Progress Energy

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

August 11, 2011
3F0811-04

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 Probabilistic Risk Assessment Licensing Branch Acceptance 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. Email from S. Lingam (NRC) to D. Westcott (CR-3) dated July 27, 2011, "CR-3 EPU LAR - RAIs from APLA Branch (TAC No. ME6527)"

Dear Sir:

By letter dated June 15, 2011, Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc., requested a license amendment to increase the rated thermal power level of Crystal River Unit 3 (CR-3) from 2609 megawatts (MWt) to 3014 MWt. The proposed license amendment is considered an Extended Power Uprate (EPU). On July 27, 2011, via electronic mail, the NRC provided a request for additional information (RAI) related to the Probabilistic Risk Assessment (PRA) quality needed to support the PRA Licensing Branch acceptance review of the CR-3 EPU License Amendment Request (LAR).

The Attachment, "Response to Request for Additional Information to Support NRC Probabilistic Risk Assessment Licensing Branch Acceptance Review of the CR-3 EPU LAR," provides the CR-3 formal response to the RAI.

This correspondence contains no new regulatory commitments.

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

Sincerely,

Jon A. Franke
Vice President
Crystal River Nuclear Plant

JAF/scp

Attachment: Response to Request for Additional Information to Support NRC Probabilistic Risk Assessment Licensing Branch Acceptance Review of the CR-3 EPU LAR

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

A001
NRR

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

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 11 day of August, 2011, by Jon A. Franke.

Signature of Notary Public
State of Florida

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

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

Personally Known ✓ -OR- Produced Identification ______

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 /LICENSE NUMBER DPR-72

ATTACHMENT

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION TO SUPPORT NRC PROBABILISTIC RISK ASSESSMENT LICENSING BRANCH ACCEPTANCE REVIEW OF THE CR-3 EPU LAR

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION TO SUPPORT NRC PROBABILISTIC RISK ASSESSMENT LICENSING BRANCH ACCEPTANCE REVIEW OF THE CR-3 EPU LAR

By letter dated June 15, 2011, Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc., requested a license amendment to increase the rated thermal power level of Crystal River Unit 3 (CR-3) from 2609 megawatts (MWt) to 3014 MWt (Reference 1). The proposed license amendment is considered an Extended Power Uprate (EPU). On July 27, 2011, via electronic mail, the NRC provided a request for additional information (RAI) related to the Probabilistic Risk Assessment (PRA) quality needed to support the PRA Licensing Branch acceptance review of the CR-3 EPU License Amendment Request (LAR).

<u>NRC Request for Additional Information</u>

The licensee should provide all open items related to both the Nuclear Energy Institute Peer Certification and Gap Assessment for the CR-3 PSA model of record and characterize the impact of the open items for the EPU application.

CR-3 Response:

The peer reviews of the CR-3 PRA model include an American Society of Mechanical Engineers (ASME) Self Assessment in 2007, a focused Peer Review in 2009, and a Fire PRA Peer Review in 2009. The facts and observations (F&Os) that have been resolved are incorporated into the MOR09 PRA model. Their impact on the EPU assessment is encompassed in the overall PRA evaluation of the EPU project, therefore only open F&Os are addressed in this response. Appendices 1 thru 3 provide the list of Peer Review items that have not been fully resolved and their impact on the EPU project. Appendix 1 provides the open F&Os resulting from a 2007 Self Assessment of the CR-3 PRA. Appendix 2 contains the open F&Os from the 2009 focused Peer Review of the CR-3 PRA. Additionally, Appendix 3 provides the open F&Os from the 2009 Fire Peer Review of the CR-3 Fire PRA. The "Finding" level of significance listed in the appendices indicates that the reviewed element was not satisfactory or was rated ASME Category I, which equates to level 1 or 2 (less than 3) in a 1-4 rating system. The F&Os mostly pertain to thermal hydraulic analysis, Level 2 PRA conservatism, and Fire PRA uncertainty and overcurrent protective device coordination issues. The impacts of the Thermal Hydraulic F&Os were evaluated with RELAP, Gothic, and MAAP analysis, and there is no significant impact on the EPU PRA results. The Level 2 PRA F&Os pertain to removing conservatism from the Level 2 analysis and would reduce the predicted large early release frequency (LERF). Therefore, the Level 2 PRA F&Os do not result in a significant impact to the EPU PRA. The open Fire PRA F&Os deal with uncertainty analyses and overcurrent protective device coordination issues which would not significantly impact the core damage frequency results, therefore, does not impact the EPU PRA results.

Reference

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)

Appendix 1

CR-3 ASME 2007 Self Assessment
Open Facts & Observations Potentially Impacting EPU

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Observation: **FnO-AS-C1**	**Technical Element:** **AS**	**Supporting Requirement:** **AS-C1**
The Accident Sequence notebook (Calculation P-02-0004) Figure 3, Core Damage Event Tree for a Small LOCA, has Event Z, Early Coolant, that is not utilized as a mitigating strategy or discussed in any other part of the notebook.		
Level of Significance: *C*	**AR***:*	
Resolution*:*		
The Event Z in the Small LOCA event tree is a historical input used in previous models. It can be deleted from the event tree and has no significance in the current model of record.		
EPU Application Notes:		
This is a documentation issue. No impact on the EPU application.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Observation: **FnO-SC-B1-1**	**Technical Element:** **SC**	**Supporting Requirement:** **B1**
A series of success criteria runs to determine the time for feed and bleed (F&B) are presented in RSC-01-61, run SC-2A. In run SC-2a#3, feed and bleed prior to 20 minutes showed no core damage. Run SC2a#4 showed F&B at 30 minutes caused the core damage criteria to be exceeded, but 30 minutes was chosen as the feed and bleed time. These items are not called out in the sensitivity studies. Uncertainty and sensitivity runs should be done to examine the affect of using an F&B criterion of less than 30 minutes, without reactor building (RB) pressure signal initiation.		
Level of Significance: Finding	**AR:**	
Resolution:		
Sensitivities have been run using MAAP 4.06 that indicate no core damage if feed and bleed cooling is delayed until 30 minutes.		
EPU Application Notes:		
In addition to the sensitivities using later version of MAAP, a RELAP and Gothic analysis of containment response has been performed on the timing of high pressure injection (HPI) auto initiating based upon the proposed EPU modifications. The results for success of this analysis are documented in calculation P10-0001. No impact on the EPU application.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Observation: FnO-SC-B1-2	**Technical Element:** SC	**Supporting Requirement:** SC-B1
Table 5, small loss of coolant accident (SLOCA) of P-02-0004, indicates that feed-and-bleed cooling must be initiated within 60 minutes of the loss of secondary cooling to ensure the core remains covered. However, the reference used (RSC 01-61) states that, "On the basis of these runs it is concluded that successful feed-and-bleed cooling can occur if either the automatic signal starts safety injection or the operators initiate safety injection within 30 minutes." See also Event L under the Transient Event Tree discussion for additional use of the 60 minutes criteria.		
Level of Significance: Finding	**AR:**	
Resolution:		
Sensitivities have been run using MAAP 4.06 that indicate no core damage if feed and bleed cooling is delayed until 30 minutes.		
EPU Application Notes:		
For transient-initiated accident scenarios involving the loss of all feedwater and feed and bleed cooling, and in which the reactor coolant pumps (RCPs) are <u>not tripped</u>, the time available for operator recovery to avert core damage is much shorter, 30 minutes. This is based on MAAP results for loss of feedwater scenarios, specifically from MAAP. This analysis shows the onset of core damage occurs when the power operated relief valve (PORV) was not opened nor the engineered safety feature (ESF) actuated until 30 minutes into an accident initiated by the loss of all feedwater. The RCPs are assumed to be running in this analysis. For station blackout (SBO) events involving the loss of all feedwater (and in which RCPs are tripped), the time available for operator action to avert core damage is 60 minutes. The success is based on MAAP results for SBO cases. The RCPs are tripped in this analysis. The timing is evaluated for the post EPU conditions and is included in the PRA EPU evaluation.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Observation: **FnO-SC-B4**	**Technical Element:** **SC**	**Supporting Requirement:** **B4**

Initiation of feed and bleed on loss of all feedwater is assumed to occur in response to high reactor building pressure, caused by coolant discharge through the PORV. Operation action to initiate feed and bleed is ANDED with auto-SI as a backup action in the CAFTA model. Report RSC-01-61, run SC-2b shows HPI actuation at 33 minutes, with MAAP variable TCRHOT defined as peak core temperature exceeding 1800°F for a short time. The time for feed and bleed by operator action is determined to be 30 minutes from run SC-2A#4. If the MAAP analysis does not consider heat absorption by containment structures, the calculation may not be suitable for success criteria. The assumption of RB hi pressure signal as an initiator for F&B is not mentioned as an uncertainty item. In order to establish the impact, a sensitivity study should be run with the auto actuation deleted and the Human Error Probability for manual action calculated for 25 minutes. This would serve as a conservative bound for feed and bleed actuation time.

MAAP typically uses conservative assumptions to result in higher than normal containment pressures, because this is considered conservative for most success criteria. MAAP does not model all the heat sinks in containment and looks only at one season of operation. When taking a fully realistic approach to containment pressure, the actual expected pressure would be less than calculated by MAAP.

Level of Significance: Finding	**AR:**

Resolution:

Sensitivities have been run using MAAP 4.06 that indicate no core damage if feed and bleed cooling is delayed until 30 minutes.

EPU Application Notes:

As part of the EPU evaluation, a more realistic containment evaluation using RELAP and Gothic analysis of containment response has been performed on the timing of HPI auto initiating based upon the proposed EPU modifications. The results of this analysis are documented in calculation P10-0001 and the results indicate that an operator action to initiate HPI feed and bleed is not required to prevent core damage.

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CR-3 ASME 2007 Self Assessment
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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS

Observation:	Technical Element:	Supporting Requirement:
FnO-HR-G4-1	HR	G4

The offsite power recovery analysis is based on the ability to cope for 50 minutes in SBO with no HPI or emergency feedwater (EFW). If power is restored at 50 minutes, core cooling can be recovered.

Offsite power is allowed to be recovered at 50 minutes. This leaves no time for restoration of power to the emergency busses and restoration of systems. A time lag must be accommodated between the time AC power is restored to the switchyard and feed and bleed (or EFW) is initiated.

Level of Significance: Finding

AR:

Resolution:

The 50 minutes is a historical value that has remained in use. The value appears to be a valid and conservative estimation based on the following considerations.

1. A review of RSC 01-61 (Thermal-Hydraulic Analysis) and underlying MAAP calcs for SBO was performed. Based on a flag event in the summary file, RCP steam binding occurs at 67.6 minutes which would indicate an approximated time for the loop seal for the once-through steam generators (OTSGs) to be broken. When the loop seal breaks, this marks the end of any potential recovery of secondary heat removal. In general, 60 minutes is the adopted time to recover secondary cooling, which is consistent with this timing.
2. Comparison of the SBO MAAP run to the F&B success criteria runs indicates that the SG dry out occurs at approximately 30 minutes earlier than in the SBO case due to continued operation of the RCPs. F&B cooling run SC-2a is used to illustrate that F&B cooling at 30 minutes is successful. Removing the RCP heat addition not present for SBO, provides at least 30 more minutes for the SBO case to implement F&B cooling (60 minutes).
3. The 60 minutes recovery time is consistent with the time to core uncovery in the SBO case of 1.2 hours and 1.5 hours to core damage.
4. The loss of offsite power recovery time includes restoration of power to emergency buses as discussed in NUREG/CR-6890 Vol.1. No additional time should be added for emergency power restoration from the switchyard.
5. The estimated time of 50 minutes for LOSP recovery gives the operators an additional 10 minutes to restart electric feed pumps EFP-1 or FWP-9, or to initiate F&B by starting makeup pumps (MUPs). SBO MAAP results indicate that containment pressure reaches 4 psi at approximately 50 minutes, so an HPI signal would be in place and F&B cooling would autostart before the time limit is reached.

Therefore, 50 minutes is a conservative value and does not need to be adjusted. The documentation in the loss of offsite power recovery notebook has been updated to clarify this timing.

EPU Application Notes:

As part of the review of affects of EPU on the PRA, the non recovery of offsite power was evaluated using 45 minutes based upon post EPU thermal hydraulic analysis and is documented in calculation P10-0001.

Appendix 2

CR-3 ASME 2009 Focused Peer Review
Open Facts & Observations Potentially Impacting EPU

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Element: IE	**Supporting Requirement**: A4a	**Observation #:** 01
Consideration of multiple failure and routine and non-routine (RG 1.200) system alignments do not appear to be considered when assessing the potential for initiating events on an individual system basis.		
Level of Significance: Finding	**Related SRs:** IE-A4a	
Basis for Significance:		
Review of each of the attachments (System Notebooks) of P02-0005, Rev. 4 (System Analysis) shows no evidence that multiple failures (from common cause failure) was explicitly considered in Section 12, where the potential for initiating events was assessed. Also, except for the fault tree models developed for IE_T10 and IE_T16 (see Sections 3.3.2.1 and 3.3.2.3 in P-02-0003, Rev. 2, Attachment 1), there is no evidence that routine system aligns were explicitly considered. Further, RG 1.200 clarification includes non-routine system alignments, which do not appear to have been addressed.		
Possible Resolution:		
Provide additional documentation in the System Notebooks to indicate that multiple failures were considered (when appropriate -- note that some of the systems do not include common cause failures in the system model). Provide additional documentation that routine and non-routine system alignments were considered (when appropriate -- note that some of systems will not have non-routine system alignments). Section 2 of the System Notebooks discusses normal and off-normal system operation.		
EPU Impact Review:		
CR-3 initiating event fault trees do include common cause failures and the ability to quantify with abnormal alignments. The fault trees also explicitly model various routine system alignments for applicable systems. This finding is an issue with documentation, and therefore is not expected to impact the results for the EPU application.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Element: IE	**Supporting Requirement:** A6	**Observation #:** 01
Capability Category I for IE-A6 does not require interviews with plant personnel to determine if potential initiating events have been overlooked. The Roadmap indicates there is no documentation of such interviews. P-02-0003 is silent about interviews.		
Level of Significance: Finding	**Related SRs:** IE-A6	
Basis for Significance:		
To meet Capability Category II, there must be some evidence that interviews occurred.		
Possible Resolution:		
Conduct interview with system engineers to confirm that no other potential initiating events need to be considered in the PRA model.		
EPU Impact Review:		
Plant operations were Individual Plant Examination (IPE) development team members and thus involved with developing initiating events for the PRA in the IPE. System engineers have reviewed the System Notebooks during development and during significant updates. The current PRA documentation does not describe the plant involvement in enough detail and needs to be enhanced. This finding is an issue with documentation, and therefore is not expected to impact the results for the EPU application.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Element: IE	**Supporting Requirement:** C1b	**Observation #:** 01
P-02-0003, Rev. 2/Attachment 1 (Initiating Event Notebook)/Section 2.14.1 and Section 2.14.3.2 use recovery actions. However, there is no justification in terms of identifying procedures or training materials.		
Level of Significance: Finding	**Related SRs:** IE-C1b	
Basis for Significance:		
No justification was provided for the identified recovery actions.		
Possible Resolution:		
Identify procedure, training material, or other justification for the cited recovery actions.		
EPU Impact Review:		
It should be noted that the above recovery actions are not used to quantitatively screen initiators. The recoveries discussed are used to provide additional qualitative grouping justification for initiators. Section 2.14.1 - Spurious Low Pressure Signal This event is grouped into a plant trip; an evaluation has been performed in the simulator to show that an Engineered Safeguards signal will not be received. Further ,this event was included in early Babcock & Wilcox Owners Group PRA studies, but has been removed from more recent ones. This initiating event can be grouped with a reactor trip without credit for operator action. It should be noted that operator action was only used as additional justification to group this initiating event with a reactor/turbine trip. This initiating event was not screened out. Section 2.14.3.2 - Loss of Offsite Power to 230kV Switchyard Without Station Blackout The fault tree for loss of the 230kV switchyard only credits auto start and loading of Emergency Diesel Generators (EDGs). The plant will not trip, because cooling water and turbine systems are powered from the 500kV switchyard. The 230kV fault tree results in a loss of offsite power event if the 230kV switchyard fails and the EDGs fail to auto start and supply the safety bus. There is no direct recovery action required by the operators. Documentation needs to be enhanced to better describe the grouping and use of the initiating event above to make the use and application clear. This finding appears to be an issue with documentation, and therefore is not expected to impact the results for the EPU application.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS

Element: IE	**Supporting Requirement:** D2	**Observation #:** 01

Some instances of missing documentation were identified.

(1) P-02-0003, Rev. 2, Section 1.3 identifies a list of plant systems that were reviewed for additional potential initiating event impacts; there were three that were considered. This process satisfies the intent of SR IE-A3, however, there is no documentation to indicate what process was used to identify these three items.

(2) The documentation is not clear on the definitions of IE_T10 and IE_T11.

Level of Significance: Finding **Related SRs:** IE-D2

Basis for Significance:

(1) Documentation does not permit a peer reviewer or PRA analyst to understand or duplicate the process used to identify the three items reviewed in Section 1.3.

(2) P-02-0003, Rev. 2/Attachment 1 (Initiating Event Notebook)/Section 2.4.1 identifies IE_T10 as loss of raw water pumps. Section 2.4.3 also identifies IE_T10 as loss of service water. Section 2.4.5 identifies IE_T11 as loss of intake structure. Table 6 identifies IE_T10 as loss of service water and IE_T11 as loss of raw water, but the frequency method refers to Reference 20 (Intake Structure Analysis from ERIN). Section 3.3.2.1 discusses the frequency determination of IE_T10 (loss of service water) and Section 3.3.2.2 discusses IE_T11 (loss of intake structure). Noting that the raw water system is different from the service water system, this is confusing and a potential initiating event (loss of raw water) may not be modeled.

Possible Resolution:

(1) Provide the process used to identify the three items in Section 1.3 in the documentation.

(2) Clarify the documentation.

EPU Impact Review:

(1) The system analysis is documented in the initial issuance of the initiating event calc. The plant systems were categorized by power conversion systems and front line systems or support systems. The initial issuance of the initiating event calc documents the systems that were reviewed and a later revision of the calc dropped the documentation.
(2) The Raw Water-Service Water System includes one normally running pump and two safety-related standby pumps. A loss of the normally running pump and failure of the two standby pumps to start would result in a manual trip per AP-330, "Loss of Nuclear Services Cooling," and would eventually cause an automatic reactor trip following a RCP trip. A loss of Raw Water is included as a contributor to the loss of Service Water Initiator and is modeled as an input to initiator IE_T10. The PRA model is correct; this is a documentation only issue.

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Element: QU	**Supporting Requirement:** E4	**Observation #:** 01
The sensitivity analyses included in Quantification Calc P-09-0001 Ref 0 are provided to defend the low quantification results and deal with the impact of some important and unique features of the CR-3 design, but do not address the uncertainties and assumptions associated with the model.		
Level of Significance: Finding	**Related SRs:** QU-E4, LE-F2	
Basis for Significance:		
The sensitivities required by the Supporting Requirement are not provided.		
Possible Resolution:		
From the uncertainty analysis and the modeling assumptions, identify issues that could impact the results of the PRA and perform sensitivities to determine the potential impact of them.		
EPU Impact Review:		
NUREG 1.200 does not require sensitivities or uncertainties to be quantified, only identified. Calculation P10-0001 contains sensitivities made for the EPU project.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Element: QU	**Supporting Requirement:** F5	**Observation #:** 01
Reviewed Quantification Calc P-09-0001, Rev. 0 and did not see a discussion of the limitations of the quantification processes in the calc.		
Level of Significance: Finding	**Related SRs:** QU-F5, LE-G5	
Basis for Significance:		
Requirement of Supporting Requirement F5.		
Possible Resolution:		
Add a section discussion limitations of the quantification process that would impact applications in the calc.		
EPU Impact Review:		
There were no significant limitations of the quantification process identified with the EPU analysis; therefore, this finding does not impact the EPU application. Complete designs for all EPU plant changes have not been issued or installed; the PRA model is limited to the designs as they are documented at the time of the submittal. The major dependencies, such as power and air, have been included in the model. No significant change to risk insight is expected based upon final implemented designs.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Element: LE	**Supporting Requirement:** B1	**Observation #:** 01
Containment venting should be explicitly addressed as a potential large early release frequency (LERF) issue.		
Level of Significance: Finding	**Related SRs:** LE-B1	
Basis for Significance:		
Standard requires consideration of containment venting as a LERF contributor per Table 4.5.9-3 of the standard.		
Possible Resolution:		
Provide detailed basis for screening of containment venting as a potential containment isolation failure mode.		
EPU Impact Review:		
The resolution of this F&O is independent of the EPU. The consideration of the vent path as a potential containment isolation failure mode should be addressed for the current model. No significant EPU impact is anticipated. Additionally, containment venting is not expected to be a LERF contributor, but may contribute to late releases.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Element: LE	**Supporting Requirement:** C2a	**Observation #:** 01
Operator actions to depressurize RCS for In Vessel Recovery are quantified conservatively (see P02-0012 Section 4.2.3). Other potential operator actions are not credited.		
Level of Significance: Finding	**Related SRs:** LE-C2a	
Basis for Significance:		
Realistic treatment of operator actions is required for Capability Category II.		
Possible Resolution:		
Model operator actions realistically with an appropriate Human Reliability Analysis (HRA).		
EPU Impact Review:		
Providing more realistic, i.e., lower HRA value, than the conservative screen value is not expected to have any increasing impact on the LERF value, therefore not a significant impact to EPU.		

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Open Facts & Observations Potentially Impacting EPU

FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Element: LE	**Supporting Requirement:** C2b	**Observation #:** 01
Repair of equipment not incorporated or discussed.		
Level of Significance: Finding	**Related SRs:** LE-C2b	
Basis for Significance:		
Consideration of repair for significant sequences required for Capability Category II.		
Possible Resolution:		
Review significant EPU accident progression sequences to see if repair can be credited, and justify any credit taken.		
EPU Impact Review:		
This is a documentation issue. CR-3 has not credited repair of equipment beyond off-site power restoration. The review of sequences and discussion of repair will be documented in future revisions. There is no impact on EPU due to this being a documentation only issue.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Element: LE	**Supporting Requirement:** C3	**Observation #:** 01
The Level 2 model does not credit potential scrubbing for steam generator tube rupture (SGTR) sequences as discussed in the report.		
Level of Significance: Finding	**Related SRs:** LE-C10, LE-D4	
Basis for Significance:		
Capability Category II requires consideration of mitigating operator actions, fission product scrubbing, and beneficial failures. Note that credit for scrubbing may be difficult and it is not unusual to assume SGTR sequences are large early releases.		
Possible Resolution:		
For Capability Category II, incorporate realistic modeling of SGTR sequences to determine if any accident progressions may not lead to large early releases due to for example, scrubbing, delayed releases, or cycling secondary relief valves.		
EPU Impact Review:		
The cost and additional conservative removal from the PRA is not required for the EPU submittal. The determination of scrubbing in a OTSG is subject of great uncertainty and is highly dependent upon the rupture location and SG inventory. Thus the current PRA assumption for no scrubbing in a SGTR is acceptable for both the base model as well as EPU. Progress Energy does not have any plans to upgrade the Supporting Requirement to Capability Category II. Realistic modeling of SGTR sequences (e.g., scrubbing) could be reduce the LERF contribution of SGTR core damage sequences leading to LERF. Thus Capability Category I is acceptable for this Supporting Requirement.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS

Element: LE	**Supporting Requirement:** C8b	**Observation #:** 01

No evidence of a review of significant accident progression sequences for potential LERF reduction.

Level of Significance: Finding **Related SRs:** LE-C8b

Basis for Significance:

Capability Category II requires such review.

Possible Resolution:

Perform review of significant sequences to see if continued equipment operation or operator actions beyond design environmental conditions could reduce LERF.

EPU Impact Review:

This is a documentation issue. CR-3 has actively looked for a method to reduce both CDF and LERF by removal of both model conservatism and plant changes (mods and procedures). Progress Energy will document the review of LERF accident sequences in future revisions. There is no impact on EPU due to this being a documentation only issue.

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CR-3 ASME 2009 Focused Peer Review
Open Facts & Observations Potentially Impacting EPU

FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Element: LE	**Supporting Requirement:** C9a	**Observation #:** 01
No credit is taken for system operation or operator actions after containment failure.		
Level of Significance: Finding	**Related SRs:** LE-C9a	
Basis for Significance:		
Capability Category II requires justification of any credit for system operation or operator actions after containment failure. Note that Capability Category I is not unusual for this Supporting Requirement.		
Possible Resolution:		
If available, identify and justify potential credit for equipment operation or human actions after containment failure.		
EPU Impact Review:		
CR-3 LERF is dominated by SGTR and thus Containment Failure has almost no contribution to LERF and the crediting of equipment or operator actions after containment failure will have no meaningful impact on the CR-3 LERF results. There is no impact on EPU due to this being a documentation only issue.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Element: LE	**Supporting Requirement:** C9b	**Observation #:** 01
There was no evidence of a review of significant accident progression sequences for potential LERF reduction through continued equipment operation or operator actions after containment failure.		
Level of Significance: Finding	**Related SRs:** LE-C9b	
Basis for Significance:		
Review required for Capability Category II.		
Possible Resolution:		
Perform review of significant sequences to see if continued equipment operation or operator actions after containment failure could reduce LERF.		
EPU Impact Review:		
CR-3 LERF is dominated by SGTR and thus Containment Failure has almost no contribution to LERF and the crediting of equipment or operator actions after containment failure will have no meaningful impact on the CR-3 LERF results. There is no impact on EPU due to this being a documentation only issue.		

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<table>
<tr><td colspan="3">FACT/OBSERVATION REGARDING PRA
TECHNICAL ELEMENTS</td></tr>
<tr><td>Element: LE</td><td>Supporting Requirement: C10</td><td>Observation #: 01</td></tr>
<tr><td colspan="3">Containment bypass analysis should be performed realistically with any credit for scrubbing justified. Assessment of SGTR events assumes that all SGTRs lead to containment bypass with no credit for scrubbing, a conservative assumption. Assessment of interfacing system LOCA (ISLOCA) is more realistic and justifies the lack of credit for scrubbing</td></tr>
<tr><td colspan="3">Level of Significance: Finding   Related SRs: LE-C3, LE-D</td></tr>
<tr><td colspan="3">Basis for Significance:</td></tr>
<tr><td colspan="3">Lack of credit for scrubbing defines Capability Category I. Capability Category II requires a realistic analysis with any credit justified.</td></tr>
<tr><td colspan="3">Possible Resolution:</td></tr>
<tr><td colspan="3">Perform and document a more realistic assessment of SGTR accident progression to include potential effects of scrubbing.</td></tr>
<tr><td colspan="3">EPU Impact Review:</td></tr>
<tr><td colspan="3">The cost and additional conservative removal from the PRA is not required for the EPU submittal. The determination of scrubbing in a OTSG is subject of great uncertainty and is highly dependent upon the rupture location and SG inventory. Thus, the current PRA assumption for no scrubbing in a SGTR is acceptable for both the base model as well as EPU. Progress Energy does not have any plans to upgrade the Supporting Requirement to Capability Category II. Capability Category I considered acceptable for this Supporting Requirement.</td></tr>
</table>

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Element: LE	**Supporting Requirement:** D4	**Observation #:** 01
Assessment of SGTR events conservatively groups all SGTR sequences as LERF with a stuck-open secondary side relief valve. A realistic analysis should be performed.		
Level of Significance: Finding	**Related SRs:** LE-C3, LE-C10	
Basis for Significance:		
Capability Category I allows a conservative evaluation. Capability Category II requires a realistic secondary side isolation capability analysis for significant sequences.		
Possible Resolution:		
Include details to differentiate SGTR sequences with stuck-open secondary side relief valves from those with cycling or closed relief valves.		
EPU Impact Review:		
The cost and additional conservative removal from the PRA is not required for the EPU submittal. The determination of the number of times a SG relief valve may stick open is subject of great uncertainty and thus the potential failure of a SG relief valve would also be uncertain. It is the judgment of Progress Energy that the increase number of cycles of a relief valve has some increase potential for setpoint drift and other mechanisms that could cause more Containment Bypass leakage. Thus, the current PRA assumption for stuck open relief valves for both the base model as well as EPU is acceptable. Progress Energy does not have any plans to upgrade the Supporting Requirement to Capability Category II.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS

Element: LE	**Supporting Requirement:** D5	**Observation #:** 01

The approach taken for consideration of induced SGTR was the application of a draft generic template from Babcock & Wilcox, according to P02-0012, Rev. 2. This template indicates that the contribution to LERF is very small (~2E-09). This is a conservative estimate, per the discussion, so it must be considered Capability Category I but the small contribution would only become smaller if a more realistic approach were applied. The SR indicates the conservative assessment is considered Capability Category I.

Level of Significance: Finding **Related SRs:** LE-D5

Basis for Significance:

The Supporting Requirement indicates that a conservative assessment must be considered Capability Category I, even though the contribution to overall LERF is very small.

Possible Resolution:

To achieve Capability Category II, a more realistic assessment would be necessary. However, due to the small potential impact on overall LERF, it may not be desirable to expend the resources to develop a realistic analysis.

EPU Impact Review:

Capability Category I is acceptable for this Supporting Requirement. The further decrease in LERF due to induced SGTR is very expensive and Progress Energy does not believe EPU requires Capability Category II for this element and does not have any other risk application that needs to obtain Capability Category II for this Supporting Requirement.

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS

Element: LE	**Supporting Requirement:** E4	**Observation #:** 01

Quantification of LERF should report a mean LERF (via uncertainty calculation) and include a convergence test to show the appropriate selection of the truncation limit.

Level of Significance: Finding **Related SRs:** LE-F2

Basis for Significance:

While most of the applicable requirements of ASME Table 4.5.8-2(a)-(c) are met, two items are not addressed.

Possible Resolution:

As was performed for the CDF quantification, an uncertainty calculation should be performed to allow reporting of a mean LERF rather than a point estimate, and an independent convergence test should be performed to show that the same truncation level is appropriate for LERF as used for CDF.

EPU Impact Review:

This has no impact on EPU. The CR-3 LERF convergence point has been selected by using 1 decade below CDF convergence. The convergence point for LERF will be further evaluated in the next CR-3 Model of Record. There is no impact on EPU due to this being a documentation only issue.

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Element: LE	**Supporting Requirement:** F2	**Observation #:** 01
To meet Capability Category II for this Supporting Requirement, it is necessary to conduct sensitivity analyses on issues identified as uncertainties in the LERF analysis. Uncertainties have been identified in P057607008-2865, but sensitivity analyses have not been performed on this issue.		
Level of Significance: Finding	**Related SRs:** LE-F2	
Basis for Significance:		
Attaining Capability Category II requires that sensitivity analyses be performed on the areas identified as uncertain.		
Possible Resolution:		
Conduct sensitivity analyses on uncertainties. Since the uncertainties have been ranked qualitatively with respect to impact, it is recommended that this ranking process be integrated into a process for identification of items to be selected for performance of sensitivities.		
EPU Impact Review:		
Progress Energy disagrees with this Finding: the NRC has stated the expectation of uncertainty sensitivity is driven by the application rather than the baseline Model. Progress Energy will perform LERF sensitivities after expectations are clarified further by the NRC. The lack of sensitivities for LERF has no impact on EPU results.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Element: LE	**Supporting Requirement:** F3	**Observation #:** 01
The contributors to LERF and LERF uncertainties are to be presented similarly to the presentation for CDF. Additional detail is needed for the LERF results (e.g., operator contributions). Additionally, an estimation of uncertainty interval is needed to satisfy this Supporting Requirement.		
Level of Significance: Finding	**Related SRs:** LE-F3	
Basis for Significance:		
The items indicated in the assessment are specified in the Supporting Requirement as required to satisfy the Supporting Requirement.		
Possible Resolution:		
Provide the needed information as specified in the Supporting Requirement. It is suggested that a parallel presentation of the material be provided for both CDF results and LERF results for ease of review.		
EPU Impact Review:		
There is no impact on EPU. This Supporting Requirement is documentation of the uncertainty interval for LERF. The CR-3 LERF is mostly driven by SG tube rupture and thus the CDF uncertainty can provide much of the insights for LERF uncertainty. Resolution of this F&O is applicable to the current model and independent of the EPU		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS

Element: LE	**Supporting Requirement:** G3	**Observation #:** 01

The documentation should include relative contributors to LERF according to plant damage state, phenomena, containment challenges, and containment failure modes.

Level of Significance: Finding **Related SRs:** LE-G3

Basis for Significance:

Capability Category I is met, but for Capability Category II, relative contributors by these categories should be clearly documented.

Possible Resolution:

Provide the relative contributions to LERF by Plant Damage State, key Level 2 phenomena, and containment failure modes.

EPU Impact Review:

Resolution of a documentation only F&O is applicable to the current model and independent of the EPU.

Appendix 3

CR-3 ASME Fire 2009 Peer Review
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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Observation: UNC-A1-01	**Technical Element:** UNC	**Supporting Requirement:** A1
The uncertainty analysis of the total CDF and LERF results, either by estimation or parameter uncertainty propagation, was not performed. Basis for Significance: SR UNC-A1 requires performing the uncertainty analysis in accordance with HLR-QU-E in Part 2. QU-E3 requires an uncertainty evaluation which accounts for parameter uncertainties. Possible Resolution: Propagate parameter uncertainties through the fire PRA model. This may require additional software or software modifications to perform the analysis.		
Level of Significance: Finding		
Resolution:		
The current quantification tool (FRANC) does not provide results in a format that support performing this evaluation. The numerical uncertainties have not been propagated for the CR-3 fire PRA. This, however, does not impact the risk insights related to the acceptability of this license amendment request because the numerical uncertainties are determined to be bounded by the conservative assumptions regarding the epistemic uncertainties associated with the source fires and damage sets. Most applications are not dependent of this evaluation for success. Parametric uncertainty only addresses numerical uncertainties. Sensitivity studies usually provide much more useful insights for application uncertainties.		
EPU Impact Review:		
Uncertainty analysis will not affect the CDF or LERF value, therefore does not affect the EPU.		

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FACT/OBSERVATION REGARDING PRA TECHNICAL ELEMENTS		
Observation: CS-B1-01	**Technical Element:** CS	**Supporting Requirement:** B1
The Supporting Requirement calls for a review of existing electrical overcurrent coordination and protection analysis to identify any additional circuits and cables whose failure could challenge power supply availability due to inadequate or unanalyzed electrical overcurrent protective device coordination. According to the road map, this task is not yet complete. Basis for Significance: This is required to be done to meet the Supporting Requirement. Possible Resolution: Review the overcurrent and coordination studies and identify additional cables as required per the Supporting Requirement.		
Level of Significance: Finding		
Resolution:		
A review of all power supplies credited in the fire PRA model was conducted, and it was determined that in some cases the power cables for non-credited loads had not been identified in the circuit analysis. If the components control circuit is affected by the fire, a fault on these load cables may not be cleared by the component breaker. This could lead to tripping of the upstream breaker and subsequent loss of the credited power supply. These load cables were added to the applicable switchgear as required cables in Fire Safe Shutdown Program Manager Data Base via Change Package CR3-0207.		
EPU Impact Review:		
The fire PRA model will include coordination information in the next fire PRA quantification update. The delta in fire core damage frequency should be insignificant because coordination information would be added to both the pre and post PRA models.		