Information For Extended Power Uprate License Amendment Request (TAC No. ME6527)" (ADAMS Accession No. ML12125A162)
4. FPC to NRC letter dated October 25, 2011, "Crystal River Unit 3 – Feedwater Line Break Overpressure Protection Analysis to Support NRC Reactor Systems Branch Acceptance Review of the CR-3 Extended Power Uprate LAR and LAR Approval Schedule (TAC No. ME6527)" (ADAMS Accession No. ML11300A226)

Dear Sir:

By letter dated June 15, 2011, Florida Power Corporation (FPC) requested a license amendment to increase the rated thermal power level of Crystal River Unit 3 (CR-3) from 2609 megawatts (MWt) to 3014 MWt (Reference 1). On May 1, 2012, via electronic mail, the NRC provided a draft request for additional information (RAI) needed to support the EMCB technical review of the CR-3 Extended Power Uprate (EPU) License Amendment Request (LAR) (Reference 2). By teleconference on June 21, 2012, CR-3 discussed the draft RAI with the NRC to confirm an understanding of the information being requested. On August 1, 2012, the NRC provided a formal RAI required to complete its evaluation of the CR-3 EPU LAR (Reference 3).

Attachment A, "Response to Second Request for Additional Information – Mechanical and Civil Engineering Branch Technical Review of the CR-3 EPU LAR," provides the formal response to the RAI.

A001
NRR

Crystal River Nuclear Plant
15760 W. Powerline Street
Crystal River, FL 34428

In support of the EPU technical review RAI responses, five enclosures are provided in Attachment A:

Enclosure 1, "B&W Topical Report BAW-10149, Revision 1," to Attachment A describes the Power Train hybrid computer simulation of the normal and transient power operation of Babcock & Wilcox (B&W) nuclear power plants.

Enclosure 2, "Digital Power Train Secondary System Diagrams for CR-3," provides diagrams of the CR-3 secondary plant layout modified in the Digital Power Train simulation code.

Enclosure 3, "AREVA Calculation 32-9129875-001 – CR-3 Large Transient Testing Evaluation with DPT (Proprietary)," to Attachment A provides an evaluation of the effects of the proposed EPU on the plant capability and response to large operational transients.

Enclosure 4, "AREVA Technical Report 51-9126992-000 – CR-3 Large Transient Testing AIS (Proprietary)," to Attachment A provides key information required for evaluating the selected transients using the Digital Power Train simulation code.

Enclosure 5, "AREVA Calculation 32-9128425-000 – CR-3 EPU Best-Estimate Loss of Offsite Power (LOOP) for Large-Transient Testing (Proprietary)," to Attachment A provides an evaluation of the LOOP event analyzed with the RELAP5/MOD2-B&W code considering best-estimate EPU conditions.

Enclosures 3, 4, and 5 contain information that is considered proprietary. AREVA NP, Inc., as the owner of the proprietary information, has executed the affidavit provided in Attachment B, "AREVA Affidavit for Withholding Proprietary Information from Public Disclosure," and states that the documents provided in Enclosures 3, 4, and 5 are internal to AREVA and its customers and has been classified as proprietary, is customarily held in confidence, and not made available to the public. AREVA requests that the identified proprietary information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390(a)(4). A non-proprietary summary of the information provided in these documents was provided in the CR-3 EPU Technical Report (Reference 1, Attachment 7), and thus, non-proprietary copies of these documents are not provided as part of this submittal.

This correspondence contains no new regulatory commitments.

Also, as indicated in Attachment 2, "Additional Basis for the CR 3 EPU LAR Approval Schedule," of FPC to NRC letter dated October 25, 2011 (Reference 4), FPC plans to install the remaining EPU modifications at CR-3 during the current extended outage period in conjunction with the CR-3 containment repair activities; thus resulting in one outage cycle prior to CR-3 startup to EPU conditions. This plant startup strategy deviates from the strategy provided in Section 2.12, "Power Ascension and Testing Plan," of the CR-3 EPU Technical Report (Reference 1, Attachments 5 and 7). The initial startup prior to EPU operation will occur following Refuel 16 instead of Refuel 17, eliminating an operating cycle at the current licensed power level of 2609 MWt with some of the EPU modifications installed. As a result, post maintenance testing of the proposed Refuel 16 and Refuel 17 EPU modifications and baseline performance testing will be performed in a single plant startup period following Refuel 16 and prior to exceeding 2609 MWt.

The information provided by this response letter does not change the intent or the justification for the requested EPU license amendment. FPC has determined that this supplement does not affect the basis for concluding that the proposed license amendment does not involve a Significant Hazards Consideration. As such, the 10 CFR 50.92 evaluation provided in the June 15, 2011 submittal remains valid.

If you have any questions regarding this submittal, please contact Mr. Dan Westcott, Superintendent, Licensing and Regulatory Programs at (352) 563-4796.

Sincerely,

Jon A. Franke
Vice President
Crystal River Nuclear Plant

JAF/gwe

Attachments:

A. Response to Second Request for Additional Information – Mechanical and Civil Engineering Branch Technical Review of the CR-3 EPU LAR

B. AREVA Affidavit for Withholding Proprietary Information from Public Disclosure

Enclosures:

1. B&W Topical Report BAW-10149, Revision 1

2. Digital Power Train Secondary System Diagrams for CR-3

3. AREVA Calculation 32-9129875-001 – CR-3 Large Transient Testing Evaluation with DPT (Proprietary)

4. AREVA Technical Report 51-9126992-000 – CR-3 Large Transient Testing AIS (Proprietary)

5. AREVA Calculation 32-9128425-000 – CR-3 EPU Best-Estimate Loss of Offsite Power (LOOP) for Large-Transient Testing (Proprietary)

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

STATE OF FLORIDA

COUNTY OF CITRUS

Jon A. Franke states that he is the Vice President, Crystal River Nuclear Plant for Florida Power Corporation; 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.

Jon A. Franke
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 30 day of August, 2012, by Jon A. Franke.

Carolyn E Portmann

Signature of Notary Public
State of Florida

CAROLYN E. PORTMANN
Commission # DD 937553
Expires March 1, 2014
Bonded Thru Troy Fain Insurance 800-385-7019

(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 SECOND REQUEST FOR ADDITIONAL INFORMATION – MECHANICAL AND CIVIL ENGINEERING BRANCH TECHNICAL REVIEW OF THE CR-3 EPU LAR

RESPONSE TO SECOND REQUEST FOR ADDITIONAL INFORMATION – MECHANICAL AND CIVIL BRANCH TECHNICAL REVIEW OF THE CR-3 EPU LAR

By letter dated June 15, 2011, Florida Power Corporation (FPC) requested a license amendment to increase the rated thermal power level of Crystal River Unit 3 (CR-3) from 2609 megawatts (MWt) to 3014 MWt (Reference 1). On May 1, 2012, via electronic mail, the NRC provided a draft request for additional information (RAI) needed to support the Mechanical and Civil Branch (EMCB) technical review of the CR-3 Extended Power Uprate (EPU) License Amendment Request (LAR). By teleconference on June 21, 2012, CR-3 discussed the draft RAI with the NRC to confirm an understanding of the information being requested. On August 1, 2012, the NRC provided a formal RAI required to complete its evaluation of the CR-3 EPU LAR.

EMCB Requests for Additional Information

These questions pertain primarily to the licensee's analytical basis for proposing to eliminate certain large transient tests from the power ascension and testing plan

RAI EMCB-1

LAR Section 2.12, "Power Ascension and Testing Plan," page 2.12.1-7 of Attachment 5 of the original LAR discusses the Digital Power Train (DPT) code used by AREVA NP. Please provide a copy of Babcock and Wilcox (B&W) topical report BAW-10149, Revision 1, November 1981, which is referenced in the section discussing the DPT code. The DPT code was used, in part, as a basis to eliminate performing certain large transient tests as part of the licensee's power ascension and testing plan for EPU.

Response:

Enclosure 1, "B&W Topical Report BAW-10149, Revision 1," to this attachment provides a copy of the requested topical report which describes the Power Train hybrid computer simulation of the normal and transient power operation of B&W nuclear power plants. Power Train is the predecessor and the basis of the DPT simulation code referenced in Section 2.12 of the CR-3 EPU Technical Report (TR) (Reference 1, Attachments 5 and 7). B&W Topical Report BAW-10149, Revision 1, was approved for use by the NRC as promulgated in NRC (Thomas) to B&W (Taylor) letter dated November 28, 1983 (Reference 2).

As discussed in the FPC to NRC teleconference on June 21, 2012 regarding the RAI, the update of the Power Train simulation code to the DPT simulation code and subsequent revisions to the DPT simulation code have been controlled by the vendor quality assurance control program in accordance with 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

Additionally, the secondary plant layout, as shown in Figure 1-2, "Secondary System Diagram," of BAW-10149, was modified in the DPT simulation code specifically for CR-3 to more accurately simulate the CR-3 secondary plant systems. As requested by the NRC staff during the June 21, 2012 teleconference, diagrams of the modified secondary plant layout are provided in Enclosure 2, "Digital Power Train Secondary System Diagrams for CR-3," to this attachment.

RAI EMCB-2

LAR Section 2.12, page 2.12.1-8 of Attachment 5 of the original LAR refers to benchmarking performed for the DPT code. Please provide documentation that describes this benchmarking and its results in detail.

Response:

Enclosure 3, "AREVA Calculation 32-9129875-001 – CR-3 Large Transient Testing Evaluation with DPT (Proprietary)," to this attachment provides an evaluation of the effects of the proposed EPU on the plant capability and response to large operational transients analyzed with the DPT simulation code and describes the detailed process used to benchmark the DPT simulation code with CR-3 and includes the associated results.

Enclosure 3 contains information that is considered proprietary. A non-proprietary summary of the information provided in Enclosure 3 was provided in Section 2.12 of the CR-3 EPU TR (Reference 1, Attachment 7), and thus, a non-proprietary copy is not provided.

RAI EMCB-3

LAR Section 2.12, page 2.12.1-7 of Attachment 5 of the original LAR discusses the elimination of five transient tests from the EPU startup program. The tests are:

- Reactor Trip from 40% power
- Reactor Coolant Pump trip from 40% power
- Main Feedwater Pump trip from 75% power
- Loss of Offsite Power (LOOP) from 15% power.
- Dropped Control Rod from 75% power

The LAR states that the LOOP event was performed at measurement uncertainty recapture and EPU power conditions using the RELAP5/MOD2-B&W (R5/M2) computer code; and also describes the modeling of the remaining transients using the DPT code. Provide the results of these analyses at the same level of detail as those analyses discussed in Section 2.8.5, "Accident and Transient Analyses," of the original LAR.

Response:

Enclosure 3 to this attachment provides an evaluation of the effects of the proposed EPU on the plant capability and response to large operational transients analyzed with the DPT simulation code. Based on the calculational results, it is concluded that the effects of the proposed EPU on the plant operational capability will not be significant and that the plant changes that are necessary to achieve satisfactory results at EPU are consistent with the CR-3 design basis.

Enclosure 4, "AREVA Technical Report 51-9126992-000 – CR-3 Large Transient Testing AIS (Proprietary)," to this attachment provides key component and equipment modifications, setpoint changes, operating conditions, etc. required for evaluating the selected transients at pre-EPU and EPU conditions using the DPT simulation code.

Also, Enclosure 5, "AREVA Calculation 32-9128425-000 – CR-3 EPU Best-Estimate Loss of Offsite Power (LOOP) for Large-Transient Testing (Proprietary)," to this attachment provides an evaluation of the LOOP event analyzed with the RELAP5/MOD2-B&W (R5/M2) code considering best-estimate EPU conditions. The best-estimate LOOP event is first analyzed with the R5/M2 code at measurement uncertainty recapture (MUR) conditions. The results of this R5/M2 simulation was compared against a LOOP analysis performed with the CR-3 plant referenced simulator software, which has been benchmarked against actual plant data associated with an actual LOOP event.

Enclosures 3, 4, and 5 contain information that is considered proprietary. A non-proprietary summary of the information provided in Enclosures 3, 4, and 5 was provided in Section 2.12 of the CR-3 EPU TR (Reference 1, Attachment 7), and thus, non-proprietary copies are not provided.

RAI EMCB-4

LAR Section 2.12, page 2.12.1-8 of Attachment 5 of the original LAR discusses that the DPT code was developed modeling the Rancho Seco Nuclear Steam Supply System, and also provides discussion of benchmarking for Oconee and Three Mile Island events. Provide additional justification of the applicability of DPT to the CR-3 EPU design: demonstrate that there is reasonable agreement, for instance, between the results predicted for an operational event modeled using the RELAP5/MOD2-B&W code as compared to the same event modeled using the DPT code. The comparison would be especially valuable if CR-3 specific data were available to compare.

Response:

The overall process of ensuring reasonable agreement between the DPT simulation code, the CR-3 plant, and the R5/M2 code includes comparing the:

- CR-3 plant referenced simulator response data to data from actual plant transients similar in scope to those in the CR-3 large transient test (LTT) program;
- CR-3 plant referenced simulator LTT results at MUR conditions to the DPT simulation code LTT results at MUR conditions;
- DPT simulation code LTT results at MUR conditions to the DPT simulation code LTT results at EPU conditions;
- LOOP transient results using the CR-3 plant referenced simulator at MUR conditions to the LOOP transient results associated with the R5/M2 code at MUR conditions; and
- LOOP transient results using the R5/M2 code at MUR conditions to the LOOP transient results associated with the R5/M2 code at EPU conditions.

Large Transient Test Analyses Using the DPT Simulation Code

As discussed in Subsection 2.12.1.2.3, "Justification for Eliminations of Transient Testing," of Section 2.12 in the EPU TR (Reference 1, Attachments 5 and 7), the DPT simulation code was developed for the generic B&W lower primary loop plant configuration, which is consistent with the CR-3 primary loop configuration, and originally modeled the Rancho Seco secondary plant. To support the CR-3 EPU LTT program, the DPT simulation code was revised to specifically reflect the CR-3 plant configuration as described in Subsection 2.12.1.2.3 of the EPU TR. Seven

LTT scenarios were analyzed using both the CR-3 plant referenced simulator model and the DPT model; main turbine trip with runback, single reactor coolant pump (RCP) trip, single main feedwater (MFW) pump trip, dropped control rod, main turbine and reactor trip, single MFW booster pump, and loss of MFW Heater 6A. The LOOP LTT was analyzed using the R5/M2 code. Four of the LTT scenarios analyzed with the DPT simulation code support justifying the elimination of the associated LTTs listed in the Subsection 2.12.1.2.3 of the EPU TR; the reactor trip test, the RCP trip test, the MFW pump trip test, and the dropped control rod test.

To ensure the revised DPT simulation code accurately replicates CR-3 plant response, the code was initially configured to model CR-3 at MUR conditions. Key primary and secondary plant parameters were recorded during each test run and the DPT simulation code results were then compared to the CR-3 plant referenced simulator results. Refer to Enclosure 3 of this attachment for the benchmark comparison results. As shown in Appendix A, "DPT vs. CR-3 Simulator Comparison," of Enclosure 3, the benchmark comparison demonstrated good agreement between the DPT simulation code and the CR-3 simulator response at MUR conditions.

Upon successful completion of DPT simulation code benchmarking at MUR conditions, the DPT model was reconfigured to replicate the CR-3 plant configuration with major EPU modifications installed. Significant DPT simulation code inputs representing the plant at EPU conditions to support the LTT scenarios included:

- Increased condensate pump, MFW pump and MFW booster pump capacity,
- Addition of replacement steam generators,
- Integrated Control System (ICS) rescaling,
- Adjusted plant operating setpoints for T_{AVE} and main steam header pressure,
- Increased turbine bypass valve capacity, and
- Increased atmospheric dump valve capacity.

Each of the LTT scenarios analyzed at MUR conditions were re-analyzed on the EPU configured DPT simulation model. The results were then compared to the LTT results analyzed with the DPT simulation model at MUR conditions. With the exception of the single RCP trip transient test, the results of the LTT scenarios analyzed with the DPT simulation code concluded that the EPU modifications introduced no new or unexpected thermal-hydraulic (T-H) phenomena and there are no adverse or unexpected system interactions.

The results for the single RCP trip transient test indicated that, with the ICS runback endpoint set to 75% EPU power, the steam generator high level limit (HLL) on the unaffected steam generator could be challenged when MFW flow to the steam generators are rationed based on the available RCP combination. To address this issue, the ICS runback endpoint for a RCP trip runback is being reduced to approximately 70% to assure the steam generator HLL setpoint is not challenged.

As documented in Enclosure 3 of this attachment, the DPT simulation code to the CR-3 simulator benchmarking effort demonstrate there is reasonable agreement between the CR-3 LTT results predicted at MUR conditions using the DPT simulation code and the results predicted at MUR conditions using the CR-3 plant referenced simulator. The benchmarking also concluded that the LTT response at EPU conditions are as expected with no anomalous response indicated.

Loss of Offsite Power Analysis Using RELAP

The LOOP transient test was analyzed using the R5/M2 computer code. The R5/M2 model was benchmarked against the CR-3 plant referenced simulator response during a LOOP at MUR conditions using the same process as was used for the DPT model. The LOOP transient was re-analyzed with the R5/M2 configuration updated to replicate best-estimate EPU conditions and a similar comparison was made with the data obtained from the R5/M2 code in the MUR configuration. This analysis concluded that systems, structures, and components important to safety will continue to perform their functions at EPU conditions and that no adverse T-H phenomena or system interactions were introduced as a result of EPU operation.

As documented in Enclosure 5 of this attachment, the R5/M2 code to the CR-3 simulator benchmarking effort demonstrate there is reasonable agreement between the LOOP transient results predicted at MUR conditions using the RELAP code and the results predicted at MUR conditions using the CR-3 plant referenced simulator. The benchmarking also concluded that the LOOP event response at EPU conditions is as expected with no anomalous response indicated.

RAI EMCB-5

Provide additional information regarding the basis for statements in Section 2.12 of the original LAR regarding licensee credit taken for B&W pressurized water reactors industry operating experience and CR-3 plant specific operating experience at power levels greater than original licensed thermal power. The LAR did not discuss in detail the basis for these statements. In some cases, the NRC staff may use such operating experience in its review of the criteria discussed in Standard Review Plan, Chapter 14, Section 14.2.1, Paragraph III.C based on Matrix 12 of the NRC Review Standard for Extended Power Uprates.

Response:

Statements related to industry and CR-3 operational experience (OE) in Section 2.12 of the EPU TR are not intended to imply crediting OE from B&W units, including CR-3, operating at power levels greater than the original licensed thermal power. Rather, references to OE in Subsection 2.12.1.2.3 were made in the context of the initial CR-3 startup test program. During initial CR-3 startup testing; a reactor trip test and a RCP trip test were performed at 40% power, a MFW pump trip test and dropped rod test were performed at 75% power, and a LOOP test was performed at 15% power. Historically, CR-3 has experienced these transient events, some on multiple occurrences, from power levels near or greater than those specified in the initial startup test program.

Examples of large transients that have occurred at CR-3 include:

- Reactor trip from approximately 1850 MWt (approximately 61% of EPU power level) – this trip occurred on February 21, 2007 as a result of a loss of MFW flow following a circuit card failure in the ICS. This event was reported to the NRC as documented in Licensee Event Report (LER) 2007-002-00, "Reactor Trip Caused by Failed Circuit Board in the Main Feedwater Integrated Control System." (ADAMS Accession No. ML071130008)

- MFW pump trip and plant runback – this transient occurred on February 19, 2007 and involved a MFW pump trip from approximately 1960 MWt (approximately 65% of EPU power level) and an automatic ICS runback to approximately 1435 MWt (approximately 48%

of EPU power level). Operators further reduced power to approximately 1356 MWt (approximately 45% of EPU power level) in accordance with plant operating procedures. Since this transient event did not meet 10 CFR 50.73(a)(2) licensee reporting requirements, the event was not reported to the NRC; however, the event was evaluated internally as documented in the CR-3 corrective action program.

- Reactor trip from 2609 MWt (approximately 87% of EPU power level) – this transient occurred on January 27, 2009 as a result of an electrical fault on a balance of plant 4160 volt bus. The bus failure resulted in a loss of a MFW booster pump. The reactor was manually tripped prior to reaching the high Reactor Coolant System (RCS) pressure automatic trip. This event was reported to the NRC as documented in LER 2009-001-00, "Manual Reactor Trip Due To Loss of A 4160V Unit Bus Loads Caused By Incorrectly Connected Test Leads." (ADAMS Accession No. ML090830725)

- Dropped rods from 2609 MWt (approximately 87% of EPU power level) – this transient occurred on August 24, 2009 as a result of an improperly protected test jumper in the ICS. The reactor was manually tripped prior to automatic actuation of the Reactor Protection System. This event was reported to the NRC as documented in LER 2009-003-00, "Manual Reactor Trip Due To Group 7 Control Rods Insertion Caused By Inadequately Protected Test Jumper." (ADAMS Accession No. ML092960237)

- A partial LOOP from approximately 2490 MWt (approximately 83% of EPU power level) – this event occurred on September 6, 2004 when offsite power sources to the balance of plant buses and the Train B engineered safeguard 4160 volt bus were de-energized. The de-energization resulted in a reactor trip, trip of the RCPs, and trip of the MFW pumps. The plant was stabilized post-trip with the Emergency Feedwater System and the RCS operating in natural circulation mode. This event was reported to the NRC as documented in LER 2004-003-00, "Reactor Trip And Emergency Feedwater Actuation Caused By 230 Kilovolt Switchyard/Transmission Faults." (ADAMS Accession No. ML043100532)

To demonstrate the ability of the CR-3 simulator to replicate plant response during large transients, the plant-to-simulator comparison test data associated with four of the events described herein; two reactor trips, MFW pump trip, and LOOP; were evaluated and demonstrate that the CR-3 simulator accurately replicates the integrated response of the reference plant during large operational transients similar in scope to those included in the CR-3 LTT program. Thus, the plant simulator generated transient response data is an effective benchmarking tool for the DPT simulation and R5/M2 responses.

When CR-3 events occur, lessons learned from each occurrence are incorporated into plant procedures, training programs and, when necessary, result in plant design changes and/or simulator modifications. This iterative process has resulted in a progressive improvement in both plant design and the knowledge and experience base of the plant staff.

When evaluating the scope of the LTTs performed during initial CR-3 startup testing, it became apparent that re-running the tests at the percentages of EPU power specified in the initial CR-3 startup test plan would result in performing many of these tests below the power levels where similar events have historically occurred at CR-3. Historical plant experience demonstrates that plant design changes since initial plant startup have not resulted in significant differences in overall plant transient response. Analytical testing with the R5/M2 and DPT codes and comparing the results to the CR-3 plant referenced simulator further demonstrate that plant transient response has not been significantly altered by the inclusion of EPU-related modifications. The CR-3 LTTs were initiated from either an initial power level of 100% EPU power or, in the case of the RCP, MFW pump, and main turbine trip transients, from the highest power level that would not result in an automatic reactor trip.

Additionally, B&W industry OE has been incorporated into the engineering changes required to support operation at EPU condition:

Low Pressure Injection (LPI) System Cross-tie Modification – During the design of this modification, FPC evaluated the LPI cross-tie designs at the Arkansas (ANO) and Oconee (ONS) nuclear power plants. OE from these two evaluations was factored into the CR-3 cross-tie as follows:

- Based, in part, on OE from ANO, CR-3 engineering decided to reject the use of cavitating venturis from the CR-3 design. OE indicated that, with cavitating venturis, flow cavitation occurs during normal decay heat removal operation at ANO.

- Based, in part, on OE from ONS, CR-3 adopted an LPI cross-tie design similar to the cross-tie installed at ONS. The CR-3 design differs from ONS in that it replaces the orifice plate flow restrictors used in each injection line at ONS with throttle valves. Throttle valves offer more flexibility and can be adjusted without breaching the system pressure boundary.

Atmospheric Dump Valve Replacement – During the design of this modification, CR-3 sought a safety-related replacement valve option that would; (1) have the required flow capacity, (2) fit in a relatively confined space, and (3) eliminate valve body stresses imposed on the existing valve by the horizontal orientation of the valve actuator. Davis-Besse reported favorable operating experience with the safety-related, air-operated, angle valves used in their Main Steam Atmospheric Vent Valve System. The use of this type of valve at Davis-Besse served as precedent for their use at CR-3.

ICS Scaling, Runback and Automatic Unit Load Demand – The purpose of this modification is to rescale the ICS for operation at EPU power, and to establish new runback rates and endpoints. In support of this effort, OE from the operating B&W nuclear plants was obtained to compare the ICS runback rates and endpoints. Based, in part, on data gathered in this survey, CR-3 has modified the runback rates and endpoints for the three ICS pump trip runbacks; MFW pump, MFW booster pump, and RCP runbacks; and has eliminated the asymmetric rod runback.

Qualification and Preparation of Replacement Steam Generators for EPU – In conjunction with the steam generator supplier, CR-3 is reviewing OE from Davis-Besse, ONS and Three Mile Island regarding replacement steam generator response during plant startup and power operations with various steam generator downcomer orifice plate settings. Data from these units will be used, in part, to determine the initial downcomer orifice plate port area setting, and to predict the resulting CR-3 steam generator level operating range.

References

1. FPC to NRC letter dated June 15, 2011, "Crystal River Unit 3 – License Amendment Request #309, Revision 0, Extended Power Uprate." (ADAMS Accession No. ML112070659)

2. NRC (Thomas) to B&W (Taylor) letter dated November 28, 1983, "Acceptance for Referencing of Licensing Topical Report BAW-10149 Rev. 1, 'Power Train-Hybrid Computer Simulation of a Babcock & Wilcox Nuclear Power Plant.'" (ADAMS Accession No. 8312140207)

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

ATTACHMENT B

AREVA AFFIDAVIT FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. (AREVA NP) and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the AREVA NP information contained in document 51-9126992-000 entitled, "CR-3 Large Transient Testing AIS," dated December 2009, document 32-9128425-000 entitled, "CR-3 EPU Best-Estimate Loss of Offsite Power (LOOP) for Large Transient Testing," dated February 2010, and document 32-9129875-001 entitled, "CR-3 Large Transient Testing Evaluation with DPT," dated February 2011 and referred to herein as "Documents." Information contained in these Documents has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. These Documents contain information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in these Documents as proprietary and confidential.

5. These Documents have been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in these Documents

be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP'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 AREVA NP.

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

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

The information in the Documents is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in these Documents have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP 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 23rd

day of August, 2012.

Danita R. Kidd
NOTARY PUBLIC, STATE OF VIRGINIA
MY COMMISSION EXPIRES: 12/31/12
Reg. # 205569

DANITA R. KIDD
COMMONWEALTH OF VIRGINIA
NOTARY PUBLIC