Progress Energy

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

July 5, 2011
3F0711-07

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

Subject: Crystal River Unit 3 – Clarifications for TRACE Confirmatory Model Development for the CR-3 Extended Power Uprate

References:

1. Email from S. Lingam (NRC) to D. Westcott (CR-3) dated February 3, 2011, "Crystal River, Unit 3 - Pre-EPU RAIs for the Development of Confirmatory LOCA and non-LOCA Models"

2. Email from S. Lingam (NRC) to L. Wells (CR-3) dated June 13, 2011, "RE: RES Clarifications for Conference Call Supporting Development of Confirmatory TRACE Models for Crystal River EPU"

3. Email from S. Lingam (NRC) to L. Wells (CR-3) dated June 15, 2011, "RE: Follow-up on June 14, 2011 Teleconference on RES Clarification Request for TRACE Confirmatory Model Development Supporting the CR-3 EPU"

Dear Sir:

Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc., hereby provides responses to clarification requests needed to support the development of the NRC confirmatory TRACE models used in the review of the Crystal River Unit 3 (CR-3) Extended Power Uprate (EPU) License Amendment Request (LAR). These requests were provided to CR-3 via electronic mail on June 13, 2011 and June 15, 2011.

Additionally, FPC hereby provides the Analytical Input Summary (AIS) for the CR-3 EPU Loss of Feedwater and Feedwater Line Break (LOFW/FWLB) events to support the development of the NRC confirmatory TRACE models used in the review of the CR-3 EPU LAR. Providing a copy of the LOFW/FWLB AIS is a follow-up response to a request for CR-3 to provide Non-LOCA Analysis Report(s) as they are available. This request was provided to CR-3 via electronic mail (Item 4) on February 3, 2011.

Attachment A, "Response to June 13 and 15, 2011 Clarification Requests For Confirmatory TRACE Model," provides the CR-3 formal response to clarification questions requested in two electronic mails in support of a June 14, 2011 conference call supporting the development of confirmatory TRACE models for the CR-3 EPU.

Attachment C, "Crystal River Unit 3 Extended Power Uprate Loss of Feedwater and Feedwater Line Break Analytical Input Summary," contains the AREVA NP Inc. Document 51-9063671-

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
15760 W. Powerline Street
Crystal River, FL 34428

A001
NRR

001, "CR-3 EPU Loss of Feedwater-FW Line Break AIS," which also supports the development of confirmatory TRACE models for the CR-3 EPU and contains information that AREVA considers proprietary. AREVA NP Inc., as the owner of that proprietary information, has executed the affidavit provided in Attachment B and states that the identified proprietary information 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 version of the AREVA NP Inc. Document 51-9063671-001 is not available.

This correspondence contains no new regulatory commitments.

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 June 13 and 15, 2011 Clarification Requests For Confirmatory TRACE Model
B. Affidavit for Withholding Proprietary Information from Public Disclosure
C. Crystal River Unit 3 Extended Power Uprate Loss of Feedwater and Feedwater Line Break Analytical Input Summary

xc: NRR Project Manager

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 /LICENSE NUMBER DPR-72

ATTACHMENT A

RESPONSE TO JUNE 13 AND 15, 2011 CLARIFICATION REQUESTS FOR CONFIRMATORY TRACE MODEL

RESPONSE TO JUNE 13 AND 15, 2011 CLARIFICATION REQUESTS FOR CONFIRMATORY TRACE MODEL

On June 13, 2011, via electronic mail, the NRC requested the licensee/AREVA (Crystal River Unit 3 (CR-3)/AREVA NP Inc.) to clarify the plant response during the turbine trip and main feedwater line break events in support of development of the confirmatory TRACE models for the CR-3 Extended Power Uprate (EPU). In particular, the sequence of events in the primary loop, feedwater system starting from main feed pump and main steam system up to high pressure turbine inlet were requested. The NRC also requested clarifications on turbine control valves, turbine stop valves and steam chest configuration. The NRC needs to know what components (valves, pumps) will be reacting to the transients according to what control logic, and what components will remain unaffected.

The followings are transient specific questions sent in the June 13, 2011 e-mail:

1) For the turbine trip event, the NRC requests the following information:

    a. Describe the feedwater system response before and after reactor scram.
    b. Are the main steam isolation valves reacting to the transient?
    c. Please provide clarifications on the operation of the pressurizer spray and heater during the event.
    d. Is the makeup/letdown system operational during the transient?
    e. Will the operating mode of the reactor coolant pumps change after reactor scram?
    f. Please provide information on emergency feedwater operation and logic for this event.

2) For the main feedwater line break event, the NRC requests the following information:

    a. What is the break location and the response of plant components between the break location and the steam generator?
    b. Please provide additional information on emergency feedwater operation and logic for both steam generators.
    c. Please provide clarifications on the operation of the pressurizer safety valve, spray and heater during the event.
    d. Is the makeup/letdown system operational during the transient?
    e. Please also clarify the operation of the turbine control and stop valve and, in particular, control before reactor scram.
    f. Will the operation mode of the reactor coolant pumps change after reactor scram?

On June 15, 2011, via electronic mail, the NRC requested CR-3/AREVA to respond to additional questions in support of the development of the confirmatory TRACE models for CR-3 EPU. The following are transient specific questions sent in the June 15, 2011 e-mail:

1) For the turbine trip event, the NRC requests the following information:

    g. Describe the MFW flow boundary condition imposed for the turbine trip (TT) transient. The detail should include treatment of flow rate and temperature before and after reactor trip.

h. Describe the control mode of the reactor coolant pumps. Is it constant speed or constant flow control?

2) For the main feedwater line break event, the staff requests the following information:

g. Describe the control mode of the reactor coolant pumps. Is it constant speed or constant flow control?
h. Describe the sequence of events for the FWLB transient in the level detail requested for TT.
i. Describe the operation of the turbine control valve (TCV) before reactor trip occurs in the FWLB transient. Please discuss TCV operation for the plant as well as its implementation in the FWLB model.
j. Based on the plant drawings reviewed by the staff, CR-3 has two steam chests with inter-connecting pipes. Discuss or reference the bases for modeling the two steam chests with one component for FWLB. How does the current model account for unbalanced operation (one steam generator with broken feedwater and the other with intact feedwater) before turbine trips in the FWLB transient?
k. Describe the operation of main feedwater for OTSG-A (the intact feedwater loop) during FWLB before and after reactor trip.
l. Does the broken loop MSIV remain open during FWLB?
m. Describe analysis assumption regarding heat transfer in the broken loop OTSG for FWLB blowdown.
n. Describe the valves (and associated control systems) that control OTSG-A pressure during FWLB. Does OTSG-A depressurize during the event?

CR-3 Responses:

1. Turbine Trip Event Information Request Responses:

A summary of the turbine trip analysis associated with the CR-3 EPU is provided in Section 2.8.5.2.1, Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Steam Pressure Regulatory Failure," of the CR-3 EPU Technical Report (Reference 1, Attachment 7). The responses to the above NRC requests for additional information regarding the turbine trip event are as follows:

a. Describe the feedwater system response before and after reactor scram.

The turbine trip analysis uses a simplified main feedwater (MFW) model. A time dependent volume and a time dependent junction model constant MFW until the time of reactor trip, and then a linear ramp to zero flow over 3 seconds following MFW isolation. As indicated on Table 2.8.5.2.1-1 of the CR-3 EPU Technical Report, this behavior results in total isolation of MFW by ~6.5 seconds, which is before the time of peak steam line pressure of 8.0 seconds.

b. Are the main steam isolation valves reacting to the transient?

The main steam isolation valves are not modeled in the turbine trip event analysis. For a turbine trip transient, a faster isolation time of the main steam system results in a higher secondary side pressure. The turbine stop valves close much faster; (a stroke time of 0 second is considered) than the main steam isolation valves (3.0 – 5.0 seconds).

Therefore, the transient assumes that the turbine stop valves close instantaneously at the start of the event. Once the turbine stop valves are closed, there is no setpoint reached that would result in automatic closure of the main steam isolation valves. Therefore, the main steam isolation valves are not modeled in the turbine trip analysis.

c. Please provide clarifications on the operation of the pressurizer spray and heater during the event.

Pressurizer heaters are not modeled in the turbine trip transient analysis. The turbine trip transient increases the primary side pressure so the heaters are not assumed to turn on for this event.

The turbine trip event is the limiting secondary side overpressure event, but is not the limiting primary side overpressure event. As stated in Section 2.8.4.2, "Overpressure Protection During Power Operation," of the CR-3 EPU Technical Report, the limiting primary side overpressure event is Uncontrolled Control Rod Assembly Withdrawal from Low Power. As such, pressurizer spray is modeled in the turbine trip event because it delays the reactor trip on high primary side pressure. This in turn will allow more heat transfer to the secondary side which increases the peak secondary side pressure. Pressurizer spray is modeled as a constant 190 gpm of flow that begins when the primary side pressure at the hot leg tap exceeds 2205 psig and continues until the primary side pressure at the hot leg tap decreases below 2155 psig.

d. Is the makeup/letdown system operational during the transient?

The makeup and letdown systems are assumed to be in operation during the transient. These systems are not included in the RELAP5 model. The net flow in and out of the systems is considered balanced. The net heat loss associated with makeup and letdown is accounted for by adjusting the net heat addition associated with the reactor coolant pumps (RCP). When the secondary heat load indicates insufficient RCP heat, the RCP energy is supplemented by use of passive heat structures in the cold leg piping to achieve the desired heat.

e. Will the operating mode of the reactor coolant pumps change after reactor scram?

The reactor coolant pumps (RCPs) are modeled to continue operating throughout the transient. This is conservative because it continues to add the heat associated with the RCPs and it maximizes the heat transferred from the primary side to the secondary side. In other words, if the RCPs were tripped at reactor trip, the reactor coolant system (RCS) flow rate would significantly decrease and less heat would be transferred to the secondary side resulting in a lower peak secondary side pressure. The turbine trip analysis assumes the RCPs provide constant volumetric flow equal to the minimum RCS flow rate (374,880 gpm) throughout the transient.

f. Please provide information on emergency feedwater operation and logic for this event.

Emergency feedwater is not modeled for the turbine trip analysis. If emergency feedwater were to actuate, it would condense steam and reduce the secondary side

pressure. Therefore, excluding emergency feedwater operation from the turbine trip event model is conservative.

Additionally, the turbine trip event is a rapid event where the peak secondary side pressure is reached in less than 10 seconds. The delay time associated with emergency feedwater reaching the steam generators is greater than 10 seconds; 40 seconds after low steam generator level is reached. Therefore, by the time emergency feedwater reaches the steam generators, the limiting secondary pressure in the turbine trip event would have already been reached. Consequently, emergency feedwater operation is not modeled in the turbine trip event.

g. Describe the MFW flow boundary condition imposed for the turbine trip (TT) transient. The detail should include treatment of flow rate and temperature before and after reactor trip.

The turbine trip event models steady state main feedwater (MFW) flow and temperature until reactor trip, and then begins a linear ramp to no MFW flow over an additional 3 seconds. The initial MFW temperature is 460°F and does not change throughout the transient. The initial MFW flow was determined during the steady-state initialization of the model to achieve the targeted primary side average temperature at the targeted power level and targeted turbine header pressure. A flow rate of ~1860 lbm/sec was determined during the steady-state; 1865.7 lbm/sec to OTSG A and 1857.7 lbm/sec to OTSG B.

h. Describe the control mode of the reactor coolant pumps. Is it constant speed or constant flow control?

The reactor coolant pumps (RCPs) are modeled to continue operating throughout the transient. The RCPs provide constant volumetric flow equal to the minimum RCS flow rate (374,880 gpm) throughout the transient. This flow rate is based on four constant speed RCPs in operation.

2. Feedwater Line Break Event Information Request Responses:

A summary of the main feedwater line break (FWLB) analysis associated with the CR-3 EPU is provided in Section 2.8.5.2.4, "Feedwater System Pipe Breaks Inside and Outside Containment," of the CR-3 EPU Technical Report (Reference 1, Attachment 7). The responses to the above NRC requests for additional information regarding the FWLB event are as follows:

a. What is the break location and the response of plant components between the break location and the steam generator?

The noding diagrams for the Main Feedwater (MFW) System and the break location have previously been transmitted from CR-3 to the NRC per letter dated March 15, 2011 (Reference 2).

The FWLB analysis assumes the break occurs on OTSG B, five feet upstream of the entrance to the 18" x 18" x 18" tee closest to the steam generator. This location

maximizes the break area because it is in the 18 inch piping. The components between the break location and the steam generator include:

- 18" x 18" x 18" tee-connection
- 18" x 14" reducers
- Piping and elbows connecting the ring headers to the 18" x 14" reducers
- MFW ring headers (14 inches)
- MFW riser nozzles (3 inches – 16 per OTSG)
- MFW spray heads (90 holes per nozzle, each hole has 0.188 inch diameter on the lower side of the plate, 16 spray heads per OTSG)

b. Please provide additional information on emergency feedwater operation and logic for both steam generators.

The Emergency Feedwater Initiation and Control (EFIC) System provides for initiation and control of the EFW System when the MFW System is not available. As described in the CR-3 Final Safety Analysis Report (FSAR) Section 7.2.4 (Reference 3), EFW is initiated based on any of the following:

- All four RCPs are tripped;
- Both MFW pumps are tripped and reactor power > 20%;
- Both MFW pumps are tripped and NI/RPS is not in shutdown bypass;
- Low level in either OTSG;
- Low pressure in either OTSG;
- High Pressure Injection System actuation on both Engineered Safety Action System channels; or
- Total loss of all MFW; MFW flow < 17% of range in both flow loops with reactor power ≥ 50%.

The FWLB analysis assumes the low OTSG level function, sensed in either OTSG, actuates the EFW System. The low OTSG pressure function, sensed in either OTSG, is also modeled to actuate the EFW System and is the primary EFIC function assumed in the secondary system mass and energy release analysis.

Once EFW is initiated, the FWLB analysis assumes 60 seconds before the flow reaches the steam generators and a total of 550 gpm of emergency feedwater is modeled. These values are conservative with respect to the limiting values assumed in the LOFW event to quickly mitigate overheating; 660 gpm at 40 seconds.

The Feed Only Good Generator (FOGG) logic prevents feeding an affected steam generator when pressure in the affected steam generator drops below 600 psig while the pressure in the unaffected steam generator remains greater than 600 psig. The FOGG logic also prevents feeding the steam generator with the lowest pressure when pressure in both steam generators is below 600 psig and the differential pressure between the steam generator is greater than 125 psi. The FWLB analysis assumes FOGG logic actuation, thus all EFW flow is assumed to the unaffected steam generator.

c. Please provide clarifications on the operation of the pressurizer safety valve, spray and heater during the event.

Pressurizer safety valves

The pressurizer safety valves (PSVs) are modeled in the FWLB analysis with a nominal setpoint of 2500 psig. The analysis conservatively models a 3% lift tolerance and blowdown modeled is 4% of the nominal setpoint.

Per the AREVA NP Inc. methodology defined in BAW-10193 (Reference 4), no valve accumulation is modeled because Babcock & Wilcox (B&W) designed plants have PSVs with steam-to-seat internals (i.e., no liquid loop seals). BAW-10193 states that, in EPRI/C-E test of Dresser and Crosby PSVs with steam-to-seat internals, no accumulation was observed. The saturated steam discharge rate through the PSVs is 317,973 lbm/hr/valve at 2750 psig, for a total relief capacity of 635,946 lbm/hr.

Pressurizer spray and heaters

Pressurizer heaters are not modeled. The FWLB increases the primary side pressure so the heaters are not assumed to be activated for this event.

Pressurizer spray was not modeled in the FWLB analysis for peak RCS pressure. However, the spray was included for the FWLB case that confirmed the PSV fluid inlet temperature remained above 600°F. For the event that modeled pressurizer spray, the flow rate is a constant 190 gpm of flow that begins when the primary side pressure at the hot leg tap exceeds 2205 psig and continues until the primary side pressure at the hot leg tap decreases below 2155 psig. A stroke time of 0.0 second is used for the pressurizer spray valve to maximize the impact of the pressurizer spray.

d. Is the makeup/letdown system operational during the transient?

The makeup and letdown systems are assumed to be in operation during the transient. These systems are not included in the RELAP5 model. The net flow in and out of the systems is considered balanced. The net heat loss associated with makeup and letdown is accounted for by adjusting the net heat addition associated with the reactor coolant pumps (RCP). When the secondary heat load indicates insufficient RCP heat, the RCP energy is supplemented by use of passive heat structures in the cold leg piping to achieve the desired heat.

e. Please also clarify the operation of the turbine control and stop valve and, in particular, control before reactor scram.

Consistent with the AREVA NP Inc. methodology defined in BAW-10193 (Reference 4), the Integrated Control System behavior is not modeled in the FWLB analysis. Prior to reactor trip, the turbine stop valves are assumed to remain fully open and the turbine control valves are assumed at the pre-accident position. Once reactor trip occurs, the turbine stop valves close using maximum nominal stroke time of 200 milli-seconds. The turbine stop valves are closed ~10 seconds after the start of the event.

f. Will the operation mode of the reactor coolant pumps change after reactor scram?

The reactor coolant pumps (RCPs) are modeled as remaining on throughout the event. This maintains the additional heat added by the RCPs. The FWLB analysis assumes the RCPs provide constant volumetric flow equal to the minimum RCS flow rate (374,880 gpm) throughout the transient.

g. Describe the control mode of the reactor coolant pumps. Is it constant speed or constant flow control?

The RCPs provide constant volumetric flow equal to the minimum RCS flow rate (374,880 gpm) throughout the transient. This flow rate is based on four constant speed RCPs in operation.

h. Describe the sequence of events for the FWLB transient in the level detail requested for TT.

A summary of the sequence of events for the FWLB analysis is provided in Section 2.8.5.2.4 of the CR-3 EPU Technical Report (Reference 1, Attachment 7). Specifically, Table 2.8.5.2.4-1 provides the timeline sequence from event initiation to termination.

i. Describe the operation of the turbine control valve (TCV) before reactor trip occurs in the FWLB transient. Please discuss TCV operation for the plant as well as its implementation in the FWLB model.

Automatic control of the turbine is accomplished by the Integrated Control System (ICS). A general description of the ICS is provided in Section 7.2.3 of the CR-3 FSAR (Reference 3). The controlling subsystems of the ICS (i.e., turbine control, steam generator feedwater control, and reactor control) normally operate in the automatic mode in response to a demand signal from the unit load demand (ULD). The integrated master control receives the megawatt (MW) demand signal from the ULD and converts this signal into a demand for the feedwater, turbine, and reactor control. The MW demand is compared with the main generator output, and the resulting MW error signal is used to change the steam pressure setpoint. The turbine control valves (TCVs) then change position to control steam pressure.

The ULD normally operates in the automatic mode under steady state conditions above 25% rated thermal power with all four reactor coolant pumps in operation. The subsystems control functions are kept within pre-established bounds under other than normal automatic operation by a "tracking" feature built into the ICS. In the tracking mode, the load demand follows the control subsystem that is in manual or the control system in a limiting condition by using the main generator output as the demand input to the ULD. Tracking continues until the limiting condition is brought back to within the pre-established deadband or the control subsystem that is in manual is returned to automatic operation.

During a FWLB event, the ICS would initiate a runback in response to the loss of feedwater pumps and the errors between the demand and the variable in the turbine control subsystem of the ICS would switch the ICS to tracking mode and follow manual input or use actual generator output as the demand input to the ICS load controller. The

ICS load controller would respond by closing the TCVs in an attempt to maintain the targeted turbine header pressure. In the FWLB analysis, this behavior would reduce the severity of the event because it would take longer to remove inventory from the steam generators. As such, the ICS response is not modeled in the FWLB analysis. The TCVs are assumed to be maintained in the steady-state position prior to reactor trip. Once reactor trip occurs, the turbine stop valves close and the TCVs no longer have any impact on the FWLB event.

j. Based on the plant drawings reviewed by the staff, CR-3 has two steam chests with inter-connecting pipes. Discuss or reference the bases for modeling the two steam chests with one component for FWLB. How does the current model account for unbalanced operation (one steam generator with broken feedwater and the other with intact feedwater) before turbine trips in the FWLB transient?

As shown on Figure 10-4, Sheet 3 of the CR-3 FSAR (Reference 3), the two steam chests each cross connect the two steam generators. Main steam line (MSL) A-1 and MSL B-1 provide steam to one steam chest and MSL A-2 and MSL B-2 provide steam to the other steam chest. Thus, it is reasonable to assume this design will not cause a significant asymmetry in steam line pressure response. Since there is no significant asymmetry between the steam lines on a given steam generator, the response seen in the two steam chests is assumed to be the same (i.e., the pressure response in MSL A-1 is the same as MSL A-2 and the pressure response in MSL B-1 is the same as MSL B-2). Since both steam chests are essentially symmetric, a simplified analysis model using only one steam chest is considered acceptable.

k. Describe the operation of main feedwater for OTSG-A (the intact feedwater loop) during FWLB before and after reactor trip.

The initiation of the FWLB results in an immediate loss of main feedwater (MFW) flow to the affected steam generator (OTSG-B), since all MFW flow is diverted to the break. Due to the rapid depressurization from the break, it is conservatively assumed that MFW flow to the unaffected steam generator (OTSG-A) is also diverted to the break. Therefore, MFW flow to the unaffected steam generator (OTSG-A) is assumed to be terminated immediately.

l. Does the broken loop MSIV remain open during FWLB?

The EFIC System will close the main steam isolation valve (MSIV) on the broken loop due to low OTSG pressure. However, the turbine stop valves (TSVs) close before this occurs; approximately 10 seconds from event initiation. Once the TSVs are closed, the behavior of the MSIV does not have a significant effect on the primary system pressure response or pressurizer level in the FWLB analysis. Nevertheless, the MSIV behavior is modeled.

m. Describe analysis assumption regarding heat transfer in the broken loop OTSG for FWLB blowdown.

Heat structures are modeled in both steam generators to simulate the heat transfer from the primary side to the secondary side across the steam generator tubes. Passive heat structures representing the OTSG shell, shroud, tube support plates and tube sheets are also included in the model. During blowdown of the affected OTSG, these heat structures are used to determine the heat transfer.

n. Describe the valves (and associated control systems) that control OTSG-A pressure during FWLB. Does OTSG-A depressurize during the event?

Prior to reactor trip on RCS high pressure in the FWLB analysis, OTSG A (intact secondary loop) depressurizes because both steam generators are connected at each turbine steam chest. Once reactor trip occurs, the turbine stop valves close and OTSG-A begins to re-pressurize. The pressure in OTSG A increases to the main steam safety valve setpoints, and the safety valves open as needed to maintain the OTSG A pressure at acceptable levels. The secondary pressure response for both SG is presented in Figure 2.8.5.2.4-4 of the CR-3 EPU Technical Report (Reference 1, Attachment 7).

References

1. CR-3 to NRC letter dated June 15, 2011, "Crystal River Unit 3 – License Amendment Request #309, Revision 0, Extended Power Uprate"

2. CR-3 to NRC letter dated March 15, 2011, "Crystal River Unit 3 - Request for Additional Information Required for the Development of Confirmatory LOCA and non-LOCA Models for the CR-3 Extended Power Uprate," (Accession No. ML11167A107)

3. Final Safety Analysis Report, Progress Energy Florida, Crystal River Unit 3, Revision 32.1

4. AREVA NP Inc. Document 43-10193PA-00, "RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors"

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 /LICENSE NUMBER DPR-72

ATTACHMENT B

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 Engineering Information Record, Document Number 51-9063671, Revision 001, entitled "CR-3 EPU Loss of Feedwater – FW Line Break AIS," dated June 2011 and referred to herein as "Document." Information contained in this Document 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. This Document contains 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 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 this Document 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 Document 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 this Document 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 27th

day of June 2011.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/14
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2014