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

January 31, 2012
3F0112-11

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

Subject: Crystal River Unit 3 – Corrections to the Extended Power Uprate LAR #309 Vendor Affidavit and Technical Report (TAC No. ME6527)

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

Dear Sir:

By letter dated June 15, 2011, Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc., 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. This correspondence provides supplemental information to the CR-3 Extended Power Uprate (EPU) license amendment request (LAR) #309 reflecting corrections to the CR-3 EPU LAR and includes revised pages, where appropriate.

Attachment A, "Revised Affidavit Regarding AREVA NP Request to Withhold Proprietary Information," replaces the vendor affidavit initially provided in the CR-3 EPU LAR; Attachment 6, "Affidavit For Withholding Proprietary Information From Public Disclosure." The revised affidavit specifically clarifies the proprietary information contained in the CR-3 EPU LAR that AREVA NP, Inc. is requesting be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390(a)(4). Please remove and replace the vendor affidavit, in copies of Attachment 6 of the CR-3 LAR #309 (Reference), with the attached revised affidavit.

Attachment B, "Revised Pages of the CR-3 EPU Technical Report," provides a revision to the CR-3 EPU Technical Report (TR) to correct initial conditions and results of Section 2.8.5.2.1, "Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Steam Pressure Regulatory Failure," and ancillary corrections to Table 2.8.5.0-7, "Non-LOCA EPU Analysis Limits and Analysis Results." During a vendor review of the documents supporting the CR-3 EPU TR, it was determined that the main feedwater (MFW) isolation was modeled coincident with a reactor trip. This is inconsistent with the analysis input summary document, which states that MFW isolates coincident with a turbine trip. To determine the impact of this inaccuracy, sensitivity cases have been run for the turbine trip event, assuming the MFW trip isolates coincident with a turbine trip. This does not affect the conclusion that the proposed EPU is acceptable with respect to the turbine trip event. Table 2.8.5.0-7 also includes a clarification for Note 9 to provide more specific detail regarding the acceptability of the results associated with the Locked Rotor Accident. In addition, Figures 4 through 7 of Enclosure 2, "ADV/Fast Cooldown System Modification," to Appendix E of the CR-3 EPU TR have been revised to

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

A001
NRR

provide the correct valve numbers and lines sizes to be consistent with the related text in the enclosure and the current atmospheric dump valve (ADV) design. This administrative oversight, with the inaccurate valve and line identifications, do not adversely affect the EPU safety analyses, ADV modification design, or operational aspects of the Fast Cooldown System (FCS) or ADVs. Please remove and replace the CR-3 EPU TR pages, in copies of Attachment 5 and Attachment 7 of the CR-3 LAR #309 (Reference), with the attached revised pages.

Attachment C, "Clarification Information to the CR-3 EPU Technical Report Section 2.7.3.1 Regarding the FCS Batteries," provides information regarding the hydrogen generation impact on the CR-3 Control Room Complex Ventilation System due to the addition of the FCS batteries and clarifies Section 2.7.3.1, "Control Room Area Ventilation System," of the CR-3 EPU TR. This clarification information does not alter the conclusion that the Control Room Complex Ventilation System will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU.

This correspondence contains no new regulatory commitments.

The information provided by this correspondence does not change the intent or the justification for the requested EPU license amendment. FPC has determined that these corrections and supplemental information to the CR-3 EPU LAR do not affect the basis for concluding that the proposed license amendment (Reference) does not involve a Significant Hazards Consideration. As such, the 10 CFR 50.92 evaluation provided in the June 15, 2011 submittal (Reference) 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/pk

Attachments:

A. Revised Affidavit Regarding AREVA NP Request to Withhold Proprietary Information

B. Revised Pages of the CR-3 EPU Technical Report

C. Clarification Information to the CR-3 EPU Technical Report Section 2.7.3.1 Regarding the FCS Batteries

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, doing business as Progress Energy Florida, Inc.; 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 31 day of January, 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

REVISED AFFIDAVIT REGARDING AREVA NP REQUEST TO WITHHOLD PROPRIETARY INFORMATION

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 Sections 2.8.1, "Fuel System Design," and 2.8.3, "Thermal and Hydraulic Design," of Engineering Information Record 51-9076487-000 entitled "Crystal River Unit 3 Extended Power Uprate Technical Report," dated March 2010, and Sections 2.8.1 and 2.8.3 in Attachment 5 of the Crystal River Unit 3 Extended Power Uprate license amendment request dated June 15, 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 these 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 26th

day of January 2012.

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

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT B

REVISED PAGES OF THE CR-3 EPU TECHNICAL REPORTS

CR-3 EPU LAR #309

ATTACHMENT 7 REVISED PAGES
(NON-PROPRIETARY)

Table 2.8.5.0-7 Non-LOCA EPU Analysis Limits and Analysis Results

FSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
14.1.2.1	Uncompensated Operating Reactivity Changes	N/A	N/A	N/A
14.1.2.2	Startup Accident	Peak thermal power	< 112 %RTP (Note 7)	53.6 %RTP
		Peak RCS pressure	< 2764.7 psia	2746.3 psia
14.1.2.3	Rod Withdrawal at Rated Power Operation Accident	Peak thermal power	< 112 %RTP (Note 7)	110.1 %RTP
		Peak RCS pressure	< 2764.7 psia	2674 psia
14.1.2.4	Moderator Dilution Accident	Peak thermal power	< 112 %RTP (Note 7)	109.5 %RTP
		Peak RCS pressure	< 2764.7psia	2698 psia
14.1.2.5	Cold Water Accident	DNBR	(Note 1)	(Note 1)
		Peak RCS pressure	< 2764.psia	2337 psia
14.1.2.6	Loss-of-Coolant Flow Accidents	4 PCD: DNBR	>1.45	1.54
		1 PCD: DNBR	>1.45	1.62
		Locked Rotor: DNBR	>1.45	1.40 (Note 9)
14.1.2.7	Stuck-Out, Stuck-In, or Dropped Control Rod Accident	DNBR	(Note 7)	(Note 7)
		Peak RCS pressure	< 2764.7 psia	< 2764.7 psia (Note 3)
14.1.2.8	Load Rejection Accident (Turbine Trip)	Peak RCS pressure	< 2764.7 psia	2576 psia
		Peak SG pressure	< 1279.7 psia	1166 psia
		Peak steam line pressure	< 1169.7 psia	1153 psia
		Fuel damage	(Note 1)	(Note 1)

Table 2.8.5.0-7 Non-LOCA EPU Analysis Limits and Analysis Results				
FSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
14.2.2.8	Waste Gas Decay Tank Rupture Accident	Radiological consequences.	2.5 rem TEDE	(Note 11)
14.2.2.9	Loss of Feedwater	Peak RCS pressure	< 2764.7 psia	2750.7 psia
14.2.2.9	Main Feedwater Line Break	Peak RCS pressure	< 3014.7 psia (Note 2)	2896.2 psia

Notes:

1. The criterion was met by confirming that the normalized power-to-normalized flow ratio remained equal to or less than the initial value throughout the transient.
2. The RCS pressure limit is based on faulted conditions (120%of design pressure).
3. A spectrum of dropped rod cases were run at different times-in-life and all cases predicted peak RCS pressures less than 2764.7 psia.
4. These parameters and limits apply to the REA cases with initial power levels less than 5% RTP. Limiting case is EOC at HZP.
5. This parameter and limit applies to the REA cases with initial power levels greater than 5% RTP. Limiting case is BOC at 20% RTP.
6. The steam line break accident results in a decrease in primary and secondary system pressures. Following dryout and depressurization of the affected steam generator, the primary system and unaffected steam generators would repressurize to normal post-trip conditions. Consequently, system pressure limits are not challenged during a MSLB accident. The reactor is conservatively tripped at time zero and the post-trip maximum thermal power does not exceed 26% RTP.
7. The limit on peak thermal power is related to DNBR. For B&W 177-FA plants, maximum allowable peaking limits are developed at the RCS DNB safety limit statepoints. The RCS DNB safety limits represent a locus of points for which the minimum DNBR is equal to the analysis limit. These safety limits are calculated at the design overpower condition (112% of reactor thermal power). Thus, demonstration that thermal power remains below 112% provides assurance that the minimum DNBR remains above limits. The limiting DNB transients are loss-of-coolant-flow transients, which are specifically analyzed for DNBR. The acceptance criterion with respect to DNBR during the Dropped Rod accident is evaluated on a cycle-specific basis using statepoints derived from the CR-3 EPU Dropped Rod accident evaluations.
8. Radiological Consequences are provided in Table 2.9.2-1.
9. A value below the analysis limit indicates that the thermal design limit (TDL) is penetrated and that there may be fuel failures. Fuel failure is acceptable for Condition IV events, including the Locked Rotor Event, and the dose analysis for locked rotor assuming fuel failures is provided in Section 2.9.2. As discussed in Section 2.8.3, other operational limits result in no fuel failures for the locked rotor event for the conceptual EPU cycles.
10. The acceptance criteria for the fuel melt and clad temperature limits is proprietary to AREVA NP. The results presented meet the acceptance criterion defined by AREVA NP, as documented in topical report ANP-2788P (Reference 6). The results provided are for the cases deemed limiting at EOC and BOC with respect to fuel failure potential.
11. As discussed in Section 2.5.6.1 and Section 2.10, there is no impact from EPU on waste gas decay tank content limit, which is controlled by the Offsite Dose Calculation Manual, to maintain tank contents within the limit specified in Improved Technical Specification 5.6.2.13, Explosive Gas and Storage Tank Radioactivity Monitoring Program.

- The analysis modeled the reactor to be at hot full power conditions with a nominal average temperature of 582°F, consistent with the increase in T_{AVG} planned in conjunction with the EPU. The hot leg pressure was assumed to be 2170 psia.
- The RCPs were assumed to continuously operate throughout the transient providing a constant reactor coolant volumetric flow equal to the minimum RCS flow rate (374,880 gpm).
- The initial pressurizer level was modeled as nominal minus uncertainty (200 inches) to maximize the secondary peak pressure.
- Two PSVs were modeled with a nominal lift setpoint of 2514.7 psia, plus 3% lift tolerance, and 0% accumulation. A blowdown of 4% was also used. The PSV capacity was modeled as 317,973 lbm/hr/valve at 2764.7 psia.
- Pressurizer spray was modeled with a design flow of 190 gpm. Pressurizer heaters were not modeled.
- The pressurizer power operated relief valve (PORV) was not credited.
- The main feedwater flow was linearly ramped down to zero flow over 3 seconds following turbine trip.
- Reactor trip was modeled to occur on a nominal high RCS pressure setpoint plus uncertainty (2400 psia).
- After reactor trip, the core heat generation rate was conservatively based on 1.0 times the ANS 1971 decay heat standard for fission plus heavy actinides.
- The tripped rod worth assumed for the analysis is based on a minimum shutdown margin of 1.0 %Δk/k. This is less than the minimum Modes 1 and 2 shutdown margin required for the EPU (i.e., 1.3 %Δk/k as detailed in a separate attachment associated with Improved Technical Specifications (ITS) changes).
- A Doppler temperature coefficient (-1.30×10^{-5} Δk/k/°F) and moderator coefficient (0.0 Δk/k/°F), typical of beginning-of-cycle conditions, were used since they yield the maximum rate of power increase.
- The MSSVs were modeled to lift at the nominal setpoints plus 3% lift tolerance and 3% accumulation. A nominal blowdown value of 5% was used.
- The actions of Emergency Feedwater Initiation and Control (EFIC), including Emergency Feedwater EFW flow, are not credited since the peak pressures occur prior to EFW delivery to the once through steam generators (OTSGs).
- OTSG tube plugging of 0% was modeled.
- Offsite power was available, which is consistent with the plant licensing basis.

Results

The sequence of events for the turbine trip accident is listed in Table 2.8.5.2.1-1 and the calculated results are tabulated in Table 2.8.5.2.1-2. Figures 2.8.5.2.1-1 through 2.8.5.2.1-5 show transient plots of the significant plant parameters following a turbine trip accident.

The RCS pressure peak was reached at 5.5 seconds and remained below the limit of 2764.7 psia. The peak steam line pressure and the peak OTSG pressure were reached at 6.0 seconds. The steam line maximum pressure was less than the corresponding limit of 1169.7 psia and the OTSG peak pressure was below the limit of 1279.7 psia.

The acceptance criteria pertaining to fuel damage and dose consequences are not challenged for this accident. This is confirmed as the power remained at or below the initial value of 3026.1 MWt, forced flow and subcooling are maintained, and RCS pressure remains within limits. The remaining acceptance criteria pertain to the RCS and secondary side (OTSG and steam line) pressures remaining below code pressure limits. From the results provided in Table 2.8.5.2.1-2, there is ample margin (i.e., more than 100 psi) between the RCS and OTSG peak pressures and their respective limits. The peak steam line pressure maintained a minimum margin of 16.7 psi. Based on these results, it can be concluded that, for a turbine trip with reactor power at the EPU conditions, the MSSVs in conjunction with the high RCS pressure reactor trip provide sufficient overpressure protection to maintain OTSG and steam line pressures below the ASME code pressure limits. Therefore, the design basis limits for fission product barriers are not exceeded or altered.

2.8.5.2.1.3 Conclusion

CR-3 has reviewed the analyses of the decrease in heat removal events described above and concludes that the analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. CR-3 further concludes that it has been demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB limits will not be exceeded as a result of these events. Based on this, CR-3 concludes that the plant will continue to meet the requirements of FSAR Sections 1.4.6, 1.4.9, and 1.4.27 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the turbine trip event.

2.8.5.2.1.4 References

1. BAW -10193PA-00, "RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors".

2. BAW-10164PA-06, "RELAP5/MOD2-B&W--An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis".

Table 2.8.5.2.1-1: Sequence of Events for Full Power Turbine Trip

Event	Time (sec)
Transient begins TSVs close	0
MSSVs begin to lift	0.8
RCS hot leg pressure reaches the RPS setpoint of 2400 psia	2.64
MFW flow reaches 0	3.0
Maximum RCS pressure is reached	5.5
Maximum Steam Line pressure is reached	6.0
Maximum Steam Generator pressure is reached	6.0

Table 2.8.5.2.1-2: Results for Full Power Turbine Trip

Parameter	Analysis Result	Acceptance Criteria
Maximum RCS pressure (psia)	2576	2764.7
Maximum steam line pressure (psia)	1153	1169.7
Maximum OTSG pressure (psia)	1166	1279.7

Figure 2.8.5.2.1-1: Turbine Trip Core Power versus Time

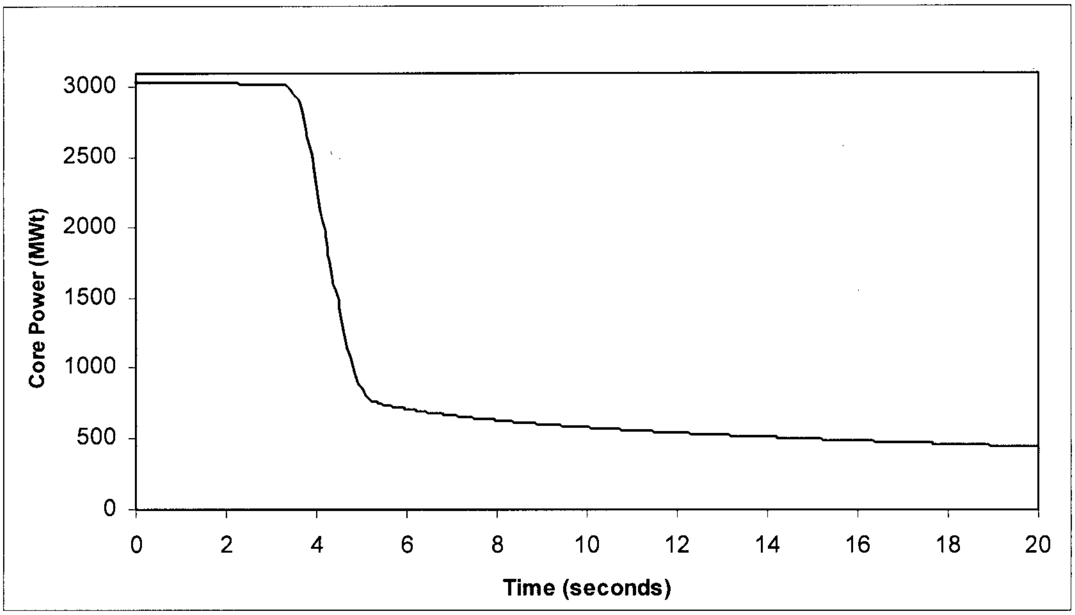


Figure 2.8.5.2.1-2: Turbine Trip RCS Pressure versus Time

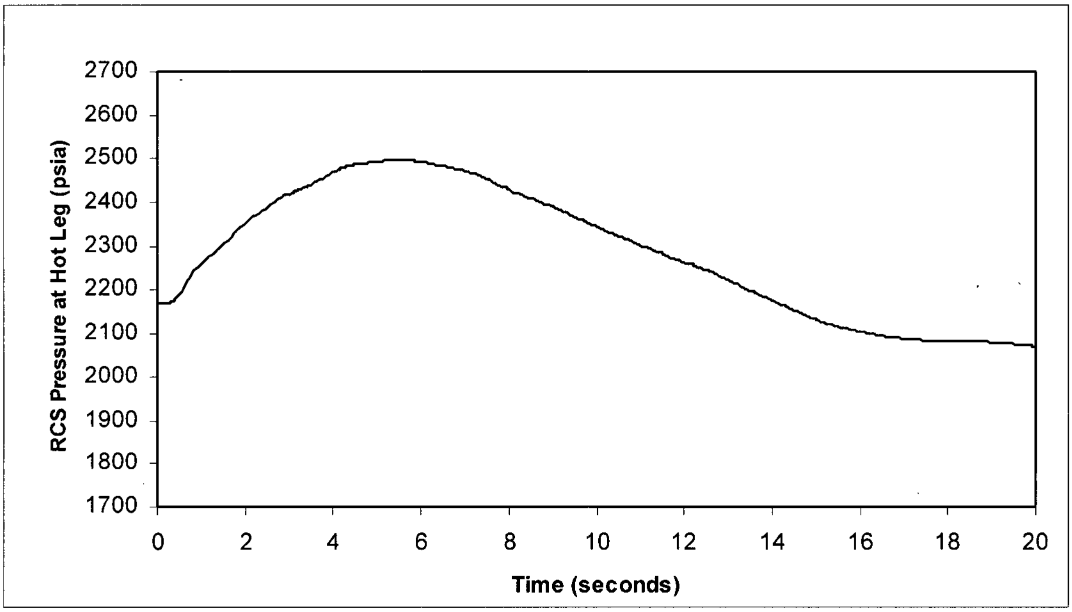


Figure 2.8.5.2.1-3: Turbine Trip RCS Average Temperature versus Time

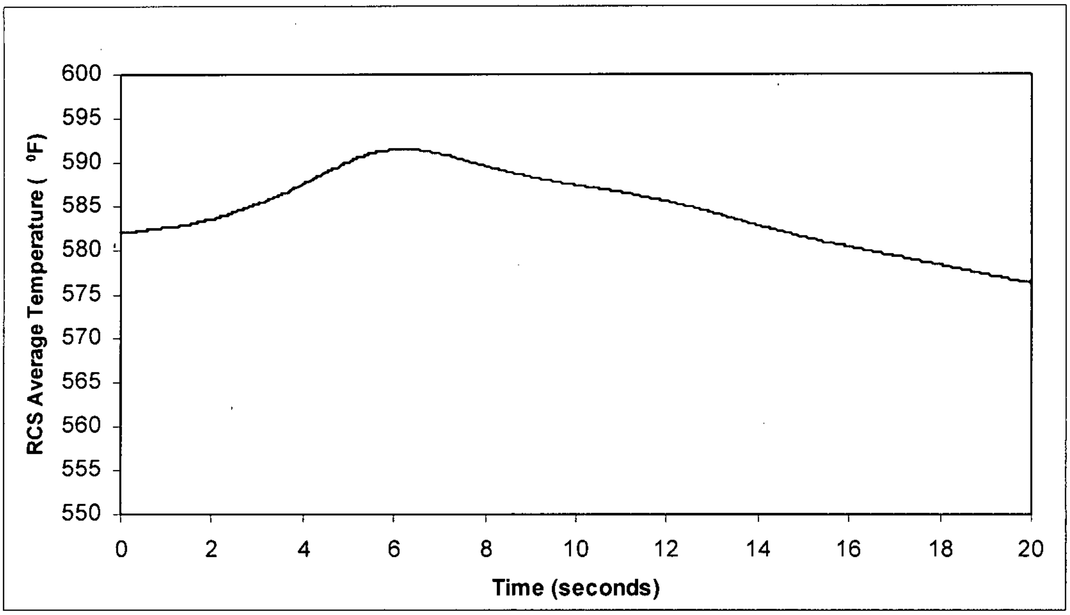


Figure 2.8.5.2.1-4: Turbine Trip Pressurizer Level versus Time

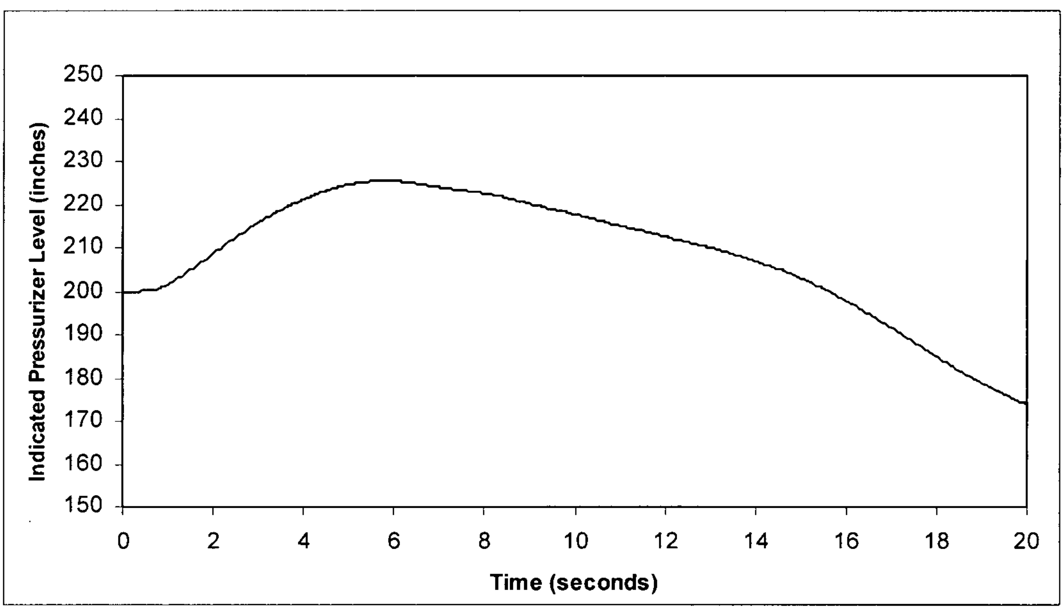


Figure 2.8.5.2.1-5: Turbine Trip Steam Line Pressure versus Time

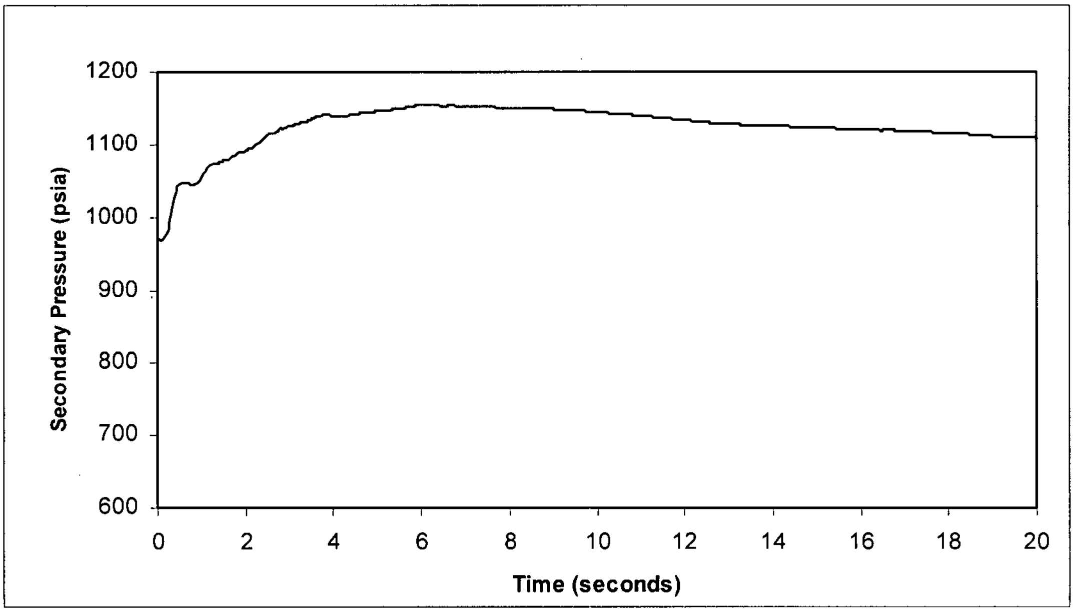


FIGURE 4: OTSB 3B – ATMOSPHERIC DUMP VALVE (CURRENT)

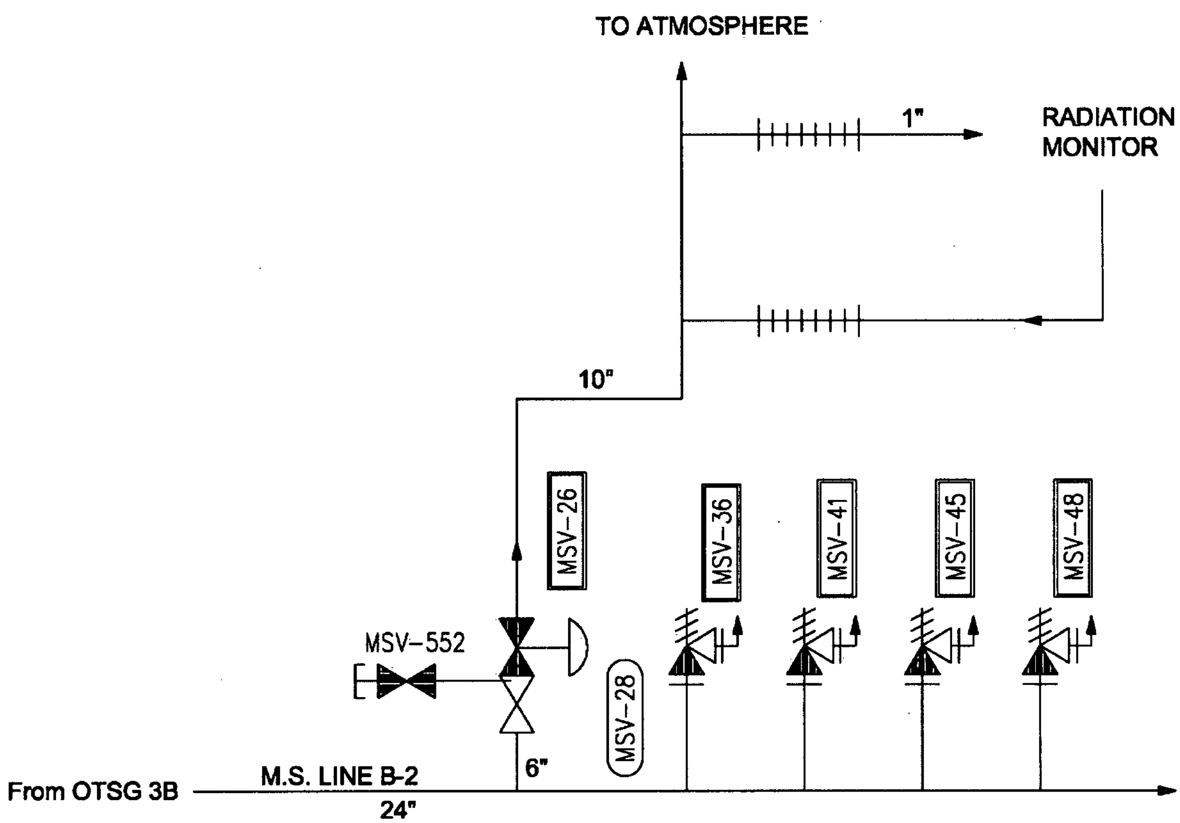


FIGURE 5: OTSB 3B – ATMOSPHERIC DUMP VALVE (NEW)

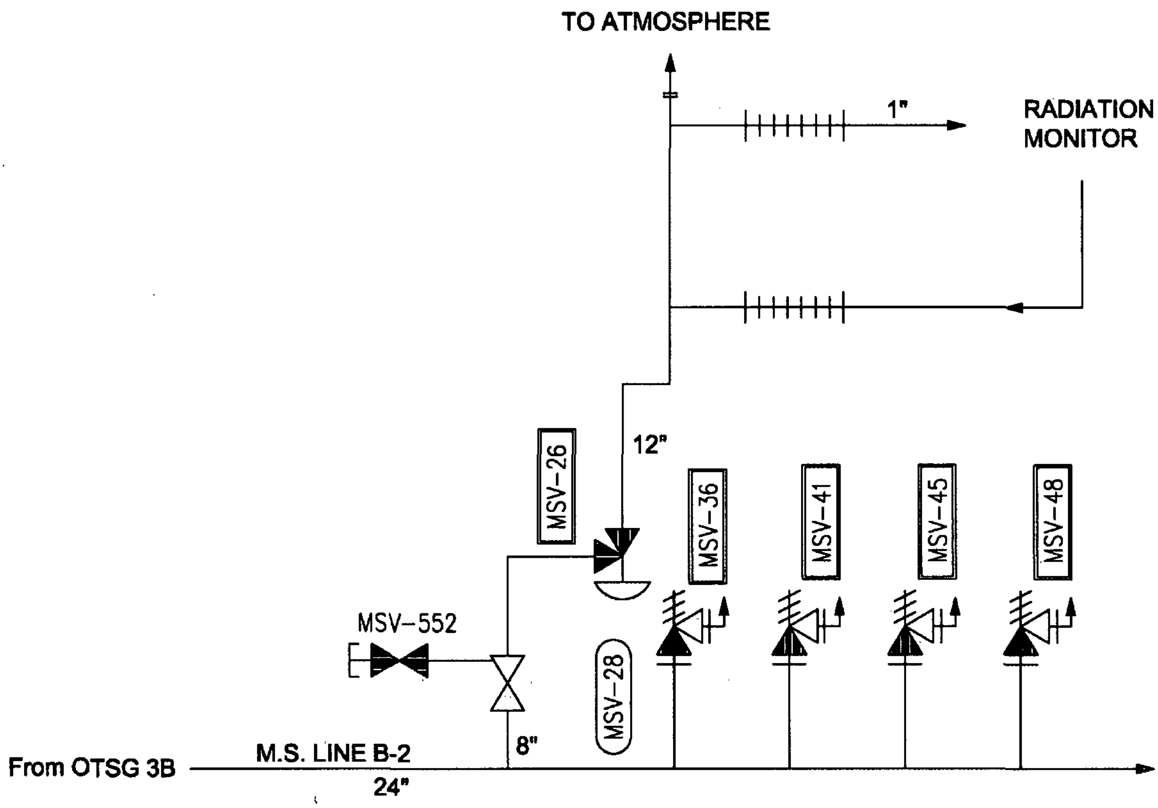


FIGURE 6: OTSG 3A – ATMOSPHERIC DUMP VALVE (CURRENT)

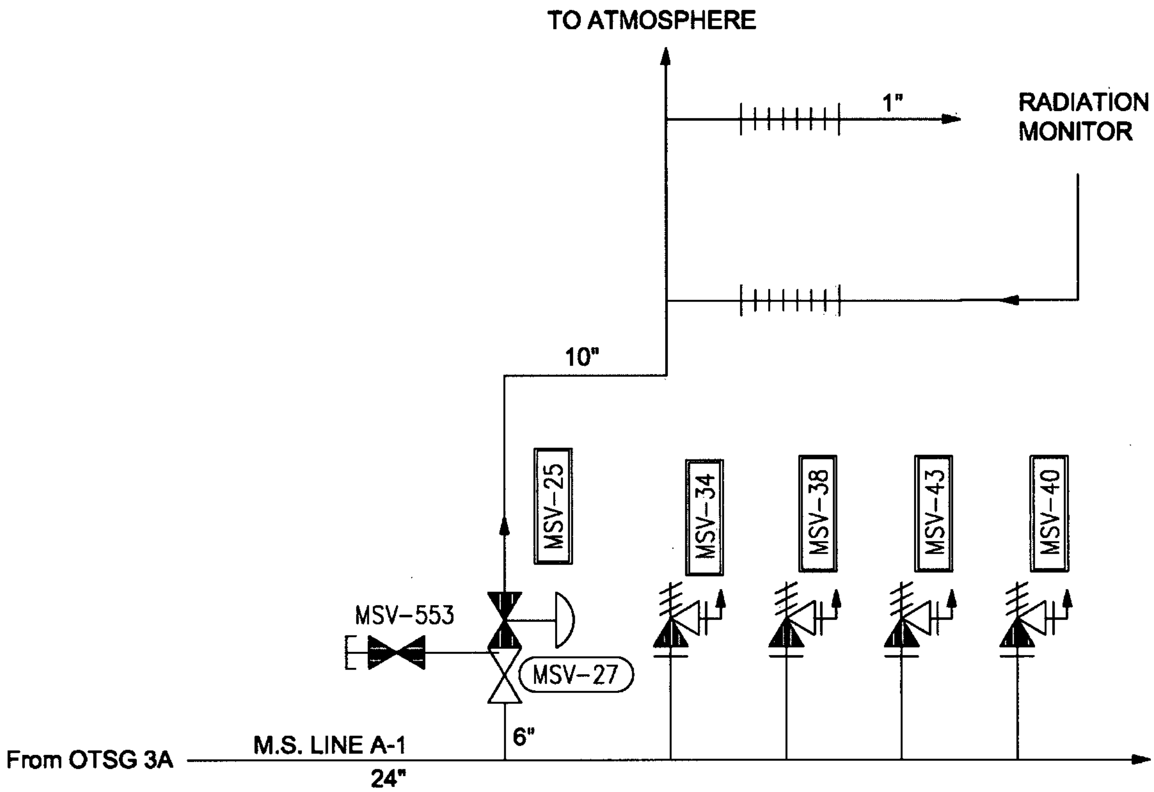
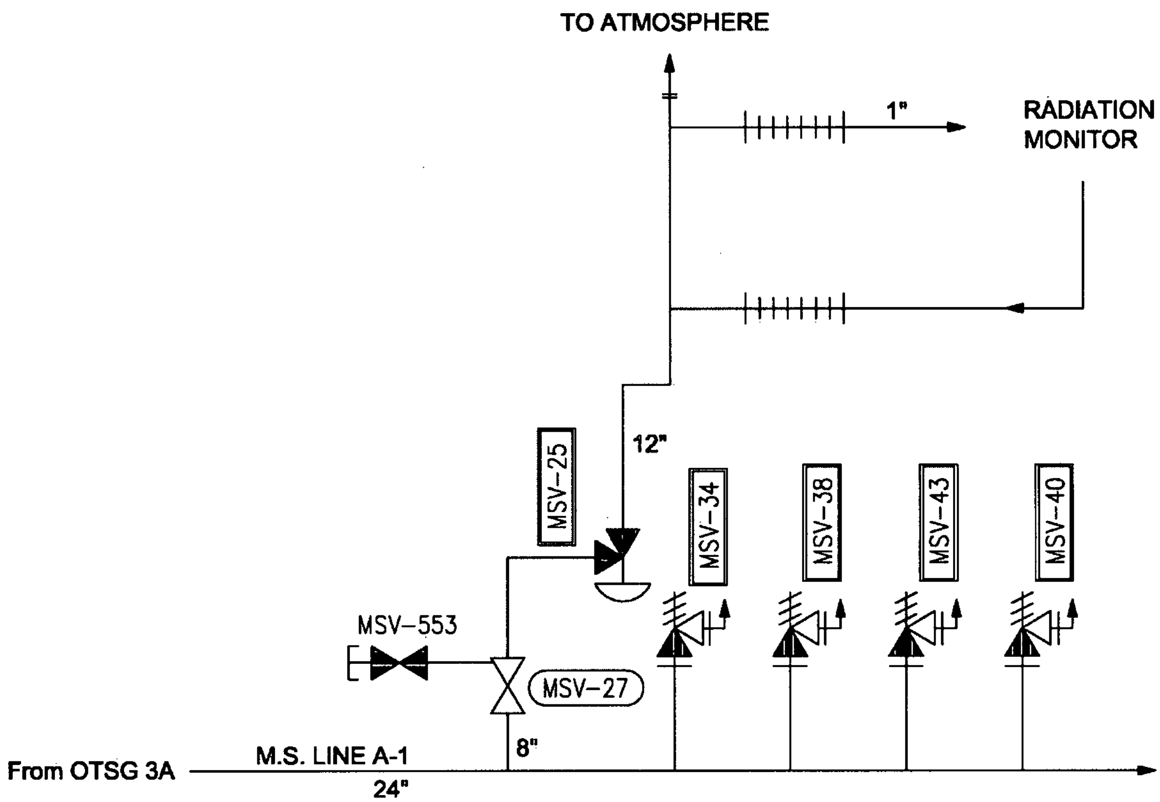


FIGURE 7: OTSG 3A – ATMOSPHERIC DUMP VALVE (NEW)



FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT C

CLARIFICATION INFORMATION TO THE CR-3 EPU TECHNICAL REPORT SECTION 2.7.3.1 REGARDING THE FCS BATTERIES

CLARIFICATION INFORMATION TO THE CR-3 EPU TECHNICAL REPORT SECTION 2.7.3.1 REGARDING THE FCS BATTERIES

By letter dated June 15, 2011, Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc., 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. This attachment provides information regarding the hydrogen generation impact on the CR-3 Control Room Area Ventilation System due to the addition of the Fast Cooldown System (FCS) batteries and clarifies Section 2.7.3.1, "Control Room Area Ventilation System," of the CR-3 Extended Power Uprate (EPU) Technical Report (TR).

As described in Enclosure 2, "ADV/Fast Cooldown System Modification," to Appendix E of the CR-3 EPU TR, the FCS will provide rapid depressurization of the Main Steam System via the atmospheric dump valves (ADVs) to support mitigation of design basis accidents. The FCS batteries have the potential for hydrogen generation in the associated Battery Rooms and the Control Complex. This potential for additional hydrogen generation was inadvertently omitted from Section 2.7.3.1, "Control Room Area Ventilation System," of the CR-3 EPU TR. Thus, the statement, in part, "...the production and buildup of hydrogen in the Control Complex and Battery Rooms do not change as a result of the EPU conditions..." infers that the change in hydrogen production and buildup is zero. This was not the intended meaning and warrants clarification.

The FCS batteries are typical vented lead acid cells and will emit hydrogen during recharging similar to the existing Class 1E station batteries. The proposed FCS battery design will add 24 cells per Battery Room; existing Class 1E stations batteries consist of 116 cells per Battery Room. The Control Complex and Battery Room hydrogen buildup calculation has been evaluated assuming the additional FCS battery cells. FPC has concluded that the change in hydrogen concentration in the Battery Rooms and Control Complex is insignificant; the calculated hydrogen concentration increase is < 0.01%. Additionally, the FCS battery chargers are supplied from non-safety related power and are therefore incapable of recharging the FCS batteries during the postulated worst-case Station Blackout or during a loss of offsite power. As a result, the FCS batteries will not emit hydrogen until non-safety related power is restored to the FCS battery chargers which typically occurs following restoration of the safety-related Battery Room ventilation. Therefore, the addition of the FCS batteries will not adversely impact the capability of the Control Room Area Ventilation System to maintain hydrogen concentrations within limits and the Control Room Area Ventilation System will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU.