Progress Energy

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

April 12, 2012
3F0412-07

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

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

References:

1. 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)
2. NRC to CR-3 letter dated March 2, 2012, "Crystal River Unit 3 Nuclear Generating Plant – Request for Additional Information for Extended Power Uprate License Amendment Request (TAC No. ME6527)" (Accession No. ML120550561)

Dear Sir:

By letter dated June 15, 2011, Florida Power Corporation, 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 (Reference 1). On March 2, 2012, the NRC provided a request for additional information (RAI) required to support the SCVB technical review of the CR-3 Extended Power Uprate (EPU) License Amendment Request (LAR) (Reference 2).

The attachment, "Response to Request for Additional Information to Support NRC Containment and Ventilation Branch (SCVB) Technical Review of the CR-3 EPU LAR," provides the formal response to the RAI needed to support the SCVB technical review of the CR-3 EPU LAR.

In support of the EPU technical review RAI responses, an enclosure is provided. The enclosure, "Summary of ECCS and BS Pump NPSH Analyses," provides a tabular presentation summary of the Emergency Core Cooling System and Reactor Building Spray System pump NPSH evaluations performed for CR-3 considering EPU conditions.

This correspondence contains no new regulatory commitments.

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A001
NRR

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

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

Attachment: Response to Request for Additional Information to Support NRC Containment and Ventilation Branch (SCVB) Technical Review of the CR-3 EPU LAR

Enclosure: Summary of ECCS and BS Pump NPSH Analyses

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 12th day of April, 2012, by Jon A. Franke.

Charlene Miller

Signature of Notary Public
State of Florida

CHARLENE MILLER
Notary Public - State of Florida
My Comm. Expires Nov 12, 2012
Commission # DD 837023
Bonded Through National Notary Assn.

(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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION TO SUPPORT NRC CONTAINMENT AND VENTILATION BRANCH (SCVB) TECHNICAL REVIEW OF THE CR-3 EPU LAR

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION TO SUPPORT NRC CONTAINMENT AND VENTILATION BRANCH (SCVB) TECHNICAL REVIEW OF THE CR-3 EPU LAR

By letter (Reference 1) 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. On March 2, 2012, the NRC provided a request for additional information (RAI) required to support the SCVB technical review of the CR-3 Extended Power Uprate (EPU) License Amendment Request (LAR).

SCVB RAIs

SCVB-1.1

The concrete surface area (105,941 ft^2) listed in Table 2.6.1-5, "Containment Structural Heat Sink Input," differs from the CR-3's containment concrete surface area as listed in Updated Final Safety Analysis Report (UFSAR) Table 14.45 (117,800 ft^2). Please explain the inconsistency between the two.

Response:

The concrete surface area of 105,941 ft^2 listed in Table 2.6.1-5 "Containment Structural Heat Sink Input" of the CR-3 EPU Technical Report (TR) (Reference 1, Attachments 5 and 7) used in the EPU containment response analyses is consistent with the Segment 5 surface area value of Option 1 provided on Sheet 2 of 2 in CR-3 Final Safety Analysis Report (FSAR) Table 14.45, "Reactor Building Data for Reactor Building Maximum Pressure Analysis." As correctly indicated on Sheet 1 of 2 of Table 14.45, Segment 5 material consists of paint and concrete. The reduced concrete surface area provides a conservative evaluation of containment pressure/temperature response as a reduced concrete surface area results in less condensation capability and reduces the energy removal rate from containment atmosphere following a loss-of-coolant accident (LOCA) or main steam line break (MSLB) accident.

SCVB-1.2

As stated in Section 2.6.1.2, the Improved Technical Specification Limiting Condition for Operation 3.6.4, maximum value for containment pressure during normal operation, is being revised from 17.7 psia (3 psig) to 16.2 psia (1.5 psig) as a result of the EPU. The change was implemented in the containment accident analysis for short term loss-of-coolant accident (LOCA). However, higher initial containment pressure was assumed for long term LOCA and main steam line break analyses. Explain the reasons for this inconsistency?

Response:

Preliminary containment analyses considering operation at EPU conditions were performed using an initial containment pressure of 17.7 psia (3 psig). However, the short term containment pressure response following a LOCA resulted in a maximum containment pressure which provided small margin against the containment design pressure of 69.7 psia (55 psig); therefore, the LOCA analyses were re-performed using a reduced initial containment pressure of 16.7 psia (2 psig). A further initial containment pressure reduction, to 16.2 psia (1.5 psig), was required to achieve a satisfactory containment pressure margin for the short term LOCA containment response. The containment pressure response for the MSLB accident and long term LOCA do not pose challenges to the available pressure margin related to the containment design pressure. Thus, the MSLB accident and long term LOCA containment responses were not re-analyzed and continue to assume a more conservative initial containment pressure of 17.7 psia (3 psig) and 16.7 psia (2 psig), respectively.

SCVB-1.3

The three postulated single case failures are described in page 2.6.1-7 of the CR-3 EPU Technical Report. Please discuss the difference between the first single failure scenario (loss of offsite power (LOOP) with failure of one emergency diesel generator (EDG)) and the second single failure scenario (One Reactor Building (RB) spray pump fails to start with or without LOOP). The information is requested because it appears that a failure of one RB spray pump would be automatically covered by the LOOP with failure of one EDG.

Response:

The three single failure scenarios discussed in Section 2.6.1, "Primary Containment Functional Design," in the CR-3 EPU TR (Reference 1, Attachments 5 and 7) are consistent with the scenarios described in the CR-3 FSAR. The following provides additional clarification for the first two single failure cases;

- The failure of an EDG to start following a LOOP results in: (1) one Emergency Core Cooling System (ECCS) train available for Reactor Coolant System (RCS) heat removal; and (2) one RB spray pump and one RB cooling unit available for containment heat removal.

- The failure of one RB spray pump to start with or without LOOP results in: (1) both ECCS trains available for RCS heat removal; and (2) one RB spray pump and one RB cooling unit for containment heat removal. In this single failure case, offsite power availability is not relevant since both EDGs are assumed to be capable of starting.

The primary difference in these two scenarios is the number of ECCS trains available for RCS heat removal.

SCVB-1.4

Section 2.6.3.2 provides the details of the main steam line break analysis at the EPU conditions. Explain the differences between the current licensing basis analysis and the EPU analysis, with special attention to the hardware modifications as a result of the EPU (e.g., modification of main feedwater (MFW) and MFW booster pumps). In particular, discuss all changes to the inputs, assumptions, single failures, MFW flow rates, MFW pump start times, and the codes used in the analysis. In addition, provide the reasons for considering a closure time of 31 seconds for the MFW isolation valves when faster closing isolation valves capable of closing in 21 seconds are being implemented for the EPU. Also, explain how feedwater flow from the MFW pump into the containment is apportioned through the MFW isolation valve during its closure?

Response:

The MSLB mass and energy (M&E) release analysis is used to generate the limiting containment response to a secondary system pipe rupture, as explained in Section 2.6.3.2, "Mass & Energy Release Analysis for Secondary System Pipe Ruptures," of the CR-3 EPU TR (Reference 1, Attachments 5 and 7). The inputs and boundary conditions as well as the codes used for the EPU MSLB analysis used to generate M&E releases to containment are discussed in detail in Section 2.6.3.2 of the CR-3 EPU TR. The CR-3 current licensing basis analysis, which includes the replacement once through steam generators, is described in FSAR Section 14.2.2.1, "Steam Line Failure Accident." The system response for both the EPU analysis and the current licensing basis analysis was modeled using the RELAP5/MOD2-B&W computer code (Reference 2). The following discusses aspects of the EPU MSLB M&E release analysis that are different from the current licensing basis analysis.

- The nominal reactor power in the EPU analysis is 3014 MWt. The initial power level considers heat balance error, for a total core power of 3026.1 MWt. The current licensing basis analysis is based on an initial power level of 2619.4 MWt.

- The RCS average temperature for the EPU analysis is 582°F, reflecting an increase to the RCS operating temperature in conjunction with the EPU. The current licensing basis analysis uses an RCS average temperature of 579°F.

- The EPU analysis reflects the increase in the minimum shutdown margin to 1.3% Δk/k at hot zero power in accordance with the requirements for EPU power operation. The current licensing basis analysis reflects the current minimum shutdown margin of 1.0% Δk/k.

- The EPU analysis is based on closure of the MFW block valves and the startup block valves in 31 seconds; 30 second stroke time and 1 second Emergency Feedwater Initiation and Control (EFIC) signal delay. The EPU analysis also assumes the MFW pump suction valves close in 21 seconds; 20 second stroke time and 1 second EFIC signal delay. The current licensing basis analysis is based on closure of these valves in 34 seconds. The closure time of the MFW low-load block valve is modeled at 67 seconds; 66 second stroke time and 1 second EFIC signal delay for both the EPU analysis and the current licensing basis analysis.

- The EPU analysis and the current licensing basis analysis both assume a maximum steam generator inventory for the power level modeled in the analysis. Thus, the inventory is higher in the EPU analysis due to the higher initial power level.

- The EPU analysis considers the MFW system hardware changes (i.e., the MFW booster pumps and the MFW pumps) that will be implemented prior to the EPU. The current licensing basis analysis reflects the current MFW system hardware.

The limiting single failure assumption is the same for the EPU analysis and the current licensing basis analysis. Both analyses assume that the MFW pump on the affected loop fails to trip after the MFW isolation signal has been generated.

Preliminary EPU MSLB analyses performed for core response and M&E release assumed a MFW isolation valve closure time of 31 seconds. However, additional changes to the MFW System were later identified as being necessary to support EPU operation. As a result, it was determined that the MFW pump suction valves needed to be replaced with faster closing valves to prevent a return to criticality in support of the MSLB analysis for core response. Consequently, the MFW pump suction valves are being replaced with valves that close in 20 seconds following a 1 second EFIC signal delay; 21 seconds assumed in the MSLB analysis for core response.

As explained in Section 2.6.3.2 of the CR-3 EPU TR, evaluations of the MSLB M&E release analysis confirmed that the reduction in feedwater flow due to the faster closing MFW pump suction valves more than offsets the MFW System flow changes as a result of the EPU modifications. As a result, assuming a 31 second closure time for the MFW isolation valves, including the MFW pump suction valves, conservatively bounds the MSLB M&E release with respect to the planned EPU configuration; new MFW and MFW booster pumps and faster MFW pump suction valve closure time. Additionally, the evaluations confirmed that the change to a faster closing MFW pump suction valve did not alter the limiting single failure considered for the event. The evaluations concluded that failure to trip the MFW pump on the affected loop continues to be the limiting single failure.

The MFW pump suction valves are located downstream of the MFW booster pumps on the suction side of the MFW pumps. Flow passing through the MFW pumps must first pass through the MFW pump suction valves. The valves are modeled to close using a constant rate of area change once the EFIC signal is received. While closing, the flow through the valves is determined by the response of the MFW booster pumps and the MFW pumps to the changing system pressures including the changes caused by valve closure. The MFW block valves, the MFW low-load block valve, and the startup block valves also close as a result of the EFIC signal. Once the MFW valves are closed, the model allows flashing in the MFW lines downstream of the valves to force inventory into the faulted steam generator and ultimately into containment.

SCVB-1.5

Please provide the EPU impact on the emergency core cooling system (ECCS) ability to provide homogeneous atmospheric mixing within containment. In accordance with the requirements of Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.44, Subsection (b)(1) as related to mixed atmosphere for currently licensed reactors, confirm that the CR-3 containment has the capability of ensuring a mixed atmosphere following a LOCA at EPU conditions. Summarize the CR-3's containment design that supports this assessment.

Response:

As discussed in Section 2.6.4, "Combustible Gas Control in Containment," of the CR-3 EPU TR (Reference 1, Attachments 5 and 7), operating at EPU conditions does not impact structures or systems currently credited for assuring hydrogen mixing in the CR-3 containment. The RB volume, RB Spray and Cooling System capability, RB sump pH control, and RB atmospheric monitoring systems are not altered as a result of EPU operation.

Also, hydrogen concentration in the CR-3 containment currently reaches a CR-3 target value of 3.1% in approximately 14.5 days, which is below the combustion and detonation concentration that could cause a loss of containment integrity as stated in 10 CFR 50.44(a)(2). The largest contributor to hydrogen accumulation is the radiolysis term, which is a linear function of reactor power. A qualitative assessment was performed assuming an overall power of 3026 MWt and concluded that the time to reach a containment hydrogen concentration of 3.1% following a design basis accident at EPU conditions is between 11 and 12 days.

Since the time to reach the CR-3 target value is not significantly reduced as a result of EPU operation and the structures and systems currently credited for assuring hydrogen mixing in the CR-3 containment are not altered as a result of the EPU, FPC concludes that CR-3 continues to have sufficient capabilities to maintain a homogenously mixed containment atmosphere consistent with 10 CFR 50.44(b)(1).

SCVB-1.6

The applicability of NRC Generic Letter (GL) 96-06 as it relates to CR-3 was addressed in Section 2.5.4.3, "Reactor Auxiliary Closed Cycle Cooling Water Systems." It was stated in this section that CR-3 implementation of the requirements of GL 96-06 was previously evaluated. Discuss how the previous evaluation regarding fluid contained in penetrations between containment isolation valves is affected (thermally induced overpressurization) and if any additional measures are required as a result of the EPU.

Response:

The impact of the CR-3 EPU on thermally induced overpressurization between the containment isolation valves was evaluated and conclusions discussed in Section 2.5.4.3 of the CR-3 EPU TR (Reference 1, Attachments 5 and 7); specifically page 2.5.4.3-6. The small increase in maximum containment temperature following a design basis event at EPU conditions is less than the value in the original GL 96-06 analysis. Additionally, there are no physical changes or operational changes required as a result of EPU operation that would affect the containment penetration piping or isolation valves associated with the Decay Heat Closed Cycle Cooling Water (DC) System and Nuclear Services Closed Cycle Cooling (SW) System (CR-3 Reactor Auxiliary Closed Cooling Water Systems). As a result, no new relief valves are required and the existing DC and SW System relief valves remain acceptable.

SCVB-1.7

Please discuss if the feedwater into and steam out of steam generator, and the steam generator metal in contact with secondary side fluid were considered when determining the sources of energy addition to containment on the mass and energy release analyses described in Section 2.6.3.2.

Response:

The M&E release analysis for the MSLB accident presented in the CR-3 EPU TR (Reference 1, Attachments 5 and 7) accounts for the energy release associated with the MFW System and Main Steam System response. The liquid inventory in the affected steam generator as well as the liquid inventory in the affected steam generator's feedwater lines downstream of the MFW block valves is included in the containment M&E release. The initial steam mass and its associated energy in the affected steam generator and affected steam generator steam line between the break and the steam line non-return check valve are also included in the containment M&E release. The stored energy transfer between the steam generator metal and the secondary side fluid is accounted for and calculated using the RELAP5/MOD2 B&W computer code (Reference 2).

Control Room Habitability and Ventilation Systems

SCVB-2.1

Section 2.7.3, "Ventilation Systems," Subsection 2.7.3.1.2 discusses the ability of the Control Room Area Ventilation System (CRAVS) to maintain a mild temperature environment for control room personnel and control room components. Specifically, CR-3 evaluated the safety-related portions of the CRAVS (Control Complex Ventilation System and Emergency Feedwater Initiation and Control System). It was stated that the "heat load increases for EPU are small." Please provide a summary of the equipment changes in the control room that have an impact on heat loads, however small the impact may be. Specify if the heat load evaluations performed are qualitative or quantitative, and if qualitative, provide a basis for your conclusion.

Response:

As a result of operation at EPU conditions, equipment changes in the Control Complex have been identified that potentially impact the CRAVS; Control Complex ventilation subsystem and the EFIC ventilation subsystem. The Control Complex heat load change has been qualitatively evaluated based on expected loading from the proposed EPU modifications and the additional heat load is within the available margin of the Control Complex and EFIC ventilation subsystems. Specifically, the expected additional heat load has been evaluated considering the current heat load data in the applicable models of the Control Complex temperature calculations. As part of the CR-3 Engineering Change (EC) process, each EC package provides the calculated additional heat loads which are added to the heat load data in the applicable Control Complex temperature calculations.

The most significant increase in Control Complex heat load originates from equipment changes associated with the addition of the Inadequate Core Cooling Mitigation System (ICCMS) and the Fast Cooldown System (FCS). The following lists the proposed ICCMS and FCS equipment and includes the projected electrical and heat loads:

Location	Description	BTU/hr
"A" 4160V ES Switchgear Room	ICCMS Cabinet 1	1024
"B" 4160V ES Switchgear Room	ICCMS Cabinet 2	1024
"B" 480V Switchgear Room	ICCMS Cabinet 3 and Online Monitor	2047
124 ft Elevation Hallway	Two Uninterruptible Power Supplies	5115

Location	Description	BTU/hr
Main Control Board Lights	Status Lights ~31 lights	109
Battery Charger Room A	Battery Chargers Control Panel: DC-DC converter, voltage relays, and diodes	196
Battery Charger Room B	Battery Chargers Control Panel: DC-DC converter, voltage relays, and diodes	196
EFIC Room A	FCS Panel: Analog isolator, isolation relay (3); and burden resistors (3) FCS Pressure Controller Cabinet: Pressure controller	108
EFIC Room B	FCS Panel: Analog isolator, isolation relay (3); and burden resistors (3) FCS Pressure Controller Cabinet: Pressure controller	108
EFIC Room C	Analog Isolator Cabinet: Analog isolators (2)	123

A qualitative evaluation was performed to determine the individual room temperatures with the additional heat loads listed above. The room temperatures increase less than 1°F and remain below the maximum allowable post-LOCA Control Complex temperatures. Also, the additional BTUs were compared to the existing margins in the Station Blackout (SBO) analyses and the additional heat loads as a result of EPU operation and associated modifications were qualitatively determined to have a small impact on these margins.

During finalization of the EC packages associated with the EPU modifications, more precise heat load values will be added to the heat load data in the Control Complex models and associated calculations revised to confirm that operation at EPU conditions has no adverse effect on the ability of the CRAVS to provide a controlled environment for the comfort and safety of control room personnel and to support the Operability of Control Complex components.

SCVB-2.2

It is stated in Section 2.7.4, "Spent Fuel Pool Area Ventilation System," Subsection 2.7.4.2 that the air temperature in the spent fuel pool area is affected by heat released from the spent fuel pool. However, it is not clear how the heat load increase to the ventilation system is considered in the EPU evaluations. It was also stated in Subsection 2.7.5.2 that a very large temperature range (55 °F to 122 °F) is acceptable within the fuel handling area. Discuss the present systems margin in maintaining this temperature range, and how the additional heat load due to the EPU is evaluated to be within the margin.

Response:

As described in Section 2.5.4.1, "Spent Fuel Pool Cooling and Cleanup System," of the CR-3 EPU TR (Reference 1, Attachments 5 and 7), cooling for the spent fuel pool is provided by the

Spent Fuel Cooling and Cleanup System. This system was evaluated and FPC concludes that the Spent Fuel Pool Cooling and Cleanup System continues to be capable of providing sufficient cooling to cool the spent fuel pool following implementation of the proposed EPU and has the heat removal capacity to ensure the spent fuel pool temperature remains below the design value of 160°F.

As described in Section 9.7 of the CR-3 FSAR, supply and exhaust fans are provided to maintain the fuel handling area within the design temperature range. In addition, spent fuel pit supply fans provide internal air to sweep across the spent fuel pools and cask loading area to assist in ambient heat removal. Heating is provided if required. The design of the Spent Fuel Pool Area Ventilation Systems is to maintain the ambient air temperature between 55°F and 122°F.

Currently, there are no specific heat load calculations at CR-3 to quantify the overall ventilation effect and temperature margins in fuel handling and spent fuel pit areas. However, the highest daily maximum temperature for the geographical area of CR-3 (i.e., Tampa, Florida) is 99°F. Assuming a maximum outside air temperature of 100°F, there is a 22°F margin between supply air temperature and the maximum design temperature of 122°F for the spent fuel pool area. Since spent fuel pool water temperature will continue to be maintained below 160°F during operation at EPU condition, it is reasonable to conclude that the Spent Fuel Pool Area Ventilation System temperature range and existing margins will not be significantly altered as a result of EPU operation.

SCVB-2.3

Section 2.7.5, "Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems," Subsection 2.7.5.2 discusses significant plant modifications that have a potential to significantly add to the heat load in the Turbine Building. Please explain if any calculations were performed to quantify the overall effect of the heat load additions on the Turbine Areas Ventilation System and the conclusions of the calculations.

Response:

The Turbine Building Ventilation System is designed assuming an outside air temperature range between 25°F and 95°F with a maximum design flow rate of 521,500 cfm with all the supply and exhaust fans running. Based on this temperature range, the system is capable of limiting the turbine hall temperature rise to 20°F above outside ambient temperature.

Currently, there are no specific heat load calculations at CR-3 to quantify the overall ventilation effect in the main turbine building. As stated in Section 2.7.5 of the CR-3 EPU TR (Reference 1, Attachments 5 and 7), the turbine building sampling room, turbine building switchgear rooms, and turbine building non-Class 1E battery room are cooled by individual package air handling units. FPC has qualitatively determined that any minor heat load increase in these areas as a result of EPU does not adversely impact the capability of the individual air handling units to maintain the temperature in the associated area.

FPC has also qualitatively determined that any minor heat load increase in the main area of the turbine building as a result of EPU does not adversely impact the design capability of the

Turbine Building Ventilation System of limiting the turbine hall temperature rise to 20°F above outside ambient temperature.

SCVB-2.4

Section 2.7.7, "Reactor Building Ventilation Systems," Subsection 2.7.7.2 discusses the Reactor Building Recirculation System's function to control containment temperature via the Industrial Cooling System (CI). The licensee further states that CR-3 has developed procedures to shift from CI to two trains of Nuclear Services Closed Cooling Water (SW) for cooling during the challenging summer month periods. However, it is not clear if this shifting procedure is part of the current licensing basis, or if it will be new due to the EPU. *Please provide additional details about the procedures. For instance, is the shifting automatic or manual? How does it affect the containment isolation function of these systems?*

Response:

The CI System is the normal source of cooling water for the RB Recirculation System during normal plant operation. The SW System is used to cool the RB during a LOCA. During normal plant operation, the cooling supply to the RB cooling units may be manually transferred from the CI System to the SW System, as necessary. In addition, upon activation of an Engineered Safeguards Actuation System signal, the cooling supply to the RB cooling units automatically transfers from the CI System to the SW System. These current licensing and design basis modes of operation are further described in Section 9.7, "Plant Ventilation Systems," of the CR-3 FSAR and Improved Technical Specification Bases 3.7.7, "Nuclear Services Closed Cycle Cooling Water System (SW)."

Based on UFSAR Section 9.7.2.1, the CI System provides chilled water to the RB recirculation system coolers. Normally, you would expect chilled water to be at a lower temperature than SW. Please explain how additional cooling is achieved through these coolers by shifting from chilled water to SW.

Response:

The heat transfer capability of the SW System is greater than the CI System due to the difference in system flow rates. The CI System flow is 700 gpm per fan assembly and the SW System provides 1780 gpm per fan assembly to the RB cooling units.

In Subsection 2.7.7.2, the licensee discusses the increased load in the RB. The licensee stated, "The ΔT across the hot-leg insulation increases by 6.4 °F (~1.5%). Since the ΔT across the pressurizer insulation is unchanged and the ΔT across the cold-leg insulation is actually decreasing, the total heat loss from the RCS [Reactor Coolant System] will increase by less than 1.5%." The NRC staff would like to know if the additional heat loads in the RB also include the EPU related increases in the steam generator heat loads and the control rod drive (CRD) mechanism heat loads.

Response:

The RB Steam Generator Cooling System consists of fans that disperse the local heated air around the steam generators to the overall RB air space and the heat is removed by the RB Recirculation System. This additional steam generator heat load is included in the EPU analyses of the RB Ventilation Systems. In addition, heat generated from the CRD mechanisms (CRDMs) is not expected to increase as a result of operation at EPU conditions. Therefore, there is no additional CRDM heat load impact on the RB Ventilation Systems.

In the same subsection, the licensee discusses the temperature limit for the CRD shroud, which is 150 °F. Based on this limit, the electrical connector and CRD position indicator enclosures located in the service structure has been challenged during the summer months. It is further stated that qualified component lifetime is trended with the cumulative impact monitored and preventive maintenance actions implemented as appropriate. Is this presently done, or will this be initiated as a result of the EPU? Are these components covered under the Equipment Qualification program?

Response:

Currently, the qualified lifetime of the CRDM electrical components are trended, the cumulative impact monitored, and preventive maintenance actions implemented as appropriate. These actions will continue during EPU operation. Also, the CRDM electrical components are not included in the CR-3 Equipment Qualification Program due to the fast response time required during postulated accidents and transients.

SCVB-2.5

Section 2.3.5, "Station Blackout," Subsection 2.3.5.2 concludes that the EPU will not affect the ability to fulfill the requirements of CR-3's Ventilation system during a station blackout event. It is stated in this section that the temperatures have been evaluated for the added EPU heat loads and found acceptable. Please provide the details of the evaluations performed, and compare the results with the pre-EPU conditions.

Response:

A qualitative assessment of the EPU impact on the SBO heat load calculations was performed and the additional BTUs were compared to the existing margins in the SBO analyses. The additional heat loads, as a result of EPU operation and associated modifications, were qualitatively determined to have a small impact on these margins. Specifically, the EFW pump recirculation line modification is conservatively estimated to add less than 250 watts to the applicable SBO areas of concern. This additional heat loading is less than 7% of the available margin in the limiting area of concern; Class 1E 480 V Switchgear Room B.

During finalization of the EC packages associated with the EPU modifications, more precise heat load values will be included in the SBO calculations to assure that maximum area temperatures are not exceeded following an SBO at EPU conditions.

ECCS Pump Net Positive Suction Head

The issue of crediting containment accident pressure (CAP) to assure adequate net positive suction head (NPSH) to the ECCS and containment heat removal pumps was given considerable attention recently by the NRC. The NRC staff acknowledges the licensee's claim in Section 2.6.5.1 that adequate NPSH margin is maintained for the low-pressure injection (LPI) and building spray (BS) pumps. However, based on new guidance on NPSH margin applicable to EPU reviews, including CR-3, the NRC staff needs to determine whether use of CAP could become necessary for plants requesting EPU, with or without uncertainties included in the calculations. Also, the maximum erosion zone (defined in the guidance document) needs to be addressed. The following are some recent documents from the NRC that led to the application of new guidance to EPU applications.

- Letter from NRC to Pressurized-Water Reactor Owners' Group [PWROG], "The Use of Containment Accident Pressure in Demonstrating Acceptable Operation of Emergency Core Cooling and Containment Heat Removal Pumps during Postulated Accidents," dated March 24, 2010 (ADAMS Accession No. ML100740579).

- NRC Commission Paper, SECY-11-0014, "Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents," dated January 31, 2011 (ADAMS Accession No. ML102780586).

- NRC Staff Requirements Memorandum, "Staff Requirements – SECY-11-0014 – Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents," dated March 15, 2011 (ADAMS Accession No. ML110740254).

In order to make an informed decision as to whether the licensee is in effect utilizing or not utilizing CAP when the new guidance is applied to NPSH evaluations and determine if the evaluations are in accordance with the guidance, the NRC staff requires additional information.

SCVB-3.1

Provide the basis for the NPSH required (NPSHR) of the high-pressure injection, LPI and BS pumps (tested value, extrapolation to flows other than tested flows), including flow rates assumed, and a comparison with the flow rate for the LOCA peak cladding temperature analyses. What head drop value is used for NPSHR (3% head drop or other)?

Response:

The following provides the bases for the NPSH required (NPSHr) for the CR-3 ECCS and RB Spray (BS) System pumps to assure adequate core cooling during a LOCA at EPU conditions:

High Pressure Injection (HPI) System

The highest HPI System flow following a LOCA at EPU conditions is a function of RCS pressure and is considered when determining pump NSPHr. The following provides the highest

HPI System flows in the CR-3 EPU LOCA analyses to ensure fuel peak cladding temperature (PCT) is within 10 CFR 50.46 criteria. These flow values are based on the lowest analyzed RCS pressure to maximize pump flow in the NSPH analyses:

ECCS	Flow (gpm)
Two HPI pumps without FCS	670 (335 per pump)
One HPI pump with FCS	502

The NPSHr for each HPI pump with suction from the borated water storage tank (BWST) is 33.6 ft at 600 gpm and considers a total dynamic head (TDH) reduction of 3%. The NPSHr was extrapolated from the most conservative tests and is conservatively adjusted for EDG frequency variations. The following table shows the NPSH values for each HPI pump (MUP-1A, MUP-1B, and MUP-1C) using test points from pump flow testing:

HPI Pump	Test Flow (gpm)	NPSHr (ft)
MUP-1A[a]	200	23
	300	24.5
	400	26
	500	28
MUP-1B[b]	300	23
	500	28
MUP-1C[a]	300	24
	500	28

a NPSH values are based on actual pump test data.
b NPSH values are based on manufacturer pump curve.

Also, the long-term NPSHr for each HPI pump when aligned for RB sump recirculation (LPI-HPI piggyback) mode of operation is 100 ft, which is conservatively based on the pump manufacturer recommended NPSHr beyond 72 hours following a LOCA specified in the mission time qualification report.

Low Pressure Injection (LPI) System

The following provides the highest LPI System flows in the CR-3 EPU LOCA analyses with suction from the BWST to ensure fuel PCT is within 10 CFR 50.46 criteria. These flow values are based on the lowest analyzed RCS pressure to maximize the required flow in the NSPH analyses.

Type of LOCA	Flow (gpm)
Design Basis Large Break	2685
Core Flood Tank Line Break	1435[a]

Type of LOCA	Flow (gpm)
RCS Cold Leg Pump Discharge Break	2886[a]

a Refer to Section 2.8.5.6.3 of the CR-3 EPU Technical Report.

The following provides the highest LPI System flows in the CR-3 LOCA analyses with suction from the RB sump to ensure fuel PCT is within 10 CFR 50.46 criteria. These flow values are based on an RCS pressure of 0 psig and reduce over time.

Time following a LOCA	Flow (gpm)
24 minutes	1000
2 hours	800
12 hours	700

The NPSHr for the LPI pumps with suction from the BWST is 19.3 ft at 3597 gpm and with suction from the RB sump is 12.57 ft at 2992 gpm and are conservatively adjusted for EDG frequency variations. These NPSHr values consider a TDH reduction of 3% and include mini-recirculation line flow rates. The NPSHr with suction from the BWST was determined by extrapolating from LPI pump flow testing and the NPSHr with suction from the RB sump is very near the actual test point from Decay Heat Pump (DHP)-1A. The following table shows the NPSH values for each LPI pump (DHP-1A and DHP-1B) using test points from pump flow testing:

LPI Pump	Test Flow (gpm)	NPSHr (ft)
DHP-1A[a]	518	13.7
	1001	9.7
	2002	8.2
	3032	12.5
DHP-1B[b]	3597	19.3

a Pump flows except 3032 gpm are based on Three Mile Island-1 (TMI-1) LPI pump impeller testing. 3032 gpm is based on DHP-1A impeller.

b DHP-1B pump NPSHr curve is extrapolated from TMI-1 LPI pump and DHP-1A test data.

RB Spray (BS) System

Following a large break LOCA at EPU conditions, the containment response analyses assume a BS System flow of 1000 gpm with suction from the BWST and 1200 gpm with suction from the RB sump.

The BS pump NPSHr is 14.4 ft at 1614 gpm with suction from the BWST and 12.9 ft at 1362 gpm with suction from the RB sump and; considers a TDH reduction of 5%, and are conservatively adjusted for EDG frequency variations. The 5% TDH reduction is the manufacturer approved reduction.

CR-3 tested four impellers associated with BS Pump (BSP) 1A (2 installed and 2 spares) between 1000 and 2000 gpm. The BS pump NPSHr is based on the BS pump flow testing results which indicated Impeller #2 installed in BSP-1A had the highest NPSHr. BSP-1B conservatively uses the BSP-1A NPSHr curve.

SCVB-3.2

Provide details of the method of calculating NPSH available (NPSHA) for all the above pumps (e.g., Refueling Water Storage Tank (RWST) level, containment atmospheric pressure, vapor pressure, head loss through suction piping, sump water temperature).

Response:

CR-3 does not have an RWST; rather, the ECCS and BS pumps take suction from the BWST. The following equation was used for calculating NPSHa for the ECCS and BS pumps with suction from the BWST:

$$\mathrm{NPSHa} = \mathrm{H_{STATIC}} + \mathrm{H_{ATMOS}} - \mathrm{H_{VP}} - \mathrm{H_{LOSS}}$$

Where:

- H_{STATIC}: Static head based on a minimum BWST level of 5.5 feet.
- H_{ATMOS}: Atmospheric head less one foot of vacuum from the BWST vacuum breaker.
- H_{VP}: Vapor pressure based on a BWST water temperature of 100°F.
- H_{LOSS}: Head loss from BWST suction piping. Includes: (1) flow rate of 5498 gpm assuming three pumps (HPI, LPI, and BS pumps) per train; (2) 325 gpm tank recirculation mixing pre-LOCA; and (3) 8545 gpm conservative gravity drain rate to the RB sump during pump suction swap for LPI and BS pump NPSHa.

The following equation was used for calculating NPSHa for the ECCS and BS pumps with suction from the RB sump:

$$\mathrm{NPSHa} = \mathrm{H_{STATIC}} - \mathrm{H_{LOSS}}$$

Where:

- H_{STATIC}: Static head based on a minimum containment water level of 97.1 ft elevation; 2.1 ft above the RB floor.

- H_{LOSS}: Head loss from RB sump strainer and suction piping. Includes a flow rate of 4254 gpm assuming three pumps (HPI, LPI, and BS pumps) per train; total 8508 gpm through the sump strainer.

H_{ATMOS} and H_{VP} are not included in the NPSHa equation when aligned to the RB sump since H_{ATMOS} is greater than or equal to H_{VP} and cancel when RB sump temperature is 204.7°F; H_{ATMOS} is based on the minimum allowable pre-accident containment pressure of 12.7 psia and H_{VP} is based on a saturation pressure of 12.7 psia.

The LPI and BS pump NPSH analyses utilize a conservatively low RB sump water temperature of 204.7°F based on a saturation temperature (T_{SAT}) at 12.7 psia to maximize suction piping friction losses due to the higher viscosity.

SCVB-3.3

Provide the results in a tabular form for both the injection phase and recirculation phase. As a minimum, include the flow rates, static head at minimum levels (RWST or sump), head loss through suction piping, containment atmosphere pressure, vapor pressure, water temperature, NPSHA, NPSHR, NPSH margin and friction losses.

Response:

An enclosure to this correspondence, "Summary of ECCS and BS Pump NPSH Analyses," provides a summary of the ECCS and BS pump NPSH evaluations performed for CR-3 considering EPU conditions. This tabular presentation of the injection and recirculation phases following a LOCA include pump flow rates, BWST and RB sump static head at minimum levels, head loss through suction piping including friction losses, RB pressure, vapor pressure, water temperature, NPSHa, NPSHr, and NPSH margin.

SCVB-3.4

Please demonstrate that NPSH margin still exists after including the uncertainties in the required NPSH. The NRC staff, in consultation with a pump expert, determined that a 21-percent margin on the "3%-required NPSH" would envelope the uncertainties in the draft guidance document. It is acceptable to the NRC staff, if the EPU applicants desire, to use this value in lieu of performing detailed plant specific uncertainty evaluations.

Response:

The NPSH analyses for the ECCS and BS pumps considering operation at EPU conditions use multiple conservative assumptions to determine NPSHa and NPSHr. These conservative assumptions assure adequate ECCS and BS pump NSPH margin following a LOCA at EPU conditions and obviates the need for additional uncertainty evaluations.

For the CR-3 EPU, the predicted peak RB sump water temperature is 262.9°F approximately 2 hours post-LOCA during the recirculation phase. The corresponding P_{SAT}, based on containment vapor pressure, is approximately 37.2 psia (22.5 psig). The predicted CAP following ECCS transition to recirculation phase is approximately 57 psia (42.3 psig). CR-3 ECCS and BS pump

NPSH analyses do not include the pressure contributions from the non-condensable gasses. This is considered conservative since relieving containment pressure requires a breach beyond design basis leakage acceptance criteria. The pressure contribution from the water vapor is credited at CR-3 when determining the minimum P_{SAT} of 12.7 psia (–2 psig) and the maximum P_{SAT} of 37.2 psia (22.5 psig) used in the NPSH analyses. The NPSH margin for the ECCS and BS pumps is based on the coldest T_{SAT} that could be present post-LOCA; a RB pressure of 12.7 psia (–2 psig) and T_{SAT} of 204.7°F. Using a T_{SAT} value of 204.7°F increases frictional losses due to the viscous effect; resulting in a conservative suction piping saturated water head loss. NPSH margin is further improved with RB sump temperature less than 204.7°F due to the subcooling effect dominating the viscosity effect.

The calculated NPSH margin of the ECCS and BS pumps with suction from the RB sump is determined from the RB sump H_{STATIC} less the suction piping frictional losses (H_{LOSS}) since the RB H_{ATMOS} and H_{VP} terms cancel. The H_{STATIC} used in the NPSHa analysis is the lowest predicted RB sump water level of 97.1 ft elevation (2.1 ft from the RB floor) and includes large volumes of water/vapor hold-up elsewhere in the RB. CR-3 uses the lowest credible RB sump water level coincident with the highest predicted pump flow rates, including instrument uncertainty, to calculate NPSH margin of the ECCS and BS pumps following a LOCA at EPU conditions.

The calculated NPSH margin of the HPI pumps when taking suction from the BWST is indicated as 8%. This calculated margin is based on a minimum BWST level of 5.5 feet, which includes instrument uncertainty, and conservatively assumes a BWST gravity drain rate. Plant emergency operating procedures require aligning the HPI System to LPI-HPI piggyback mode at a BWST level of approximately 15 feet and prior to opening the RB sump recirculation valves. The calculated HPI pump NPSH margin with a BWST level of 15 feet, including the conservative BWST gravity drain rate assumption, is 36.3%. An additional, a NPSH margin assessment has been performed considering an HPI pump suction line head loss at a BWST level of 5.5 feet without the conservative assumption of gravity drain rate to the RB sump. As indicated in the enclosure to this correspondence, "Summary of ECCS and BS Pump NPSH Analyses," the NPSH margin of the HPI pumps when taking suction from the BWST is indicated as 22.9% when the gravity drain rate is not included.

Also as shown in the enclosure to this correspondence, the NPSH margin of the ECCS and BS pumps range from 17.2% to 170% when taking suction from the RB sump. The NPSH margin of DHP-1B is 17.2% and the NPSH margin of BSP-1B is 18.4%, which are below the NRC staff recommendation of 21%.

As noted in the enclosure, the NPSH margin of BSP-1B is 34% when considering the pump-specific NPSHr curve instead of the most conservative NPSHr curve.

Regarding the NPSH margin of the DHP-1B, several individual considerations increase NPSH margin:

- When instrument uncertainty is determined using the Square Root of the Sum of the Squares method, the NPSH margin increases to approximately 19%.

- One foot of NPSHa recovery increases the NPSH margin of DHP-1B to approximately 26%. When aligned to the RB sump, one foot of NPSHa recovery to the B train is possible with either: a higher RB water level; RB sump temperature recovery; or due to termination of MUP-1B or BSP-1B.

- Significant NPSH margin is recovered when the RB sump water temperature decreases below 207.4°F. For example: 4 feet of NPSH is recovered at 200°F, and 12 feet at 180°F. Analyses indicate RB sump water temperature decreases below 180°F within 30 hours following a LOCA at EPU conditions.

- Also, 0.7 feet of NPSHr recovery is possible by crediting the high temperature effect on reducing NPSHr. This effect has not been credited in the EPU NPSH analyses and if credited would increase the NPSH margin of DHP-1B to 23%.

Considering multiple conservative assumptions in the EPU NPSH analyses, FPC has determined that the ECCS and BS pumps continue to have adequate NPSH margin to ensure post-LOCA recovery following an event at EPU conditions.

SCVB-3.5

Provide a discussion of how the post-accident debris generation at CR-3 is impacted by the EPU and the resultant impact on the sump strainer head loss and on the pump NPSH evaluations.

Response:

CR-3 post-accident debris generation is evaluated using the methodology prescribed by Nuclear Energy Institute Report NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," (Reference 3). This methodology uses Zone of Influences (ZOIs) specific to target type (insulation, coating, etc.) to determine the destruction susceptibility of selected debris targets from high energy line break steam jet impacts. There is a slight increase in stored energy due to the slightly higher RCS temperature as a result of operation at EPU conditions; however, the ZOIs are generic and do not change and the debris targets are not altered as a result of EPU operation or associated EPU modifications. As a result, additional debris types and debris quantity are not generated as a result of operation at EPU conditions.

Debris transport is a function of the cumulative ECCS and BS pump recirculation flow rate to the RB sump; i.e., transport stream velocity. The long-term sump recirculation flow rate assumptions in the CR-3 debris transportation calculation remain conservative because the RB sump transport stream velocity is not altered by operation at EPU conditions.

Also, the debris transportation calculation assumes an RB sump water temperature of 215°F. Sensitivity analyses determined that debris transport results are not significantly affected by changes in pool water temperature ranging from 100°F to 250°F. Therefore, it is reasonable to conclude that the short-term increase in RB sump water temperature (243°F to 263°F) following a LOCA at EPU conditions does not affect existing debris transport results.

To summarize; since the debris generation and debris transport conditions following a LOCA at EPU conditions are bounded by the existing analyses, there are no adverse effects created for either the RB sump strainer head loss determinations or the ECCS and BS pump NPSH evaluations.

References

1. 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)

2. AREVA NP Topical Report BAW-10164P-A, Revision 6, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

3. Nuclear Energy Institute Report NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, December, 2004.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 /LICENSE NUMBER DPR-72

ENCLOSURE

SUMMARY OF ECCS AND BS PUMP NPSH ANALYSES

Summary of ECCS and BS Pump NPSH Analyses

Injection Phase (BWST)

Pump (suction elev)	CR-3 Eqpt ID	BWST Level (feet)	Pump Flow Rate (gpm)	BWST Static Head (feet)	Suction Head Loss (feet)	BWST Pressure (psia)	Vapor Pressure (psi)	Water Temp (^{o}F)	NPSHa (feet)	NPSHr (feet)	NPSH margin (feet)	NPSH margin (%)
HPI (97.2')	**MUP-1A** (conservative)	5.5	600	27.3	21.7	14.26	0.95	100	36.3	33.6	2.7	8.0%
(transfer to piggyback)		15		36.8					45.8	33.6	12.2	36.3%
	(no gravity drain to RB sump)	5.5		27.3	16.7 est				41.3	33.6	7.7	22.9%
	MUP-1B (conservative)	5.5	600	27.3	21.7	14.26	0.95	100	36.3	33.6	2.7	8.0%
(transfer to piggyback)		15		36.8					45.8	33.6	12.2	36.3%
	(no gravity drain to RB sump)	5.5		27.3	16.7 est				41.3	33.6	7.7	22.9%
	MUP-1C (conservative)	5.5	600	27.3	21.7	14.26	0.95	100	36.3	33.6	2.7	8.0%
(transfer to piggyback)		15		36.8					45.8	33.6	12.2	36.3%
	(no gravity drain to RB sump)	5.5		27.3	16.7 est				41.3	33.6	7.7	22.9%
LPI (77.6')	**DHP-1A** (current)	5.5	3404	46.9	10.5	14.26	0.95	100	67.2	16.5	50.7	307.3%
(BWST 124.5')	(EPU flowrate, including recirc)		3597		11.7 est				66.0	19.3	46.7	242.0%
	DHP-1B (current)	5.5	3404	46.9	7.9	14.26	0.95	100	69.8	16.5	53.3	323.0%
	(EPU flowrate, including recirc)		3597		8.5 est				68.4	19.3	49.1	254.4%
BS (77.3')	**BSP-1A** (current)	5.5	1614	47.2	8.0	14.26	0.95	100	69.9	14.4	55.5	385.4%
(BWST 124.5')	(EPU fric. losses)				9 est				68.9	14.4	54.5	378.5%
	BSP-1B (current)	5.5	1614	47.2	6.8	14.26	0.95	100	71.1	14.4	56.7	393.8%
	(EPU fric. losses)				7.6 est				70.3	14.4	55.9	388.2%

Recirculation Phase (RB Sump)

Pump (suction elev)	CR-3 Eqpt ID	RB Level (feet)	Pump Flow Rate (gpm)	RB Pool Static Head (feet)	Suction Head Loss (feet)	RB Pressure (psig)	Vapor Pressure (psi)	Water Temp (^{o}F)	NPSHa (feet)	NPSHr (feet)	NPSH margin (feet)	NPSH margin (%)
HPI (97.2')	**MUP-1A**	2.1	580	350	54	-2.0	10.4	195	270	100	170	170%
(aligned to piggyback)			(<72 hrs)	(LPI head)								
	MUP-1B	2.1	580	350	54	-2.0	10.4	195	270	100	170	170%
			(<72 hrs)	(LPI head)								
	MUP-1C	2.1	580	350	54	-2.0	10.4	195	270	100	170	170%
			(<72 hrs)	(LPI head)								
LPI (77.6')	**DHP-1A** (includes recirc)	2.1	2992	19.5	4.05	-2.0	12.7	204.7	15.45	12.57	2.88	23%
RB Sump (97.12')												
	DHP-1B (includes recirc)	2.1	2992	19.5	4.77	-2.0	12.7	204.7	14.73	12.57	2.16	17.2%
	(SRSS msmt error)				4.53				14.97		2.40	19.1%
BS (77.3')	**BSP-1A**	2.1	1362	19.8	3.97	-2.0	12.7	204.7	15.83	12.91	2.92	23%
RB Sump (97.12')												
	BSP-1B	2.1	1362	19.8	4.51	-2.0	12.7	204.7	15.29	12.91	2.38	18.4%
	(actual NPSHr)								15.29	11.4	3.89	34%