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

Ref: 10 CFR 50.90

September 13, 2007
3F0907-06

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

Subject: Crystal River Unit 3 – License Amendment Request (LAR) #296, Revision 1: Measurement Uncertainty Recapture Uprate Response to Request for Additional Information (TAC No. MD5500)

Reference: FPC to NRC letter, 3F0607-05, LAR #296, Revision 1, Measurement Uncertainty Recapture Uprate, dated June 28, 2007

Dear Sir:

On July 9, July 20, and August 7, 2007, the Nuclear Regulatory Commission (NRC) issued electronic Requests for Additional Information (RAIs) regarding License Amendment Request #296, Revision 1, referenced above. In accordance with the provisions of 10 CFR 50.90, Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc., hereby provides the response to the RAIs.

This letter establishes no new regulatory commitments.

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

Sincerely,

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

DEY/par

Attachments:

A. RAI Response – Steam Generator Tube Integrity and Chemical Engineering Branch (CSGB)
B. RAI Response – Fire Protection Branch
C. RAI Response – Operator Licensing and Human Performance Branch (IOLB)
D. RAI Response – Electrical Engineering Branch (EEEB)
E. RAI Response – Capacity Increase Stability Study
F. RAI Response – Vessels and Internals Integrity Branch (CVIB)

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

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

A001
NRR

STATE OF FLORIDA

COUNTY OF CITRUS

Dale E. Young 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.

Dale E. Young

Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 13th day of Sept., 2007, by Dale E. Young.

Ellen Deppolder

Signature of Notary Public
State of Florida

ELLEN DEPPOLDER
MY COMMISSION # DD 408539
EXPIRES: July 8, 2009
Bonded Thru Notary Public Underwriters

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

Personally Known ✓ -OR- Produced Identification ______

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #296, REVISION 1
ATTACHMENT A

RAI RESPONSE –
STEAM GENERATOR TUBE INTEGRITY AND CHEMICAL ENGINEERING BRANCH (CSGB)

Request for Additional Information Response

On July 9, 2007, Florida Power Corporation (FPC) received an electronic Request for Additional Information (RAI) concerning License Amendment Request (LAR) #296, Revision 1 via email. By letter dated August 7, 2007, this RAI was received by FPC. FPC hereby provides the responses to the RAI that was received from the Steam Generator Tube Integrity and Chemical Engineering Branch (CSGB).

NRC Request CSGB-1

1. **On Page 34 of Attachment D, you indicated that "a review of calculations performed which assessed the integrity of tubes containing flaws of various types when subjected to operating and accident loads was conducted." In addition, you indicated "this review ensured that existing structural margins are maintained for the MUR Power Uprate Program design conditions." This wording is unclear; therefore, confirm that this review did ensure that all existing structural margins are maintained for the power uprate. In addition, please discuss how the various flaw types for SG tubes (existing and potential) were affected by the MUR power uprate.**

FPC Response 1

The review of calculations confirmed that existing structural margins are maintained for the Measurement Uncertainty Recapture (MUR) Power Uprate Program design conditions. Since the pre-MUR design conditions were not exceeded by the MUR power uprate, it was not necessary to re-review flaw experience. All existing analyses bound MUR design conditions and flaw evaluations.

NRC Request CSGB-2

2. **In Section 4.2.5.2, Inservice Testing (IST) Program, you indicate "...that the MUR uprate is bounded by current analysis and any changes are insignificant." Please discuss the possible "insignificant changes" that may be made to the IST Program and what makes these changes "insignificant."**

FPC Response 2

The Inservice Testing (IST) Program itself will not be impacted by MUR. No component will be added or removed from the program and no programmatic requirements will be changed. The operating conditions (pressure, temperature, flow, etc.) for the associated Systems, Structures, and Components (SSCs) are slightly increased. The increase was evaluated as part of both program and system reviews. The impacts were appropriately dispositioned and do not require system or component modifications. Some documents may require revision to reflect the new conditions. For example, some valves in the IST program may have slight changes to their stroke times due to these changed conditions. Given the relatively small shift in operating conditions, revisions to IST stroke-times are not anticipated; however, if a deviation in stroke-time occurs, the need for rebaselining will be evaluated on a case by case basis.

No significant changes are predicted in that all components will continue to be capable of meeting accident analysis assumptions without modification. Small changes to acceptance criteria for IST testing are possible due to parameter changes. These revised criteria are expected to fall within existing design criteria.

NRC Request CSGB-3

3 **Confirm that the steam generators (SGs) will continue to satisfy all original design criteria under the power uprate conditions. In addition, confirm that your analysis addresses the current condition of your SGs (e.g., plugs, tube repairs, loose parts, etc.) and addresses flow induced vibration. Also, provide confirmation that your SG tube plugging limit is still appropriate for power uprate conditions, given the guidance in Regulatory Guide 1.121, "Bases for Plugging Degraded PWR (Pressurized Water Reactor) Steam Generator Tubes."**

FPC Response 3

All post-MUR system parameters are within those evaluated in the existing Once Through Steam Generator (OTSG) analyses. Therefore, the OTSGs will continue to satisfy all original design criteria under MUR power uprate conditions.

An evaluation was performed to address flow induced vibration (FIV) implications associated with the MUR power uprate conditions and the OTSG tube bundle and installed tube repair hardware. The evaluation was based on previous FIV analyses of virgin tubes and tubes that are plugged and stabilized, considering all of the types of stabilizers that have been installed in the Crystal River Unit 3 (CR-3) OTSGs.

The FIV analyses for plugged, stabilized tubes assumed a complete sever at the limiting location based on the uprate thermal-hydraulic conditions. Scaling factors were determined based on the ratio of dynamic pressures at the nominal conditions of 2568 MWt with 0% plugging to the dynamic pressures at the power uprate conditions of 2609 MWt with 20% plugging. The scaling factors were determined at the worst-case locations of the tube bundle entrance and exit; however, the FIV evaluation conservatively applied the maximum scaling factor over the entire length of the tube.

The maximum scaling factor was applied to the worst case tubes from previous analyses for fluid-elastic instability, random turbulence-induced vibration, and vortex shedding-induced vibration. At the uprate conditions, the margin of safety against fluid-elastic instability is at least 30% for all tubes (Fluid-Elastic Stability Margin (FSM) > 1.3). The original design bases for the CR-3 OTSGs applied a minimum FSM of 1.0 for the fluid-elastic instability analysis. For random turbulence, the minimum percent ratio of the allowable displacement to the maximum tube displacement was 13.6 with the allowable displacement set to prevent tube-to-tube impacts over the life of the plant. The minimum margin against high cycle fatigue considering both in-service and stabilized tubes was greater than 117%. Therefore, the results of the evaluation show that the tube bundle in the CR-3 OTSGs will not fail due to the high cycle fatigue effects of flow-induced vibration resulting from turbulence due to cross flow conditions at the uprate conditions, nor will tube-to-tube impacts occur over the life of the plant. Similar margin was also obtained for vortex shedding-induced vibrations; (i.e., 23.8% for tube impacts and greater than 85% for high cycle fatigue).

The FIV evaluation reviewed concerns from NRC Bulletin 88-02 and NRC Information Notice 2002-02 in relation to the CR-3 power uprate. These concerns have been addressed at CR-3 or are covered by the current stabilization criteria.

OTSG repairs continue to be covered, after the MUR power uprate by the analysis in Topical Reports BAW-2346P, Revision 0, "Alternate Repair Criteria for Tube End Cracking in the Tube-to-Tubesheet Roll Joint of Once Through Steam Generators," BAW-2120P, "OTSG 80 Inch Mechanical Sleeve Qualification (Alloy 690)," and BAW-2303P, Revision 4, "OTSG Repair Reroll Qualification Report," as required by CR-3 ITS 5.6.2.10.

Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," continues to be met, and is referenced by CR-3 ITS Bases 3.4.16.

NRC Request CSGB-4

4. **Confirm that the coating qualification temperature and pressure profile used by CR-3 to originally qualify Service Level I coatings remains bounding in light of the power uprate pressures and temperatures. If the original coating qualification pressure and temperature profile is no longer bounding, discuss the conditions to be used and corrective actions that will be taken to assure that Service Level I containment coatings will be qualified.**

FPC Response 4

The post-accident temperature and pressure profile is a function of Reactor Coolant System (RCS) average temperature, which is not changing due to the MUR power uprate (T_{ave} remains 579°F). The peak Reactor Building (RB) temperature and pressure were performed at 102% of 2568 MWt (2619 MWt) and remains 276.8°F and 54.2 psig respectively after the MUR power uprate. Since post–accident temperature and pressure are not affected by the MUR power uprate, there is no impact on the RB coatings program.

NRC Request CSGB-5

5. **Please confirm the following regarding the SG blowdown system:**

   a. **That you considered whether the additional operating time due to the power uprate will result in system components to be more susceptible to flow accelerated corrosion (FAC).**
   b. **That your current evaluation of the SG blowdown system under power uprate conditions considered the effect of a potential increase of impurities in the SG water.**
   c. **That any change to the inlet pressure of the SG blowdown system is still within the range of original design operating parameters.**

FPC Response 5

Babcock and Wilcox (B&W) Nuclear Steam Supply System (NSSS) plants, including CR-3, do not require continuous steam generator blowdown. CR-3 does use steam generator blowdown to achieve secondary chemistry limits on restart from outages, which is typically terminated in

Mode 1 (approximately 20% RTP), well before reaching 100% RTP. Thus, increases in operating conditions at 100% RTP do not impact the steam generator blowdown system.

NRC Request CSGB-6

6. You indicated that "the predicted increases in maximum component wear rates and reductions in service lives will be managed by the CR-3 FAC program." Discuss how significant the increases in wear rates and reductions in service lives are for the power uprate conditions. In addition, discuss any changes made to CR-3's FAC program (i.e., criteria used for selecting components for inspection following the power uprate, criteria for repair and replacement, increased inspection scope, etc.) due to the power uprate conditions. Also, identify the systems that are expected to experience the greatest increase in wear as a result of the power uprate. Discuss whether inspections will be performed to assess wear prior to entering power uprate conditions.

FPC Response 6

A conservative Flow Accelerated Corrosion (FAC) evaluation was done to identify the limiting components. The most limiting component was determined to be the feedwater piping that feeds the feedwater ring header known as the feedwater risers. This piping has experienced the most limiting wear under current conditions and is the most likely to be impacted by MUR. The risers, and the plant in general, were modeled using MUR conditions as an input. Even under these conditions, the risers retained sufficient margin to support operation until Refueling Outage 16R (Fall 2009). The riser will be replaced as part of the OTSG replacement in 2009. The field conditions will be validated by in-service non-destructive evaluation testing in Refueling Outage 15R (Fall 2007) to ensure actual plant conditions are accurately reflected by the model.

CR-3 FAC analysis is updated on a continuous basis. The results of inspections performed in R15 will be incorporated into the FAC program. No other modifications to the FAC program are required for MUR uprate implementation. It is not expected that the MUR uprate will have any significant impact on piping or component wear rates or service lives due to the small increases in flow rate and temperature.

NRC Request CSGB-7

7. Provide confirmation that your evaluation for the chemical and volume control system demonstrates that the conditions for the power uprate are bounded by the original design conditions (thermal performance, letdown and makeup requirements, etc.).

FPC Response 7

The principal function of the makeup and letdown system is to provide a path for supplying makeup liquid to the RCS and a means for processing primary coolant, i.e., boron concentration and sampling. There are no required changes to the Makeup System configuration for the MUR. The nominal RCS pressure, RCS flow and average system temperature are not affected by the power uprate, therefore the amount of coolant required to offset temperature changes will not be affected. The boric acid content of the Borated Water Storage Tank (BWST) and Boric Acid

Addition Systems are verified in the core design process so that the required ability to add adequate amounts of negative reactivity are not impaired nor affected by the MUR power uprate. The hot leg temperature will increase by 0.4°F and cold leg temperature will decrease by 0.4°F. As a result, the letdown line, which is taken from the "D" cold leg will experience a slightly lower temperature as a result of the power uprate. Therefore, the letdown coolers are bounded by current operation and there is no adverse impact on the cooling function of the letdown coolers.

PROGRESS ENERGY FLORIDA, INC.

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LICENSE AMENDMENT REQUEST #296, REVISION 1

ATTACHMENT B

RAI RESPONSE –
FIRE PROTECTION BRANCH

Request for Additional Information Response

On July 20, 2007, Florida Power Corporation (FPC) received a Request for Additional Information (RAI) concerning License Amendment Request (LAR) #296, Revision 1 via email. Per a telephone call on August 15, 2007, it was determined that one response would satisfy the questions. FPC hereby provides the response to this RAI that was received from the Fire Protection Branch.

NRC Request Fire Protection 1

1. **License amendment request, Attachment D, Section 2.0 "Accidents and Transients for Which the Existing Analyses of Record Bound Plant Operation at the Proposed Uprated Power Level," mentions safe-shutdown fire analysis. However this section does not discuss the impact of measurement uncertainty recapture power uprate on the fire protection system(s). Clarify whether this request involves changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe-shutdown capability in accordance with Title10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix R. Provide the technical justification for whether and, if so, why, existing analyses bound any impact on accidents or transients resulting from any changes.**

NRC Request Fire Protection 2

2. **The staff notes that license amendment request, Attachment D, Section 3.0, "Accidents and Transients for Which the Existing Analyses of Record do not Bound Plant Operation at the Proposed Uprated Power Level," does not include any discussion regarding changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe shutdown capability in accordance with Appendix R. Clarify whether this request involves changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe-shutdown capability in accordance with 10 CFR Part 50, Appendix R. Provide the technical justification for whether and, if so, why, existing analyses do not bound any impact on accidents or transients resulting from any changes.**

NRC Request Fire Protection 3

3 **The U.S. Nuclear Regulatory Commission criteria which is applicable to the Crystal River Unit 3 Nuclear Generating Plant (CR-3) post-fire safe-shutdown is contained in Sections III.G and III.L of Appendix R to 10 CFR Part 50, in Generic Letter 81-12, "Fire Protection Rule (45 FR 76602, November 19, 1980)," and its subsequent clarification in Generic Letter 86-10, "Implementation of Fire Protection Requirements." The staff requests the licensee to verify with the increased reactor power level of 2609 MWt the safe-shutdown equipment for CR-3 would be in compliance with Appendix R.**

FPC Response

The 1.6 percent increase in power (from 2568 to 2609 MWt) resulting from the MUR uprate will slightly increase the natural circulation cooldown time. The time to cool the plant to 200°F will increase from 68.54 to 70.38 hours. This is still less than the 10 CFR 50 Appendix R requirement of 72 hours for natural circulation scenarios.

The additional heat in the Intermediate Building from the MUR uprate will not prevent required manual actions from occurring at their designated time. No other 10 CFR 50 Appendix R safe shutdown capability is impacted by the implementation of the MUR uprate.

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

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ATTACHMENT C

RAI RESPONSE – OPERATOR LICENSING AND HUMAN PERFORMANCE BRANCH (IOLB)

Request for Additional Information Response

On July 20, 2007, Florida Power Corporation (FPC) received a Request for Additional Information (RAI) concerning License Amendment Request (LAR) #296, Revision 1 via email. FPC hereby provides the responses to this RAI that was received from the Operator Licensing and Human Performance Branch (IOLB).

NRC Request 1

1. Operator Actions (RIS 2002-03 Section VII.1)

a. In its review of the existing operator actions and their available times, how did the licensee determine that they would not be impacted by the proposed MUR Power Uprate?

b. Have any available times for significant operator actions been reduced for certain accident scenarios and events, such as Anticipated Transients Without Scram (ATWS), due to the MUR Power Uprate? If so, describe how the new available times were validated.

FPC Response

1.a. The MUR has not impacted the time requirements for the Emergency Operating Procedure (EOP) operator actions. This was determined by considering each time critical action and determining the impact that MUR has on the action. For most actions, the analysis of record was performed at 102% of 2568 MWt and there are no changes to post accident dose projections. Other actions were not affected because the heat load was not significantly increased inside the control complex or other locations and because battery loading is unaffected by the MUR uprate. See Table C.1.

1.b. There are no specific EOP operator actions for ATWS. There are several actions associated with Station Blackout (SBO), however, the MUR has not impacted the time requirements for these EOP operator actions. This was determined by considering each time critical action and determining the impact that MUR has on the action. For most of the actions, the analysis of record was performed at 102% of 2568 MWt. Other actions were not affected because the heat load was not significantly increased inside the control complex or the battery loading was not affected by the MUR uprate.

NRC Request 2

2. Emergency and Abnormal Operating Procedures (RIS 2002-03 Section VII.2.A)

a. Describe any changes to operator actions in the Emergency Operating Procedures (EOPs) and Abnormal Operating Procedures (AOPs) required by the MUR Power Uprate and how these changes will be integrated into the operator training program.

b. Discuss if any additional operator actions are needed due to the MUR Power Uprate and if they will be reflected in the EOPs and AOPs

FPC Response

2.a. Several EOPs and Abnormal Operating Procedures (APs) are being changed to adjust percent power values to the maximum stable power level with the loss of the main feedwater pumps and the main feedwater booster pumps. For example, a trip of the Main Feedwater Booster Pump with four Reactor Coolant Pumps running requires the reactor power to be reduced to 52% power. With the MUR uprate, this limit will be approximately 50% power. No new EOP/AP operator actions will be added, and the changes to the EOPs/APs will be incorporated into the normal operator training cycles in a timely manner during implementation of the MUR uprate.

2. b. No new operator actions will be added to the EOPs and APs.

NRC Request 3

3. Control Room Controls, Displays (Including the Safety Parameter Display System), and Alarms (RIS 2002-03 Section VII.2.B)

a. How much margin will the operators have to maintain the increased Power Production Heat Transfer within the prescribed parameters?

b. Will any control room indicators be modified to account for any operational changes after MUR Power Uprate implementation?

c. How has the licensee verified that the proposed software changes to the Fixed Incore Monitoring System will not have any adverse effects on the operators? Are there potential software changes that could introduce false indications in the control room indicators? If so, describe how the operators will be able to recognize and address the erroneous indications.

d. Describe how the operators will be notified of self-diagnosed errors with the Caldon Leading Edge Flowmeter (LEFM) in the event that the new control room alarm is either inoperable or fails to alert the operators.

FPC Response.

3.a. CR-3 re-reviewed the referenced guidance and our previous response. The margin between operation at rated thermal power (RTP) and the Reactor Protection System (RPS) Trip Setpoints with the Leading Edge Flow Meter (LEFM) in service is not reduced. The setpoint is the same as it currently is (104.9% RTP) for the Nuclear Overpower – High Trip Setpoint. The margin between RTP and some of the alarms has been slightly reduced. This is not expected to be an operational burden because the instrumentation used to calculate the heat balance, which is also used to establish the alarms and control the plant, are being significantly improved as part of the MUR uprate project.

CR-3 controls reactor power with the Automated Unit Load Demand (AULD) subsystem. AULD controls reactor power through the Integrated Control System based directly on the secondary heat balance. Neither the AULD software nor administrative controls, when it is not in-service, authorize power operation above RTP. However, minor fluctuations in plant conditions do result in minor changes in actual power. The AULD (or the operator) promptly respond to these conditions restoring operation to less than or equal to RTP. Further, they reduce power as needed to assure the shift average is maintained less than

RTP. Power fluctuations are generally in the range of 5 to 10 MWt and are promptly adjusted by the AULD. Such conditions are well bounded by margin provided in the analytical limits for the RPS Nuclear Overpower Trip Setpoint (analytical limit is 110.2% of 2609 MWt – see Table D.2-2 of LAR 296, Revision 1, dated June 28, 2007, for detail on analytical margin).

The current NRC guidance on such conditions will continue to be used to evaluate all departures from RTP. CR-3 has evaluated RIS 2007-21 and recognizes that the guidance does not authorize steady state operation above 100% RTP.

3.b. There are no changes to controls and no significant changes to control room indications. The LEFM system does include alarm functions that will be displayed through the AULD and the plant computer system. The fundamental function of the AULD and the man-machine interface is largely unchanged. The most significant change is that the AULD will be capable of automatically controlling the plant to 2609 MWt when the appropriate inputs are available or 2568 MWt when they are not. The selection of which input set to control the plant to is an operator selected option.

3.c. Both the new Fixed Incore Detector Monitoring System (FIDMS) and the modified AULD will be thoroughly tested in Information Technology laboratory settings, the plant simulator, and in the field. This testing is an integral part of the Engineering Change (EC) and is being actively supported by operator and operator training staff. The EC process includes appropriate features unique to modifications which involve software or other digital controls. These features provide reasonable assurance that the man-machine interfaces are thoroughly understood, validated, and tested.

The design of the AULD and LEFM contain a number of features to assist the operators in detecting and diagnosing failed inputs or other hardware failures.

3.d. Routine operator rounds will monitor local indications of LEFM failures. Further, the AULD is also capable of detecting and alerting operators of differences between certain redundant inputs. Finally, operator monitoring of related control board indications will assist in the detection and diagnosis of failed inputs or outputs

NRC Request 4

4. Control Room Plant Reference Simulator (RIS 2002-03 Section VII.2.C)

a. How will the licensee verify the plant simulator's fidelity after the MUR Power Uprate-related modifications are made?

FPC Response

4.a. Due to the training significance of this particular modification, the simulator changes will actually be made ahead of the plant change, but maintained in a separate software configuration so as not to impact the Initial Operator Training program.

It is intended to utilize the simulator to expose the crews to the scope of MUR during a September Just-In-Time training. The simulator performance will be verified to be consistent to the intended plant design by conducting a formal integrated simulator

Acceptance Test which spans the ECs comprising MUR. This is the Verification portion of simulator testing as described in CR-3 training procedures TAP-206 and TAP-422. This process is used for all changes made on the simulator. After MUR is physically installed at the plant, the simulator will be modified (as necessary) to reflect any field changes (i.e., EC revisions) that affect the performance/appearance of the change as compared to the original design and also to "tune" simulator results to actual plant data. This is the Validation portion of the simulator testing and also described in TAP-206 and TAP-422. The totality of the simulator change will be maintained as a Simulator Work Package for future traceability and ease of debugging. The Work Package will also include the Acceptance Test and sufficient documentation to describe the differential change (before versus after).

Table C.1

Summary of the Operator Actions Associated with EOPs

Below is a list of operator actions that must be accomplished to satisfy FSAR Chapter 14 events, licensing events or protect equipment. The list defines the operator action, time requirements, basis and why it is not affected. This list does not cover standard Limits & Precautions (L&P) that must be performed to protect equipment. These actions are equipment specific and are not affected by the MUR.

Operator action	Time	Basis	Why it is not affected
Trip RCPs on loss of adequate subcooling margin (ASCM)	1 min	Requirement is assumed in the Small Break Loss of Coolant Accident (SBLOCA) analysis. The specific limit is based on a Core Flood Line Break.	The analysis is based on 102% power. The MUR project is not changing the analysis limit.
Isolating letdown on loss of Adequate Sub-Cooling Margin (ASCM)	10 min	Requirement is to satisfy the letdown line break analysis outside containment. The analysis assumes the operator will isolate letdown within 10 minutes of a loss of ASCM to stop the RCS release to the auxiliary building. This requirement is also performed in the event of a SBLOCA.	The analysis is based on 102% power. The MUR project is not changing the analysis limit. Also, this analysis is based on a specific release rate and initial RCS concentration limits that would not be impacted by the MUR.
Initiate the action to raise OTSG levels to the Inadaquate subcooling margin OTSG level on loss of ASCM	20 min	Requirement is assumed in SBLOCA analysis.	The analysis is based on 102% power.
Start / ensure control room emergency ventilation system (CREVS) is operating in the emergency mode	30 min	Requirement is assumed in the Control Complex habitability analysis. Performing the action maintains the control complex within the analyzed temperature profile. Also, the action is assumed in the Control Room dose analysis.	The increase in power does not affect the heat load within the control complex. The addition of new equipment in the control complex due to the MUR project is adding additional load, but is within the analyzed limits. The Control Room dose analysis is based on 102% and the MUR is not affecting the overall limit of 2619 MWt.

Operator action	Time	Basis	Why it is not affected
Start / ensure a control complex chiller is operating	120 min (loss of offsite power) 130 min (non loss of offsite power events)	Requirement is assumed in the Control Complex habitability analysis. Performing the action maintains the control complex within the analyzed temperature profile.	The increase in power does not affect the heat load within the control complex. The addition of new equipment in the control complex due to the MUR project is adding additional load, but is within the analyzed limits.
Stop the make-up pumps DC lube oil and gear oil pumps during a station blackout event	30 min	Requirement assumed in the Station Blackout Analysis for battery loading to stay within the 4 hour coping requirements for a station blackout event.	The increase in power is not affecting the loading on the 1E batteries.
Open electrical cabinets, Control Room doors and Control Cabinets during a station blackout event	30 min	Requirement assumed in the Station Blackout Analysis to maintain adequate temperature profile within the electrical and control cabinets and the control room.	The increase in power does not affect the heat load within the electrical and control cabinets or control room. The addition of new equipment in the control complex due to the MUR project is adding additional load, but is within the analyzed limits.
De-energize VBIT-1E and DCS Server Cab A/B during a station blackout event	30 min	Requirement assumed in the Station Blackout Analysis to eliminate a heat load within the Control Room.	The increase in power does not affect the heat load analysis for the control room.
Depressurizing the main electrical generator and securing TBP-10 on loss of offsite power	60 min	Requirement is tied to preserving the non-1E batteries to allow time to purge the main electrical generator of hydrogen to prevent a hydrogen explosion.	The increase in power does not affect the non-1E battery loading.
Purge the main electrical generator with nitrogen	150 min	Purge the main electrical generator of hydrogen to prevent a hydrogen explosion. The limit is tied to the non-1E battery loading.	The increase in power does not affect the non-1E battery loading.

Operator action	**Time**	**Basis**	**Why it is not affected**
Initiate a cooldown using the Atmospheric Dump Valves (ADV's) for a Steam Generator Tube Rupture (SGTR)	45 min (after reactor trip)	Requirement is assumed in the Control Complex habitability analysis for a SGTR with a loss of offsite power and a single failure.	The analysis is based on a specific RCS concentration which is not being affected by the MUR project. Also, the analysis is based on 102% which is not being affected.
Initiate an RCS depressurization with the Power Operated Relief Valve (PORV) for a SGTR	47 min (after reactor trip)	Requirement is assumed in the Control Complex habitability analysis for a SGTR with a loss of offsite power and a single failure.	The analysis is based on a specific RCS radionuclide concentration which is not being affected by the MUR project. Also, the analysis is based on 102% which is not being affected.
Manually open failed ADV for a SGTR	70 min (after reactor trip)	Requirement is assumed in the Control Complex habitability analysis for a SGTR with a loss of offsite power and a single failure. The worst case failure is a loss of the ADV on the OTSG that does not have a tube rupture. This requires the failed steam generator to be steamed until an operator can be sent to the valve locally to manually control the ADV position.	The analysis is based on a specific RCS radionuclide concentration which is not being affected by the MUR project. Also, the analysis is based on 102% which is not being affected.
Isolate affected OTSG for a SGTR	100 min (after a reactor trip)	Requirement is assumed in the Control Complex habitability analysis for a SGTR with a loss of offsite power and a single failure. Isolating the OTSG stops the release.	The analysis is based on a specific RCS radionuclide concentration which is not being affected by the MUR project. Also, the analysis is based on 102% which is not being affected.
Trip RCPs on Reactor Building Isolating and Cooling (RBIC) actuation	30 min	The limit is to protect the integrity of the RCP seals. With a RBIC, the RCP control bleed-off isolation valve closes which requires the RCPs to be tripped.	The increase in power does not affect the RCP seal integrity.

Operator action	Time	Basis	Why it is not affected
Place the RB hydrogen monitors in service	90 min	This is a commitment based on NUREG-0737.	The increase in power does not affect the NUREG-0737 requirements for monitoring hydrogen. Also, the hydrogen monitoring requirements have been down graded and are not part of the ITS requirement.

PROGRESS ENERGY FLORIDA, INC.

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LICENSE AMENDMENT REQUEST #296, REVISION 1

ATTACHMENT D

RAI RESPONSE –
ELECTRICAL ENGINEERING BRANCH (EEEB)

Request for Additional Information Response

On August 8, 2007, Florida Power Corporation (FPC) received a Request for Additional Information (RAI) concerning License Amendment Request (LAR) #296, Revision 1 via email. FPC hereby provides the responses to this RAI that was received from the Electrical Engineering Branch (EEEB).

NRC Request EEEB-1

1. **Provide the existing and uprated power level in MW(e).**

FPC Response

The current power level in MWe (Megawatts electric) is approximately 900 MWe. The proposed power level in MWe, after the MUR uprate is implemented, will be approximately 914 MWe.

NRC Request EEEB-2

2. **Provide a detailed comparison of existing ratings with uprated ratings and the effect of the power uprate on the following equipment:**

   A. **main generator rating and power factor**
   B. **isophase bus**
   C. **main generator breaker**

FPC Response

The existing CR-3 electrical generator is rated at 989.4 MVA, 0.90 power factor (0 MVAR). The generator is operated with an administrative limit of 300 MVARs. At this reactive load, the maximum output from the generator reactive capability curve is approximately 950 MWe. The MUR will increase output from approximately 900 MWe to 914 MWe. Therefore, the increase from the MUR uprate is still well below the main generator maximum capability.

The iso-phase bus is rated for 27,500 Amps. The typical 100% power operating current ranges between 23,500 Amps and 24,500 Amps. The maximum of 1.6% increase from the MUR uprate will increase the current on the iso-phase bus to approximately 24,000 Amps to 25,000 Amps. Therefore, the increase from the MUR uprate is still well below the iso-phase bus maximum capability.

There is no circuit breaker between the generator and the generator step-up transformer. The 500 KV line from the step-up transformer joins the 500 KV ring bus between the two generator breakers. The main generator output breakers are rated for 3000 Amps with a short circuit current rating of 37,000 Amps. The current through a breaker before uprate is approximately 1300 to 1800 Amps which is well below the 3000 Amp rating of these breakers. With a 1.6% MUR uprate, the expected to be approximately 1325 to 1850 Amps, and therefore, there will still be significant margin to the breakers rating.

NRC Request EEEB-3

3. For the current uprate of 1.6%, please address and discuss the following:

A. Quantity and nature of MVAR support necessary to maintain post-trip loads and minimum voltage levels. Address the effects of the power uprate on MVAR support.

B. How the power uprate changes the MVAR contributions credited by the TSO.

C. Compensatory measures taken to compensate for the depletion of the nuclear unit MVAR capability on a grid-wise basis due to this power uprate.

D. Provide an evaluation of the impact of any MVAR shortfall listed in item C on the ability of the offsite power system to maintain post-trip voltage levels and to supply power to safety buses during peak electrical demand periods. The subject evaluation should document any information exchanges between the TSO and Crystal River on this matter.

FPC Response

3.a. There is no minimum MVAR support required to be maintained on the grid in order to support the minimum post trip voltage for CR-3. CR-3 maintains a maximum MVAR limit based on manufacturer's recommendations to prevent damage to the generator. This limit is approximately 430 MVARs (leading or lagging).

Note: CR-3 output is provided to the 500 KV switchyard. The power to both the preferred and alternate sources of offsite power is provided from the 230 KV switchyard. The two switchyards are electrically separate and loss of CR-3 generator output will only affect the voltage for the 500 KV switchyard, and not the 230 KV switchyard. Hence, the 230 KV switchyard and the two offsite power sources are unaffected by the loss of CR-3 generator output. There is no interconnection between the 500 KV and the 230 KV lines for a distance of approximately 35 miles from the plant.

3.b. The Transmission System Operator (TSO) will not require additional MVARs from CR-3 post implementation of the MUR uprate. A study has been performed that demonstrates the additional power production will not have to include additional MVARs in order to meet North American Electric Reliability Corporation (NERC) reliability criteria. CR-3 will be able to continue to provide the same MVAR contribution that is currently credited by the TSO (approximately 430 MVARs).

3.c. There is no depletion (shortfall) of MVAR capability resulting from the MUR uprate. As such, there are no compensatory measures to be taken to compensate for the shortfall.

3.d. There is no shortfall in MVAR capability resulting from the MUR uprate. As such, the MVAR contribution to the post trip voltage levels is not impacted by the MUR uprate.

Note: the offsite power for CR-3 post-trip and emergency loads comes from the 230 KV switchyard. Power produced by the CR-3 main generator is stepped up and sent out through the 500 KV switchyard. There is no interconnection between the 500 KV and the 230 KV lines for a distance of approximately 35 miles from the plant.

NRC Request EEEB-4

4. **Please confirm that the accident profile to which environmental qualification of electrical equipment is performed includes any radiation dose changes due to the current power uprate.**

FPC Response

The source terms used for the radiation aspects of the equipment qualification (EQ) profile were evaluated at 102% of 2568 MWt (2619 MWt) which remains the appropriate value post-MUR (100.4% of 2609 MWt). Thus, there is no change to radiation dose and the EQ profiles remain bounding.

NRC Request EEEB-5

5. **In Attachment D, section 5.2.4., "Grid Stability" of the submittal dated June 28, 2007, the licensee concludes that an increase in capacity will not have an adverse effect on the grid. Provide a more detailed discussion and/or supporting evidence for the factors or criteria considered for the above conclusion.**

FPC Response

Response is included as Attachment E, "Capacity Increase Stability Study." Please note that the study addresses all phases of the planned power uprates at CR-3 (180 MWe) over the next four years.

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #296, REVISION 1

ATTACHMENT E

RAI RESPONSE –
CAPACITY INCREASE STABILITY STUDY

Crystal River #3

Capacity Increase

Stability Study

Progress Energy Florida, Inc.

Transmission Planning

August 8, 2007

Crystal River #3

Capacity Increase

Table of Contents

Summary

Transmission Planning of Progress Energy Corporation has reviewed the 180 MW capacity increase for Crystal River No. 3 (CR3). Specifically, requests were made through NRC and through the FERC Large Generator Interconnection procedures to uprate CR3 in three phases: a) 14 MW in December 2007, b) 26 MW in December 2009 and c) 140 MW in December 2011. Various dynamic and steady state scenarios were studied. The uprates were analyzed as a single increase of 180 MW in 2011, with the intent that any unsatisfactory results for the total uprate would necessitate analysis of the individual phases.

The stability simulation for NERC category D5 type faults (breaker failure for three phase fault) does not indicate any adverse effect of the capacity increase on the bulk grid system. The critical clearing time simulation indicated an approximate reduction of one (1) cycle, that is from 9.5 cycles to 8.5 cycles.

The steady state analysis was performed to examine post transient power flow for the bulk Florida transmission system to determine if the loss of line and units subsequent to breaker failure will cause voltage and or overload of system components. It is determined that Post Transient Power Flow does not cause any problem.

Introduction

Progress Energy Florida is considering the capacity increase for the Crystal River #3 generator. An increase of 180 MW will be accomplished in steps. The analysis was performed using the following methodologies and assumptions:

For the transient stability analysis a 2012 summer dynamic case developed by the Stability Working Group (SWG) of the Florida Reliability Coordinating Council (FRCC) was used as a starting point. The case (y06_12s-dynd.sav) was modified to reflect the necessary changes. The following scenarios were studied:

1. Three phase fault on Crystal River 500 kV bus, cleared normally. NERC Category B1.
2. Three phase fault on Crystal River 500 kV bus, with breaker failure condition, NERC Category D1.
3. Three phase fault on Crystal River 500 kV bus, critical clearing time.

A comparison of the results indicated that an increase in capacity will not have an adverse effect on the stability of power grid. A NERC category C fault was not studied since a more severe category D event was simulated and results are satisfactory. Note that Progress Energy Florida's Transmission Planning unit is required to demonstrate compliance with the NERC Transmission Planning Standards TPL-001-0, TPL-002-0, TPL-003-0 and TPL-004-0. For more information on NERC Reliability Standards for

Transmission Planning (TPL), please reference the following link at NERC's website: http://www.nerc.com/~filez/standards/Reliability_Standards_Regulatory_Approved.html

Conclusion

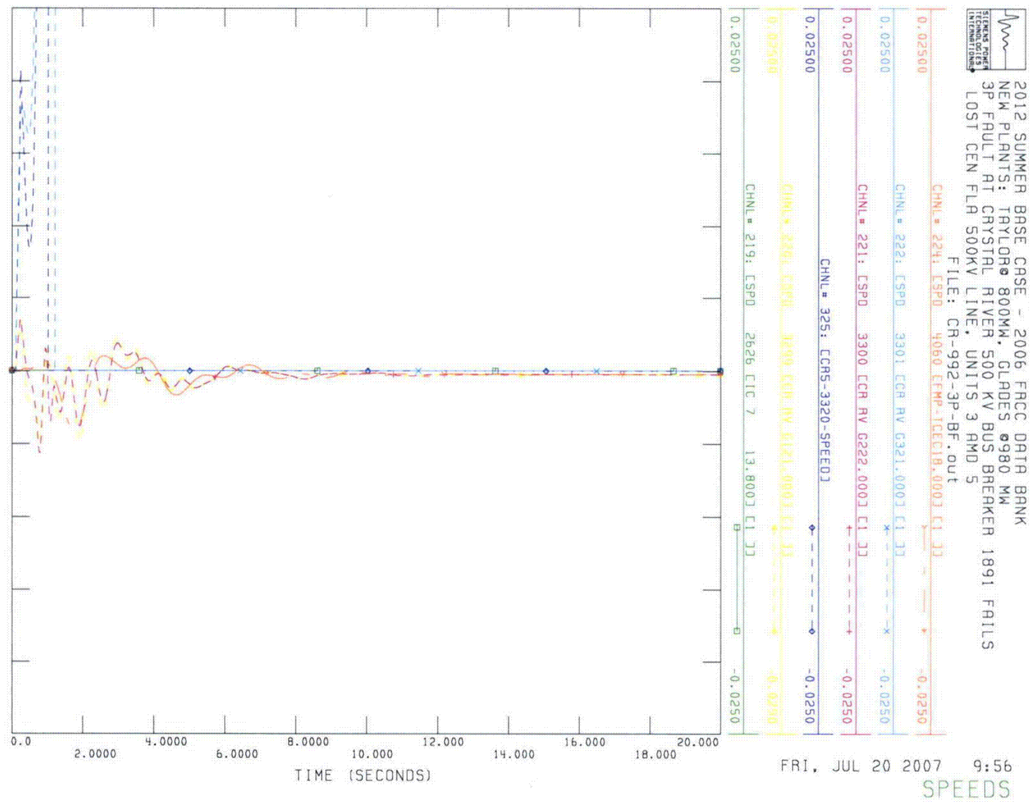
Based on the analysis performed, it is concluded that the scenarios studied do not indicate any adverse effect on grid stability. All the simulations and studies are based upon the full 180 MW increase, which is considered a worst case situation for transient stability.

Study Results

Case #:	CR-992-3P-BF
Initial Conditions:	2012 Summer Peak Load
Disturbance:	Three phase fault at the Crystal River 500 kV with breaker failure (Bkr. 1891)

Event Sequence Time (sec)	Event
0.100	Apply three phase fault on Crystal River 500 kV bus.
0.258	Open Crystal River – Central Florida 500 kV, open breaker #1660 by BFBU and clear fault. (9.5 cycles BFBU clearing time, Crystal River #5 is isolated)
1.2167	Machine #3 trips on overspeed..

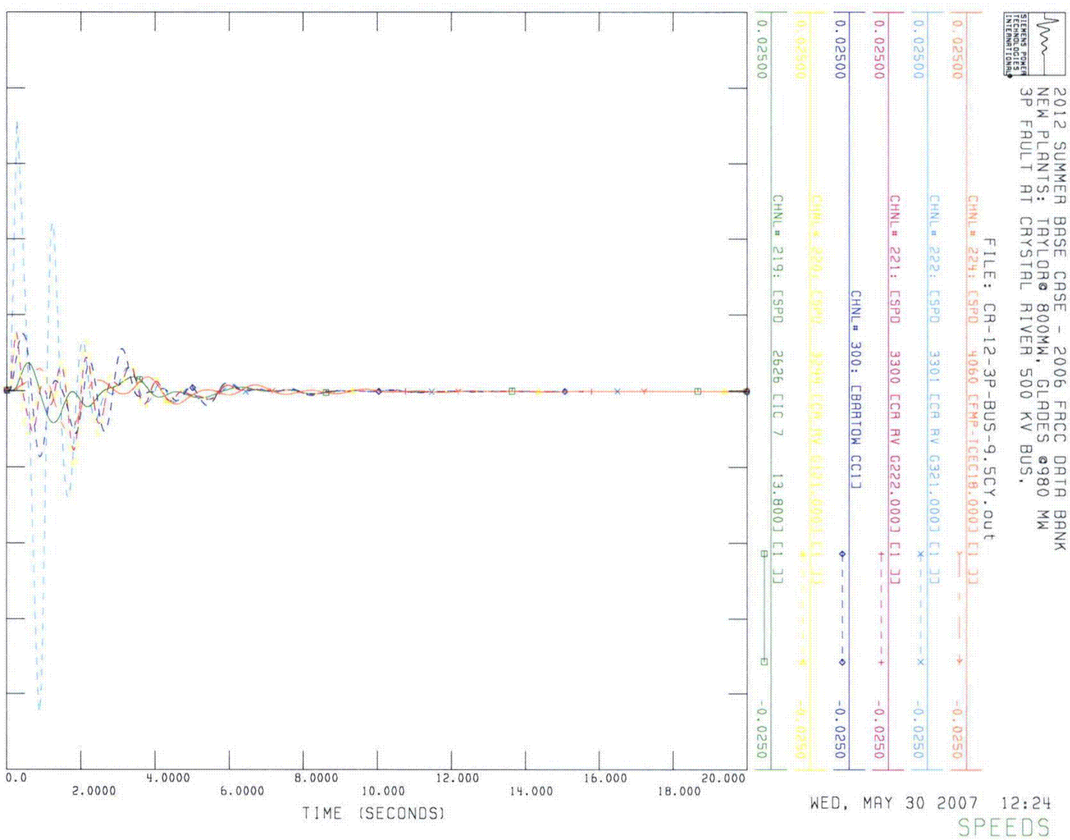
Load Shedding:	720.0 MW
Comments:	1529 MW generation lost. Post transient power flow analysis shows no voltage or loading problems.



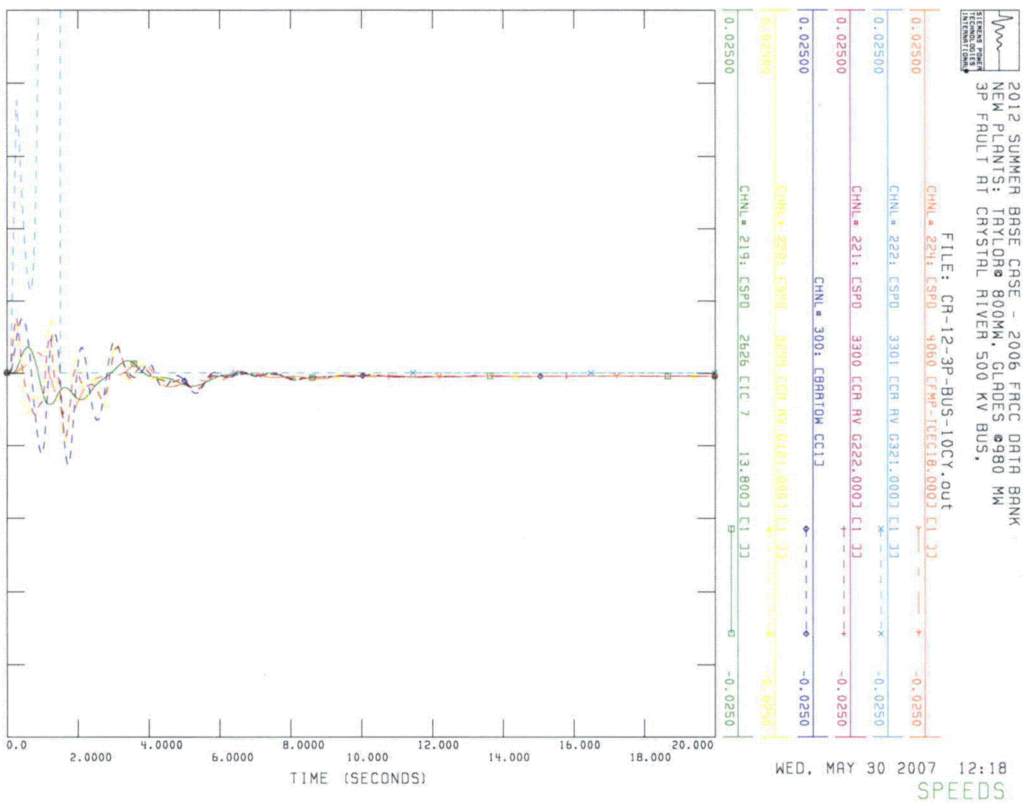
BREAKER FAILURE

Transient Stability Study, Critical Clearing Time

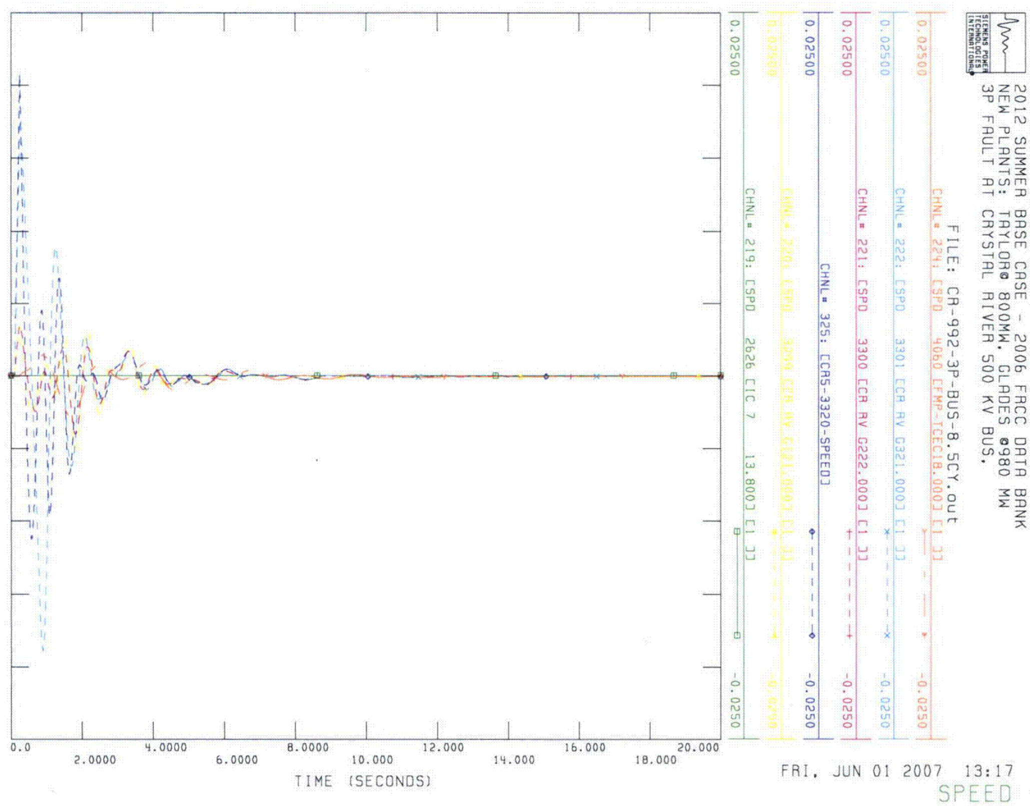
1. Three phase fault for 9.5 cycles duration is stable, generator at 812 MW.



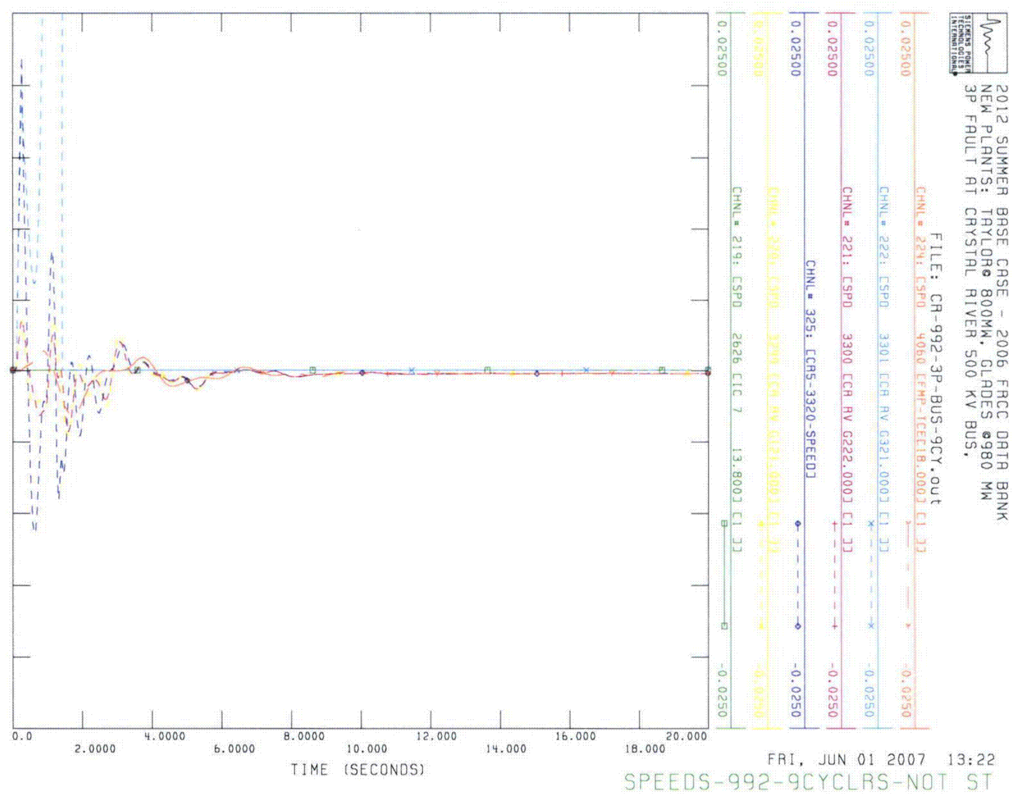
2. Three phase fault for 10.0 cycles duration results in loss of synchronism, generator at 812 MW.



3. Three phase fault for 8.5 cycles duration is stable, generator at 992 MW.



4. Three phase fault for 9.0 cycles duration results in loss of synchronism, generator at 992 MW.



PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #296, REVISION 1

ATTACHMENT F

RAI RESPONSE –
VESSELS AND INTERNALS INTEGRITY BRANCH (CVIB)

Request for Additional Information Response

On August 20, 2007, Florida Power Corporation (FPC) received a Request for Additional Information (RAI) concerning License Amendment Request (LAR) #296, Revision 1 via email. FPC hereby provides the responses to this RAI that was received from the Vessels and Internals Integrity Branch (CVIB).

NRC Request CVIB-1

In Section 4.2.3.3, Heatup and Cooldown Pressure/Temperature Limit Curves, it is stated that based on the additional credible reactor vessel surveillance data, the chemistry factors utilized in the ART [adjusted reference temperature] calculations were reduced leading to an overall reduction in ART at 32 EFPY. Identify the capsule providing the additional data. Also identify the limiting material for ART with the following information (at 32 EFPY): the inner diameter fluence, margin, chemical composition values (weight percent Cu and Ni), technical rationale for updating the chemical composition values, and chemistry factor. Reference 4.3.7, CR-3 MUR RV Integrity Summary, is identified as containing supporting information. Provide this document for reference or provide all information in the document relevant to the evaluation of Crystal River, Unit 3 heatup and cooldown pressure/temperature limit curves.

FPC Response 1

ART calculations were performed in 1997 in support of the currently licensed CR-3 32 EFPY P-T curves. Weld SA-1769 was the 1/4T limiting material and weld WF-169-1 was the limiting 3/4T material with ARTs of 213.0°F and 144.5°F, respectively. The ART value for SA-1769 was calculated using RG 1.99, Revision 2, Position 2.1 and the ART value for WF-169-1 was calculated using RG 1.99, Revision 2, Position 1.1. These ART calculations were based on the existing best estimate Cu and Ni values available at that time.

In response to Generic Letter 92-01, Revision 1, "Reactor Vessel Structural Integrity," additional chemistry data for B&W fabricated welds were assembled as part of NRC review of the B&W Owners Group response to GL 92-01, Revision 1. These proceedings were documented in NRC inspection report no. 99901300/97-01 dated January 28, 1998. The result of consideration of the additional data was that the Cu and Ni values for various welds either increased or decreased. This effect was determined not to impact any existing B&W plant P-T curves as documented in BAW-2325, Revision 1, which was submitted to the NRC. No additional capsule data was utilized to update these chemistry values. As a result of the NRC review of the B&W Owners Group's response to GL 92-01, Revision 1, the reactor vessel surveillance data for weld SA-1769 was determined to be non-credible in accordance with RG 1.99, Revision 2, Position 2.1.

ART calculations performed in 2002 for CR-3 used the applicable post-GL 92-01, Revision 1, chemistry values which resulted in lower chemistry factors for welds SA-1769 and WF-169-1 and higher chemistry factors for welds WF-8 and WF-18. The 2002 ART calculations are based on RG 1.99, Revision 2, Position 1.1. As a result, SA-1769 remained the limiting location and its ART decreased from 213.0°F to 195.7°F, even when accounting for a small increase in fluence due to the MUR power uprate. WF-169-1 had its ART decrease from 144.5°F to 138.2°F, even when accounting for a small increase in fluence due to the MUR power uprate. WF-8 and WF-18, which contain the same weld wire heat, had their ARTs increase from 142.7°F to 144.1°F becoming the new limiting 3/4T location, accounting for a small increase in fluence due to the MUR power uprate. The fluence values used in the ART calculations were conservatively increased by 7% while a 2006 fluence assessment concluded that the actual 32 EFPY fluence values would only increase by 2% due to the MUR uprate.

The following table provides the detailed information used in the 2002 MUR uprate ART calculations requested in the RAI.

Location	**Material**	**ID Fluence**	**Fluence Factor**	**Margin**	**Cu**	**Ni**	**Chemistry Factor**	**ART**
Limiting 1/4T	Weld SA-1769	7.58E+18 E>1.0MeV	0.774 (@ 1/4T)	56°F	0.23	0.59	167.6	195.7
Limiting 3/4T	Welds WF-8 & WF-18	7.92E+18 E>1.0MeV	0.529 (@ 3/4T)	68.5°F	0.19	0.57	152.4	144.1

<u>NRC Request CVIB-2</u>

In Section 4.2.3.5, Effect on Upper-Shelf Energy (USE), identify the limiting material, Cu, and fluence values used to determine that the equivalent margins analysis is still applicable under the MUR power uprate. Reference 4.3.7, CR-3 MUR RV Integrity Summary, is identified as containing supporting information. Provide this document for reference or provide all information in the document relevant to the evaluation of Crystal River, Unit 3 USE analysis

<u>FPC Response 2</u>

The current licensing basis for CR-3 low USE is contained in BAW-2192PA and BAW-2178PA, both of which were submitted to the NRC and an SER was obtained in 1994. The limiting materials for CR-3 are WF-70 and WF-8 / WF-18. Pertinent information used in BAW-2192PA and BAW-2178PA is contained in the table below. The fluence data used in BAW-2192PA and BAW-2178PA is based on a 1992 fluence projection.

Material	ID Fluence	Cu	Ni
Weld WF-70	8.22E+18 E>1.0MeV	0.35	0.59
Welds WF-8 & WF-18	7.96E+18 E>1.0MeV	0.20	0.55

Since 1994, additional fluence evaluations have been performed. The current 32 EFPY fluence projections, accounting for a 2% increase in fluence due to the MUR, are bounded by those fluence values used in BAW-2192PA and BAW-2178PA listed above. The 1994 low USE analyses have not been updated, thus they utilize pre-GL 92-01, Revision 1 chemistry data (Cu & Ni values). For welds WF-70, WF-8, and WF-18, the GL 92-01, Revision 1 evaluations resulted in the Cu contents for these welds to decrease, which makes BAW-2192PA and BAW-2178PA bounding. Therefore, it was concluded that the current low USE licensing basis remains valid. See *FPC Response 1* above for more information on B&W Owners Group GL 92-01, Revision 1 chemistry data.

NRC Request CVIB-3

In Section 4.2.3.1, Pressurized Thermal Shock, additional information is required to review the reference temperature for pressurized thermal shock (RT_{PTS}), including identification of the limiting material, chemical composition values (weight percent Cu and Ni), inner diameter fluence, initial reference temperature, increase in the RT_{PTS} caused by irradiation (ΔRT_{PTS}), and margin for the determination of RT_{PTS}. Reference 4.3.7, CR-3 MUR RV Integrity Summary, is identified as containing supporting information. Provide this document for reference or provide all information in the document relevant to the evaluation of Crystal River Unit 3 RT_{PTS} analysis.

FPC Response 3

The relevant information requested in the RAI above is provided below.

Reactor Vessel Beltline Region Material	Material Ident.	Cu wt%	Ni wt%	32 EFPY Fluence at Inside Surface, n/cm^2	Chemistry Factor	Fluence Factor	ΔRT_{PTS}, °F	Initial RT_{NDT}, °F	Margin, °F	RT_{PTS}, °F	Screening Criteria
US Longit. Weld (100%)	WF-8 & WF-18	0.19	0.57	7.92E+18	152.4	0.935	142.5	-5	68.5	[206.0]	270

NRC Request CVIB-4

Section 4.2.1.2, Reactor Vessel Internals Structural Evaluation, requires additional information. Table Matrix 1 of Nuclear Regulatory Commission RS-001, Revision 0, "Review Standard for Extended Power Uprates," provides the staff's basis for evaluating the potential for extended power uprates to induce aging effects on RV internals. Depending on the magnitude of the projected reactor vessel (RV) internals fluence, Table Matrix 1 may be applicable to the MUR application. In the Notes to Table Matrix 1, the staff states that guidance on the neutron irradiation related threshold for irradiation assisted stress corrosion cracking (SCC) for PWR RV internal components are given in BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," and WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals." The "Notes" to Table Matrix 1 state that for thermal and neutron embrittlement of cast austenitic stainless steel, SCC, and void swelling, licensees will need to provide plant-specific degradation management programs or participate in industry programs to investigate degradation effects and determine appropriate management programs.

Discuss your management of the above mentioned aging effects on RV internals in light of the guidance in BAW-2248A and WCAP-14577, Revision 1-A.

Please also confirm whether you have established an inspection plan to manage the age related degradation in the Crystal River, Unit 3 RV internals, or whether you have participated in the industry's initiatives on age-related degradation of PWR RV internals.

Reference 4.3.6, CR-3 MUR Power Uprate RCS Structural Assessment, is identified as containing supporting information. Provide this document for reference or provide all information in the document relevant to the evaluation of the Crystal River 3 RV internals structural evaluation.

FPC Response 4

FPC is actively participating in the EPRI Materials Reliability Program (MRP) Reactor Internals Focus Group, which is working to establish generic I&E guidelines for PWR internals during the license renewal period. FPC is developing a CR-3-specific RV internals inspection program considering CR-3-specific parameters (including MUR conditions), based on the EPRI MRP recommendations. The nominal 2% increase in 32 EFPY fluence due to the MUR power uprate is expected to have an insignificant impact on irradiation related aging degradation of the RV internals.