**Progress Energy**

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

Ref: 10 CFR 54

December 18, 2009
3F1209-04

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

Subject: Crystal River Unit 3 – Response to Follow-up to Progress Energy RAI Responses on CR-3 SAMA Evaluation (TAC NO. ME0278)

References:

(1) CR-3 to NRC letter dated December 16, 2008, "Crystal River Unit 3 – Application for Renewal of Operating License"

(2) NRC to CR-3 letter dated August 10, 2009, "Request for Additional Information Regarding Severe Accident Mitigation Alternatives for Crystal River Unit 3 Nuclear Generating Plant License Renewal Application (TAC NO. ME0278)"

(3) CR-3 to NRC letter dated October 9, 2009, "Crystal River Unit 3 – Response to Request for Additional Information Regarding Severe Accident Mitigation Alternatives for Crystal River Unit 3 Nuclear Generating Plant License Renewal Application (TAC NO. ME0278)"

(4) NRC to CR-3 Electronic Mail dated November 2, 2009, "Follow-up to Progress Energy RAI Responses on CR-3 SAMA Evaluation"

Dear Sir:

On December 16, 2008, Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc. (PEF), requested renewal of the operating license for Crystal River Unit 3 (CR-3) to extend the term of its operating license an additional 20 years beyond the current expiration date (Reference 1). Subsequently, the Nuclear Regulatory Commission (NRC), by letter dated August 10, 2009, provided a request for additional information (RAI) concerning the CR-3 License Renewal Application (Reference 2). CR-3 responded to the RAI by letter dated October 9, 2009 (Reference 3). By electronic mail received on November 2, 2009 (Reference 4), the NRC requested a follow-up response to information provided in Reference 3. The Enclosure to this letter provides the response to Reference 4.

No new regulatory commitments are contained in this submittal.

If you have any questions regarding this submittal, please contact Mr. Mike Heath, Supervisor, License Renewal, at (910) 457-3487, e-mail at mike.heath@pgnmail.com.

Sincerely,

Jon A. Franke
Vice President
Crystal River Unit 3

JAF/dwh

Enclosure: Response to Request for Additional Information

xc: NRC CR-3 Project Manager
NRC License Renewal Project Manager
NRC Regional Administrator, Region II
Senior Resident Inspector

**Progress Energy Florida, Inc.**
Crystal River Nuclear Plant
15760 W. Power Line Street
Crystal River, FL 34428

A140
NRR

**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 18th day of December, 2009, by Jon A. Franke.

Carolyn E Portmann

Signature of Notary Public
State of Florida

CAROLYN E. PORTMANN
Notary Public - State of Florida
My Commission Expires Mar 1, 2010
Commission # DD 524380
Bonded By National Notary Assn.

(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

## ENCLOSURE

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

## RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

**RAI 1.c**

Progress Energy did not characterize the significant review comments from the full-scope PRA self-assessment or the limited-scope peer review cited in the RAI response. Without providing any details, Progress Energy concluded that the findings from these two review "should not have a significant impact on the results of the SAMA analysis." Provide the information requested in the last sentence of this RAI relative to these two reviews (i.e., "Describe any significant review comments, their resolution, and the potential impact of any unresolved comments on the results of the SAMA analysis").

***Response***

*The following section lists the significant comments from self-assessments and peer reviews on the Crystal River Unit 3 (CR-3) Probabilistic Risk Assessment (PRA) model since the Model of Record (MOR) 2006 was issued. The term "significant" for this response is defined as having any change to the PRA model or results. The non-significant comments were limited to documentation improvements. The review comments that affect the model do not have a significant impact to the issued PRA model. The following discussion gives each comment and its impact to the MOR 2006 PRA model which was the one used for the SAMA analysis. For the three findings described below, there was little impact on the SAMA analysis such that a potential plant enhancement could have been masked by improper modeling. That is, the MOR 2006 adequately captured an appropriate importance listing for the modeled basic events, and also was able to satisfactorily calculate an averted cost-risk so that a reasonable estimate could be made as to whether a particular SAMA enhancement was cost beneficial. Additionally, in Section E.7.2.3 of the Environmental Report (ER), each of the quantified SAMAs was reevaluated at the 95$^{th}$ percentile for averted cost, which was meant to account for general uncertainty in the overall PRA model so as to bound the cost benefit for each of the identified SAMAs.*

***2007 Significant Self Assessment Items:***

***Fact and Observation (FnO)-AS-B6***

*Significant Comment*

*The recent addition of the non-safety Emergency Diesel Generator (EDG) has not been included in the Offsite Power (OSP) recovery evaluation.*

*Resolution*

*The OSP recovery evaluation was included in the 2008 draft model which includes credit for the non-safety EDG.*

*Impact*

*The impact to the MOR 2006 model was not significant because the effect of the addition of the non-safety EDG to the OSP recovery reduced the overall EDG system contribution to Core Damage Frequency (CDF).*

***FnO-HR-G4-2***

*Significant Comment*

*The accident sequence timing for Human Error Probability (HEP) in Appendix B of calculation P-02-0006 is not referenced to the latest Modular Accident Analysis Program (MAAP) analysis. The timing is also not related to a specific sequence. The result is some suspect timing and HEPs. Examples:*

> *QHUEFP1Y: Fail to start Emergency Feedwater Pump EFP-1. The available time is 60 minutes, based on Core Damage (CD) at 90 minutes. Success criteria in RCS-01-61 show the need to go to feed and bleed at 30 minutes. If no action occurs at 30 minutes, then CD results at 45 minutes, assuming that the Reactor Coolant Pump (RCP) is on. If the RCP is off, action must occur by 60 minutes.*
>
> *QHUEFW9Y: Human Failure Event for raising the water level in the Once-Through Steam Generator (OTSG). Timeline suggests this is used for a Small Break Loss of Coolant Accident (SLOCA). Thermal-hydraulic reference is for a Station Blackout (SBO) sequence. Not clear if the reference applies to a SLOCA.*
>
> *QHUMSIVY: Isolate Main Steam Isolation Valve. The timing reference is for SBO and states that 60 minutes is allowed for recovery. Sixty minutes is for RCP's not operating. If RCPs operate, the time to CD is 30 minutes. There is not enough documentation in the worksheet to know if this event is only used in sequences with the RCPs tripped.*
>
> *QHUFWPTY: Uses 60 minutes for time to recover. Reference for 60 minutes is from the SBO analysis. Not clear if this event only applies to SBO analysis.*
>
> *RHUHPRRY: Initiate High Pressure Recirculation following a Steam Generator Tube Rupture (SGTR). The event does not make sense. After Refueling Water Storage Tank drains for SGTR, there is no inventory in sump to switch to.*

*Resolution*

*The MAAP timing analysis for the human events has been used to calculate the Human Reliability Analysis (HRA) timelines and included in the draft model 2009.*

*Impact*

*The impact to the MOR 2006 is not considered significant because in both the MOR 2006 model and draft model 2009, the operator actions account for a large portion of the overall CDF.*

***FnO-DA-C1***

*Significant Comment*

*The frequency of Loss of Offsite Power (LOSP, LOOP) is developed through Bayesian update of generic data with plant specific historical events. This is developed through screening of historical data, rather than using generic data verbatim. The Offsite Power (OSP) historical data is screened to remove events that "are not typical" to the CR-3 site. The screening process and rules are not substantiated. Grid and weather events at plants with less robust OSP connections than CR-3 were discarded as applicable failures, but the operating time for these plants was included in the plant population. The resulting generic initiating event frequency of 6.6E-3 conflicts with several NUREG's on LOSP. The result of these two practices is that the plant-specific data for LOSP is reduced by a factor of 10, with no justification as to why the methods and data are allowable for CR-3.*

*Resolution*

*The 2008 model revision no longer incorporates a screening approach for LOOP data. The generic LOOP event frequencies are directly taken from NUREG/CR-6890, "Reevaluation of Station Blackout Risk at Nuclear Power Plants." These values are Bayesian updated against plant experience (or grid region experience). As part of this update, the four categories of LOOP events were also adopted.*

*Impact*

*The LOSP analysis was revised to include the four categories of LOOP events. The LOSP contribution slightly decreased from 6% of CDF to 4% of CDF which was a small impact to the model and, therefore, not considered a significant impact to the MOR 2006 model.*

***2009 Significant Items from Focused PEER Review:***

*The Focused PEER review was conducted against the 2008 PRA model. There were no FnO's from the 2009 focused PEER review that required changes to the model or changes to the results.*

**RAI 2.e.i**

Progress Energy did not identify or describe the Plant Damage States (PDS) from the Level 1 PSA or provide their frequency. As recommended by NEI 05-01, "provide a table or matrix describing the mapping of Level 1 accident sequences into Level 2 release categories."

***Response***

*The Level 1 accident sequences are grouped into core damage bins (CDB) based upon conditions presented in the following table.*

### *Table 1 - Mapping of Accident Sequences to Core Damage Bins*

| Accident Sequence | CDB(s) | RCS leakage rate | Timing of RPV failure | SSHR* status | RCS pressure at RPV failure | Cavity status at RPV failure |
|---|---|---|---|---|---|---|
| TBQU | 1 | Small LOCA | Early | Failed | High | Dry[1] |
| TKBQU | 1 | Small LOCA | Early | Failed | High | Dry[1] |
| TKBU | 1 | Small LOCA | Early | Failed | High | Dry[1] |
| TBL1U | 2 | Cycling Relief | Early | Failed | High | Dry[1] |
| SU | 3 | Small LOCA | Late | Available | High | Dry[1] |
| TKU | 3 | Small LOCA | Late | Available | High | Dry[1] |
| TQU | 3 | Small LOCA | Late | Available | High | Dry[1] |
| SBP | 4 | Small LOCA | Early | Failed | High | Wet |
| SX | 5 | Small LOCA | Late | Available | High | Wet |
| TBL1WX | 5 | Small LOCA | Late | Available | High | Wet |
| TBQX | 5 | Small LOCA | Late | Available | High | Wet |
| TKBQX | 5 | Small LOCA | Late | Available | High | Wet |
| TQX | 5 | Small LOCA | Late | Available | High | Wet |
| SBX | 6 | Small LOCA | Late | Failed | High | Wet |
| TBL1L2X | 6 | Small LOCA | Late | Failed | High | Wet |
| AU | 7 | Large LOCA | Early | N/A | Low | Dry[1] |
| RV | 7 | Large LOCA | Early | N/A | Low | Dry[1] |
| TBP | 7 | Large LOCA | Early | Failed | Low | Dry[1] |
| TKB1B2/(TKBL) | 7 | Large LOCA | Early | Failed | Low | Dry[1] |
| TKBM | 7 | Large LOCA | Early | Failed | Low | Dry[1] |
| TKBP | 7 | Large LOCA | Early | Failed | Low | Dry[1] |
| MU | 9 | Medium LOCA | Early | N/A | Medium | Dry[1] |
| AX | 11 | Large LOCA | Early | N/A | Low | Wet |
| MX | 12 | Medium LOCA | Late | N/A | Medium | Wet |
| RCQGY | 18 | Small LOCA (Bypass) | Late | Available | Medium | Dry |
| RQGY | 18 | Small LOCA (Bypass) | Late | Available | Medium | Dry |
| RUG | 18 | Small LOCA (Bypass) | Late | Available | Medium | Dry |
| RUQ | 18 | Small LOCA (Bypass) | Late | Available | Medium | Dry |
| RBQY | 19 | Small LOCA (Bypass) | Late | Failed | High | Dry |
| RUC2 | 20 | Small LOCA (Bypass) | Late | Available | High | Dry |
| RBQX | 21 | Small LOCA (Bypass) | Late | Failed | High | Wet |
| RBX | 21 | Small LOCA (Bypass) | Late | Failed | High | Wet |
| RBP | 22 | Small LOCA (Bypass) | Early | Failed | High | Dry |
| RCP | 22 | Small LOCA (Bypass) | Early | Failed[2] | High | Dry |
| RUB | 22 | Small LOCA (Bypass) | Early | Failed | High | Dry |
| ISLOCA | 23 | Large LOCA (Bypass) | Early | N/A | Low | Dry |

** SSHR - Secondary-Side Heat Removal*
*1. Assumes no reactor building spray.*
*2. SSHR available to intact OTSG, but unavailable for cooldown.*

*The Level 2 fault tree determines the Containment Safeguards Event Tree (CSET) End State. The list of CSET End States is listed in Table 2 below. The CDBs are combined with the CSET to produce the PDS. The output of the quantification of the Level 2 fault tree provides the frequencies for each PDS.*

***Table 2 - Containment Safeguards Event Tree***

| ***CSET End State*** | ***Description*** |
|---|---|
| *A* | *Successful isolation, sprays and fans functioning* |
| *B* | *Small isolation failure, sprays and fans functioning* |
| *C* | *Large isolation failure, sprays and fans functioning* |
| *D* | *Successful isolation, sprays fail in recirculation and fans functioning* |
| *E* | *Small isolation failure, sprays fail in recirculation and fans functioning* |
| *F* | *Large isolation failure, sprays fail in recirculation and fans functioning* |
| *G* | *Successful isolation, sprays functioning and fans fail* |
| *H* | *Small isolation failure, sprays functioning and fans failed* |
| *I* | *Large isolation failure, sprays functioning and fans failed* |
| *J* | *Successful isolation, sprays fail in recirculation and fans failed* |
| *K* | *Small isolation failure, sprays fail in recirculation and fans fail* |
| *L* | *Large isolation failure, sprays fail in recirculation and fans fail* |
| *M* | *Successful isolation, sprays fail in injection and fans succeed* |
| *N* | *Small isolation failure, sprays fail in injection and fans succeed* |
| *O* | *Large isolation failure, sprays fail in injection and fans succeed* |
| *P* | *Successful isolation, sprays fail in injection and fans failed* |
| *Q* | *Small isolation failure, sprays fail in injection and fans fail* |
| *R* | *Large isolation failure, sprays fail in injection and fans fail* |
| *S* | *Containment bypass* |

**Table 3 - CSET Definition**

| Identifier | Description | Definition |
|---|---|---|
| IC-1 | Containment Intact | This release category represents an accident sequence in which the containment is intact. The source term for this type of sequence is very small and limited to the containment design leakage rate. |
| RC-1 | Release Category 1 | This release category is a late containment failure caused by gradual overpressurization. The core debris is assumed to be coolable. This type of gradual pressure increase is assumed to result in a benign containment failure and the duration of the release could be over a long period of time. Either the containment sprays or a pool of water over the core debris scrubs the release from the containment. |
| RC-1A | Release Category 1A | This release category is similar to RC-1 except that re-vaporization occurs. Re-vaporization is caused by the self-heating of radionuclides plated out on the Reactor Coolant System (RCS), becoming re-suspended in the containment atmosphere. This re-vaporization is postulated to occur late in the accident sequence after the containment has failed. This allows the radionuclides to be released from the containment after only a limited holdup time. The impact of re-vaporization on the source terms is to increase the contribution of volatile radionuclides to the source term. |
| RC-1B | Release Category 1B | This release category is similar to RC-1 except that no scrubbing by containment sprays and/or water pools is available. If containment sprays function, or the Borated Water Storage Tank (BWST) inventory is otherwise dumped into containment, then both debris cooling and scrubbing will be attained (unless debris uncoolability is assumed). This can be assumed because for the CR-3 containment, when the BWST is discharged, the water level reaches several feet over the basemat (lower compartment), completely covering the debris bed for the duration of all applicable sequences studied. This category implies a debris bed that eventually dries up resulting in considerable core-concrete interaction (CCI). |
| RC-1BA | Release Category 1BA | This release category is similar to RC-1 except that both re-vaporization and no containment scrubbing are assumed to occur. |
| RC-2 | Release Category 2 | This release category represents a large early containment failure. The core debris is assumed to be coolable. The large failure significantly reduces radionuclide holdup time in the containment. The CR-3 specific liner failure releases are assumed to belong to this category. The release from the containment is scrubbed by containment spray operation at the time following fission product releases from the primary side. In this case, the releases will be driven by the prompt release of fission products at containment failure and the effect of re-vaporization, if any, should be small. Thus, release categories with re-vaporization will not be postulated for the large early containment failures. However, care will be taken when assigning source terms to pick a representative sequence for RC-2 (and RC-2B) that exhibits re-vaporization. |
| RC-2B | Release Category 2B | This release category is similar to RC-2 except that no scrubbing by containment sprays and/or water pools is assumed to happen. |
| RC-3 | Release Category 3 | This release category represents an early containment isolation failure with a small leakage rate (4 in. diameter). The core debris is assumed to be coolable. Either the containment sprays or a pool of water over the core debris scrubs the release from the containment. For the larger of the small leakage failures (i.e., close to 4 in. in diameter), the releases will be driven by the prompt release of fission products at containment failure and the effect of re-vaporization, if any, should be small. Smaller diameter isolation failures will result in reduced source terms due to the longer time available for natural removal mechanisms, such as gravity settling, to take place. Thus, release categories with re-vaporization will not be postulated for the small early containment failures. However, care will be taken when assigning source terms to pick a representative sequence for RC-3 (and RC-3B) that exhibits re-vaporization. |

| Identifier | Description | Definition |
|---|---|---|
| RC-3B | Release Category 3B | This release category is similar to RC-3 except that no scrubbing by containment sprays and/or water pools is assumed to happen. |
| RC-4 | Release Category 4 | This release category represents a containment bypass accident sequence with a small leakage rate. The leakage rate that would correspond to a SGTR sequence with cycling Safety-Relief Valves (SRVs), or an Interfacing System Loss of Coolant Accident (ISLOCA) in which operators react in time to mitigate effects by closing the valves on the Residual Heat Removal (RHR) suction line. The core debris is assumed to be coolable and releases from the containment scrubbed. Scrubbing by water in the faulted Steam Generator (SG) above the break is assumed to occur. Note that the operating procedures direct the operator to isolate the faulted SG. Thus, the faulted SG will be dry in the majority of the cases and no fission product scrubbing would occur. This category will be kept for future use (in case the procedures change), but for the purposes of this study, the unscrubbed source term (RC-4C) will be conservatively assigned to these low probability branches. |
| RC-4C | Release Category 4C | This release category is similar to RC-4 except that no scrubbing by water in the faulted SG above the break occurs. The core debris is assumed to be coolable and releases from the containment scrubbed. Note that a release category for no scrubbing by containment sprays and/or water pools is not postulated in this case. This is so because, for the bypass sequences, most of the release would be directly from the primary to the environment or the auxiliary building. Re-vaporization is also assumed to be negligible as compared to the direct releases. |
| RC-5 | Release Category 5 | The sequence represents a containment bypass accident with a large leakage rate. Such rate is representative of a SGTR accident with a stuck open SRV in the faulted SG, or an unmitigated ISLOCA accident. The core debris is assumed to be coolable and releases from the containment scrubbed. The releases from the faulted SG are assumed to be scrubbed by water above the break line. However the probability of scrubbed releases is very low due to the present procedures. Thus, similarly to RC-4, the unscrubbed source term (RC-5C) will be conservatively assigned to these low probability branches. |
| RC-5C | Release Category 5C | This release category is similar to RC-5 except that scrubbing by water in the faulted SG above the break occurs. The releases from the faulted SG are not scrubbed by water above the break line. The core debris is assumed to be coolable and releases from the containment scrubbed. |

*The Containment Release Category Fractions are obtained from containment failure event trees. Each PDS from the Level 2 quantification is multiplied by the Containment Release Category fractions in that row to obtain the frequency of containment release category. For example, the frequency of IC-1 release category from the Plant Damage State 1M (first row of Table 4) would be calculated as*

$$5.67E\text{-}12 * 9.89E\text{-}1 = 5.61E\text{-}12.$$

*Table 4 provides the mapping of Level 2 quantification cutset file results and does not include PDSs that did not appear in the cutset.*

## *Table 4 - Mapping of Quantification results to Containment Release Category*

| Level 2 Quantification results - Frequency of PDS | Level 1 accident sequences mapped to plant damage states | Containment Release Category Fraction | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | PDS | IC-1 | RC-1 | RC-1A | RC-1B | RC-1BA | RC-2 | RC-2B | RC-3 | RC-3B | RC-4 | RC-4C | RC-5 | RC-5C |
| 5.67E-12 | TBQU, TKBQU, TKBU | 1M | 9.89E-01 | 0 | 0 | 9.12E-03 | 1.01E-03 | 1.00E-03 | 1.83E-04 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1.06E-10 | TBL1U | 2M | 9.94E-01 | 0 | 0 | 4.55E-03 | 5.06E-04 | 3.29E-04 | 2.84E-04 | 0 | 0 | 0 | 0 | 0 | 0 |
| 9.29E-13 | TBL1U | 2O | 0 | 0 | 0 | 0 | 0 | 0 | 1.00E+00 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1.36E-11 | TBL1U | 2P | 7.46E-01 | 0 | 0 | 2.28E-01 | 2.53E-02 | 7.40E-04 | 1.43E-04 | 0 | 0 | 0 | 0 | 0 | 0 |
| 2.85E-12 | TBL1U | 2R | 0 | 0 | 0 | 0 | 0 | 0 | 1.00E+00 | 0 | 0 | 0 | 0 | 0 | 0 |
| 5.74E-12 | SU, TKU, TQU | 3M | 9.89E-01 | 0 | 0 | 1.01E-02 | 1.01E-05 | 1.15E-03 | 7.93E-05 | 0 | 0 | 0 | 0 | 0 | 0 |
| 2.67E-11 | SBP | 4M | 9.89E-01 | 8.15E-03 | 9.06E-04 | 9.93E-04 | 1.10E-04 | 0 | 1.23E-03 | 0 | 0 | 0 | 0 | 0 | 0 |
| 2.51E-10 | SX, TBL1WX, TBQX, TKBQX, TQX | 5M | 9.89E-01 | 9.21E-03 | 9.22E-06 | 9.27E-04 | 9.28E-07 | 0 | 1.19E-03 | 0 | 0 | 0 | 0 | 0 | 0 |
| 7.76E-13 | SX, TBL1WX, TBQX, TKBQX, TQX | 5P | 7.86E-01 | 1.93E-01 | 1.93E-04 | 1.94E-02 | 1.94E-05 | 0 | 1.81E-03 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1.09E-10 | SBX, TBL1L2X | 6M | 9.89E-01 | 8.15E-03 | 9.06E-04 | 9.93E-04 | 1.10E-04 | 0 | 1.23E-03 | 0 | 0 | 0 | 0 | 0 | 0 |
| 9.29E-13 | SBX, TBL1L2X | 6O | 0 | 0 | 0 | 0 | 0 | 8.91E-01 | 1.09E-01 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1.36E-11 | SBX, TBL1L2X | 6P | 7.85E-01 | 1.71E-01 | 1.89E-02 | 2.08E-02 | 2.31E-03 | 0 | 2.26E-03 | 0 | 0 | 0 | 0 | 0 | 0 |
| 2.85E-12 | SBX, TBL1L2X | 6R | 0 | 0 | 0 | 0 | 0 | 8.91E-01 | 1.09E-01 | 0 | 0 | 0 | 0 | 0 | 0 |
| 2.39E-10 | AU, RV, TBP, TKB1B2/(TKBL), TKBM, TKBP | 7A | 9.89E-01 | 9.67E-03 | 9.76E-05 | 7.26E-04 | 7.34E-06 | 0 | 7.50E-04 | 0 | 0 | 0 | 0 | 0 | 0 |
| 5.79E-12 | AU, RV, TBP, TKB1B2/(TKBL), TKBM, TKBP | 7D | 9.89E-01 | 0 | 0 | 1.04E-02 | 1.05E-04 | 6.91E-04 | 5.89E-05 | 0 | 0 | 0 | 0 | 0 | 0 |
| 6.39E-11 | AU, RV, TBP, TKB1B2/(TKBL), TKBM, TKBP | 7M | 9.89E-01 | 0 | 0 | 1.04E-02 | 1.05E-04 | 6.91E-04 | 5.89E-05 | 0 | 0 | 0 | 0 | 0 | 0 |
| 2.86E-12 | MU | 9M | 9.88E-01 | 0 | 0 | 9.43E-03 | 1.05E-03 | 5.36E-04 | 5.95E-04 | 0 | 0 | 0 | 0 | 0 | 0 |
| 1.05E-11 | AX | 11A | 9.89E-01 | 9.67E-03 | 9.76E-05 | 7.26E-04 | 7.34E-06 | 0 | 7.50E-04 | 0 | 0 | 0 | 0 | 0 | 0 |
| 5.64E-11 | AX | 11D | 9.89E-01 | 9.67E-03 | 9.76E-05 | 7.26E-04 | 7.34E-06 | 0 | 7.50E-04 | 0 | 0 | 0 | 0 | 0 | 0 |
| 2.23E-12 | AX | 11M | 9.89E-01 | 9.67E-03 | 9.76E-05 | 7.26E-04 | 7.34E-06 | 0 | 7.50E-04 | 0 | 0 | 0 | 0 | 0 | 0 |
| 4.49E-11 | MX | 12M | 9.88E-01 | 4.71E-03 | 5.24E-04 | 4.71E-03 | 5.24E-04 | 0 | 1.19E-03 | 0 | 0 | 0 | 0 | 0 | 0 |
| 8.19E-09 | RCQGY, RQGY, RUG, RUQ | 18S | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1.00E+00 | 0 | 0 |
| 6.08E-11 | RUC2 | 20S | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1.00E+00 | 0 | 0 |
| 3.26E-13 | RBQX, RBX | 21S | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1.00E+00 | 0 | 0 |
| 3.09E-07 | RBP, RCP, RUB | 22S | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1.00E+00 | 0 | 0 |
| 5.14E-08 | ISLOCA | 23S | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1.00E+00 |

**RAI 3.b**

Progress Energy identified several fire protection related modifications made at CR-3 since the IPEEE, but did not identify the specific fire compartments where the modifications were made. Provide a separate accounting of the fire protection related enhancements for each of the dominant fire areas.

***Response***

*The following table correlates the risk significant zones per the original Individual Plant Examination - External Events (IPEEE) identified in the response to RAI 3.a from CR-3 to NRC letter dated October 9, 2009, "Crystal River Unit 3 - Response to Request for Additional Information Regarding Severe Accident Mitigation Alternatives for Crystal River Unit 3 Nuclear Generating Plant License Renewal Application (TAC NO. ME0278)," with fire protection (FP) related hardware modifications.*

| Zone | Description | FP-Related Modification(s) |
|---|---|---|
| CC-108-106 | BATTERY CHARGER ROOM 3A | Enhanced Emergency Lighting, Upgraded Fire Detectors |
| CC-108-108 | 4160V ES SWITCHGEAR BUS ROOM 3A | Enhanced Emergency Lighting, Upgraded Fire Detectors, Electrical Cable Re-Routing for DHV-42 (Reactor Building Sump to Decay Heat Pump 1A Suction Isolation Valve) |
| CC-108-107 | 4160V ES SWITCHGEAR BUS ROOM 3B | Upgraded Fire Detectors |
| CC-124-117 | 480V ES SWITCHGEAR BUS ROOM 3A | Upgraded Fire Detectors, Electrical Cable Re-Routing for DHV-42 (Reactor Building Sump to Decay Heat Pump 1A Suction Isolation Valve) |
| CC-108-105 | BATTERY CHARGER ROOM 35 | Enhanced Emergency Lighting, Upgraded Fire Detectors |
| CC-108-102 | HALLWAY AND REMOTE SHUTDOWN ROOM | Upgraded Fire Detectors |
| CC-124-111 | CRD & COMMUNICATION EQUIP ROOM | Enhanced Emergency Lighting, Upgraded Fire Detectors |
| CC-108-109 | INVERTER ROOM 38 | Enhanced Emergency Lighting, Upgraded Fire Detectors |
| CC-145-118B | CONTROL ROOM | Upgraded Fire Detectors |
| CC-134-118A | CABLE SPREADING ROOM | Upgraded Fire Detectors, Electrical Cable Re-Routing for BSV-3, -4 (Reactor Building Spray Header Inlet Isolation Valves) |

**RAI 4.a**

Progress Energy failed to provide the requested analysis of the impacts of the planned 20% extended power uprate on the SAMA analysis, despite the fact that this request is within the scope of NEI 05-01 (Section 8.1). Provide the requested analysis.

***Response***

*As the response to this follow-up RAI, a description of the changes to the PRA model, from the issued Model of Record 2006 to the current draft model, is presented below to gain insight to the effects that the Extended Power Uprate (EPU) will have on the PRA.*

*Model of Record MOR2006 - CDF = 4.99E-6 / LERF = 3.69E-7*

*Since the Model of Record 2006 was issued, the CR-3 model has undergone draft updates that have not yet been issued as a new model of record. These updates include closing gaps from the American Society of Mechanical Engineers (ASME) gap self-assessment and focused peer review. Also the PRA model was modified to include multiple spurious actuations as required for a fire PRA that meets the requirements of Regulatory Guide 1.200, Rev. 2. The following sections describe the changes that have been made and the affects to CDF and Large Early Release Frequency (LERF).*

*PSA Model 2008 Update - CDF = 3.78E-6 / LERF = 4.30E-7*

*The CR-3 Probabilistic Safety Assessment (PSA) model update 2008 was completed in February 2009. This revision was performed to incorporate the ASME gap self-assessment findings, update plant specific data, and incorporate fault tree logic to support the fire PRA model. The change in the model results are dominated primarily by the data update. The generic and plant specific data were both updated and the values generally trended downward (lower failure probabilities) due to improved industry and plant experience. For systems, the most significant change is that Decay Heat Removal (DH) System importance has decreased significantly due to the data update improving both the availability and the reliability of the DH system. A PRA model comment was resolved by including the Emergency Diesel Generator (EG) system where more credit was provided for the recently added non-safety diesel in LOSP recovery thereby reducing system contribution to CDF. Reactor Coolant (RC) system importance reduction was also primarily related to data improvement in valve reliability. The increase to LERF is due to the inclusion of the ISLOCA fault tree into the model, which was previously a point estimate; and SGTR contribution increased which translates directly to LERF.*

*PSA Model 2009 Update - CDF = 3.63E-6 / LERF = 1.82E-7*

*The CR-3 PSA model update 2009 was completed in July 2009. This revision was performed to incorporate the ASME gap self-assessment findings associated with the HRA and incorporate fault tree logic to support the fire PRA.*

*The noticeable effect on the PRA model results was in the decrease in LERF. The SGTR initiator decrease is related to HRA analysis improvements; this change has had a significant impact on reducing the Level 2 results. The safety Alternating Current (AC) bus initiators have been removed based on stable plant operation with the loss of a safety AC bus. The makeup contribution also decreased based upon HRA analysis.*

*PSA Model 2009a Update/Pre-EPU model - CDF = 3.4E-6 / LERF = 1.6 E-7*

*Prior to including the EPU modifications into the PRA model, a couple of deficiencies were noted in the 2009 base model. These items were not related to EPU, therefore were added to the pre-and post-EPU model to be consistent in modeling the plant. These changes included*

*Emergency Feedwater Initiation and Control (EFIC) OTSG pressure control channel fails high mission time of 2 years which is actually 1 month; and the second item was modeling the recovery of Heating, Ventilation & Air Conditioning (HVAC) from a pair of two HRA events to one HRA event.*

| Basic Event | Description | Mission time Old/New | Prob Old/New |
|---|---|---|---|
| QCMPA00H | EFIC OTSG-A PRESSURE CONTROL CHANNEL A FAILS HIGH | 2 year / 1 month | 2.01E-3 / 6.48E-4 |
| QCMPB00H | EFIC OTSG-B PRESSURE CONTROL CHANNEL B FAILS HIGH | 2 year / 1 month | 2.01E-3 / 6.48e-4 |

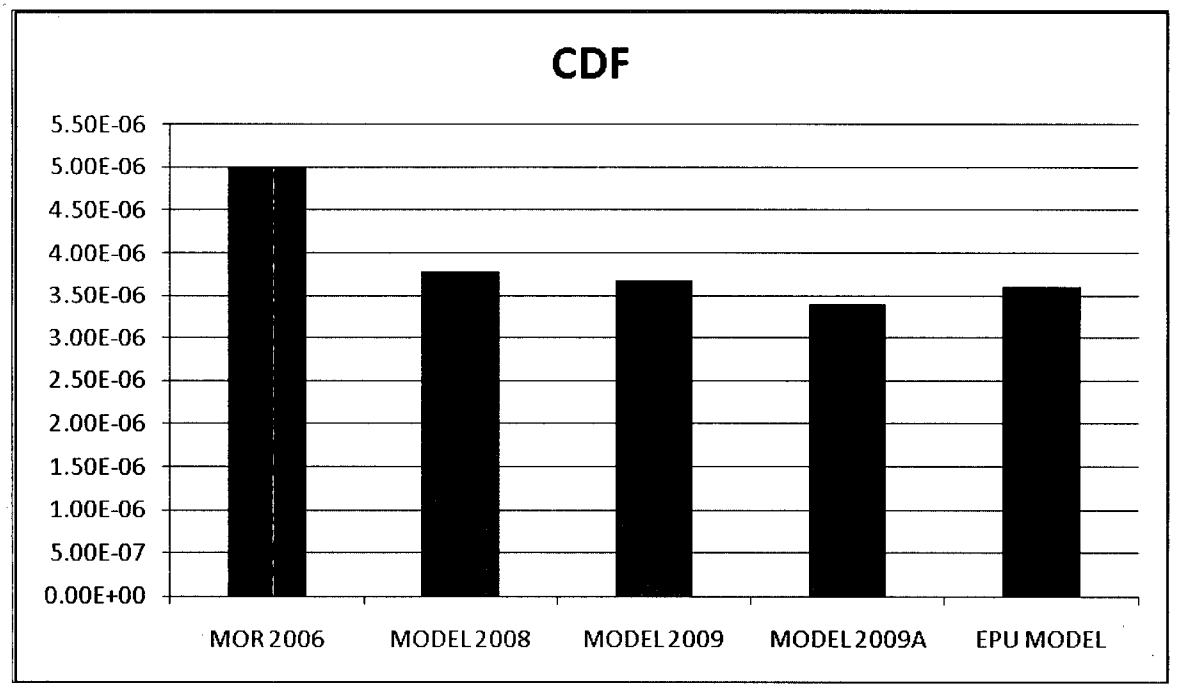
*In the 2009 model, the failure of the HVAC was divided into failure to start the fans and failure of starting the chilled water pumps which caused over conservative dependencies in the cutsets; therefore, one HRA was used as the failure to restore HVAC. JHUCHPARRR_M (OPERATORS FAIL TO START VENTILLATION SYSTEMS) replaced the following HRA events in the fault tree:*

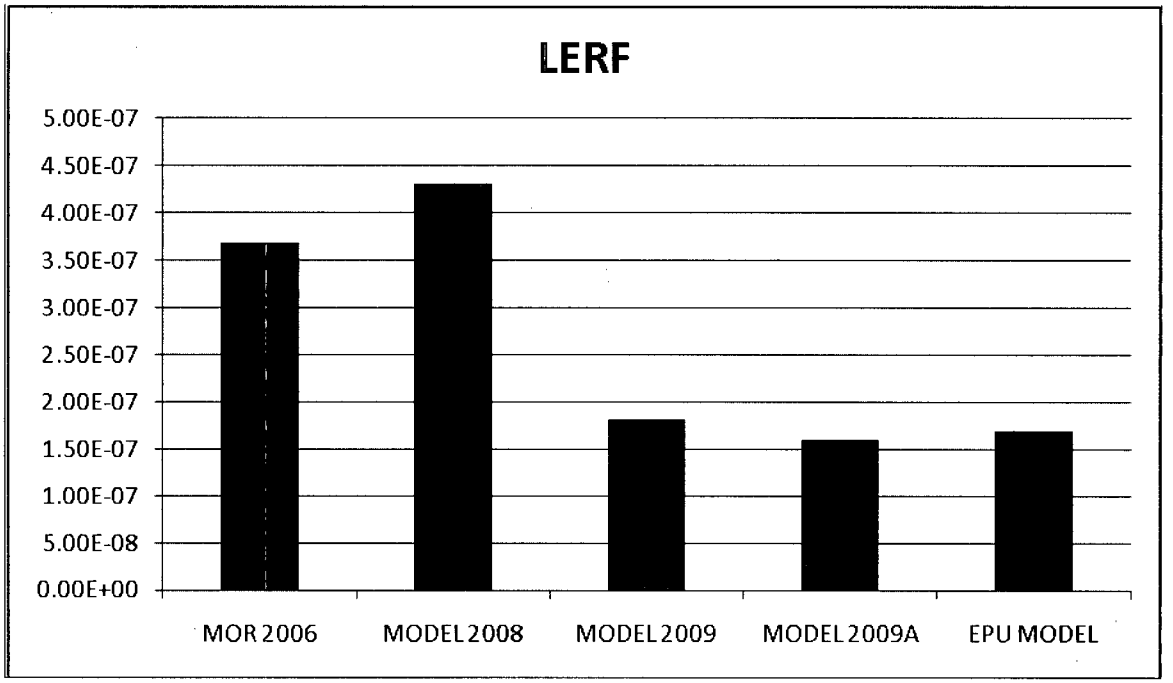
*JHUAHFSY OPERATORS FAIL TO ALIGN AND START STANDBY AH FANS*
*JHUCHPSY OPERATORS FAIL TO ALIGN AND START STANDBY CH TRAIN*
*JHUCHP2Y OPERATORS FAIL TO START CHP-2 APP R CHILLED WATER SYSTEM*

*Post-EPU Model 2009a Update - CDF = 3.6E-6 LERF = 1.7 E-7*

*The changes that are considered for the PRA EPU update include proposed hardware modifications and updated thermal hydraulic analysis that affect operator actions. The notable changes to the PRA model include the following.*

1) *The installation of a low pressure injection cross-tie increased the ISLOCA initiating event which feeds into the LERF contribution.*
2) *Loss of offsite power non-recovery probability was reevaluated to account for new shorter time available.*
3) *The HRA and dependency analysis was affected by the new shorter time available.*
4) *The success criteria for medium LOCA would be changed to include one core flood tank being required as a result of the EPU.*




*Conservative Scaling of Offsite Dose and Consequences Using MOR2006*

*To attempt characterizing the impact that an increase in the fission product inventory, from an EPU, might have on offsite dose and consequences, the person-rem and cost values were scaled by a factor of 1.2 (i.e., 903 MWe scaled to 1083.6 MWe). The base release category frequencies were assumed to be constant. The following table shows the original table from the ER (Table E.3-7) with columns showing the new conservative "post-EPU" values for Dose Risk and Offsite Economic Cost Risk (OECR):*

**Revised Table E.3-7**
**MACCS2 Base Case Mean Results with EPU Estimates**
**Obtained by Scaling Pre-EPU Values by a Factor of 1.2**

| Source Term | Release Category | Pre-EPU Dose (p-rem) | Pre-EPU Offsite Economic Cost ($) | Post-EPU Dose (p-rem) | Post-EPU Offsite Economic Cost ($) | Frequency (/yr) | Pre-EPU Dose-Risk (p-rem/yr) | Pre-EPU OECR ($/yr) | Post-EPU Dose-Risk (p-rem/yr) | Post-EPU OECR ($/yr) |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | IC-1 | 9.81E+03 | 1.02E+04 | 1.18E+04 | 1.22E+04 | 4.15E-06 | 4.07E-02 | 4.23E-02 | 4.89E-02 | 5.08E-02 |
| 2 | RC-1 | 2.06E+05 | 2.17E+07 | 2.47E+05 | 2.60E+07 | 2.48E-08 | 5.12E-03 | 5.39E-01 | 6.14E-03 | 6.47E-01 |
| 3 | RC-1A | 1.51E+05 | 1.56E+06 | 1.81E+05 | 1.87E+06 | 3.94E-10 | 5.94E-05 | 6.14E-04 | 7.13E-05 | 7.37E-04 |
| 4 | RC-1B | 2.17E+06 | 1.85E+09 | 2.60E+06 | 2.22E+09 | 1.55E-08 | 3.36E-02 | 2.87E+01 | 4.04E-02 | 3.44E+01 |
| 5 | RC-1BA | 2.17E+06 | 1.85E+09 | 2.60E+06 | 2.22E+09 | 1.18E-09 | 2.57E-03 | 2.19E+00 | 3.08E-03 | 2.62E+00 |
| 6 | RC-2 | 3.96E+06 | 8.55E+09 | 4.75E+06 | 1.03E+10 | 8.53E-10 | 3.38E-03 | 7.29E+00 | 4.05E-03 | 8.75E+00 |
| 7 | RC-2B | 3.76E+06 | 7.70E+09 | 4.51E+06 | 9.24E+09 | 3.43E-09 | 1.29E-02 | 2.64E+01 | 1.55E-02 | 3.17E+01 |
| 8 | RC-3 | 3.00E+05 | 6.10E+07 | 3.60E+05 | 7.32E+07 | 2.16E-07 | 6.48E-02 | 1.32E+01 | 7.78E-02 | 1.58E+01 |
| 9 | RC-3B | 1.93E+06 | 4.32E+09 | 2.32E+06 | 5.18E+09 | 1.58E-07 | 3.05E-01 | 6.83E+02 | 3.66E-01 | 8.20E+02 |
| 10 | RC-4C | 7.47E+06 | 1.41E+10 | 8.96E+06 | 1.69E+10 | 3.59E-07 | 2.68E+00 | 5.06E+03 | 3.22E+00 | 6.08E+03 |
| 11 | RC-5C | 1.44E+07 | 1.96E+10 | 1.73E+07 | 2.35E+10 | 5.74E-08 | 8.26E-01 | 1.12E+03 | 9.92E-01 | 1.35E+03 |
| **FREQUENCY WEIGHTED TOTALS** | | | | | | **4.99E-06** | **3.98E+00** | **6.95E+03** | **4.77E+00** | **8.34E+03** |

*The next step involved calculating a new value for the Modified Maximum Averted Cost Risk (MMACR), which also accounted for an increase in the replacement power cost by using a factor of 1.2 and an external events multiplier of 12, which was adopted in RAI response 3.c. Based on this new input, the new post-EPU MMACR value was found to be $4,668,000.*

*The last step involved comparing the new post-EPU averted costs to the original implementation costs. The table shown below lists the status of the cost effectiveness for each of the evaluated SAMAs for both pre- and post-EPU conditions at the 95$^{th}$ percentile. In conservatively scaling the offsite dose and consequence values by a factor of 1.2, it was shown that there would be no change in the cost-beneficial status of any of the evaluated SAMAs, which implies that the SAMA results and decision for deciding which SAMAs are cost beneficial remains unchanged as a result of considering the pending EPU implementation at CR-3.*

*It should be noted that the pre-EPU numbers for SAMA 49 were multiplied by a factor of 1.2 directly to obtain the post-EPU values. This is due to the fact that in response to RAI 3.d, the averted cost-risk for SAMA 49 was determined using the ratio of the fire CDF for Battery Charger Room 3A to the internal events CDF, and that ratio multiplied by the original MACR of $341,000. This approach provides a conservative assessment for the cost benefit associated with SAMA 49 for post-EPU conditions.*

**Summary of the Impact of Using the 95$^{th}$ Percentile PRA Results**

| SAMA ID | Cost of Implementation | Pre-EPU Averted Cost Risk (95th Percentile with x12 EE multiplier) | Pre-EPU Net Value (95th Percentile) | Post-EPU Averted Cost Risk (95th Percentile with x12 EE multiplier) | Post-EPU Net Value (95th Percentile) | Change in Cost Effectiveness? (Yes or No) |
|---|---|---|---|---|---|---|
| 34 | $50,000 | $1,238,754 | $1,188,754 | $1,358,227 | $1,308,227 | No |
| 33 | $50,000 | $201,223 | $151,223 | $211,556 | $161,556 | No |

***Summary of the Impact of Using the 95th Percentile PRA Results***

| SAMA ID | Cost of Implementation | Pre-EPU Averted Cost Risk (95th Percentile with x12 EE multiplier) | Pre-EPU Net Value (95th Percentile) | Post-EPU Averted Cost Risk (95th Percentile with x12 EE multiplier) | Post-EPU Net Value (95th Percentile) | Change in Cost Effectiveness? (Yes or No) |
|---|---|---|---|---|---|---|
| 9 | $50,000 | $210,954 | $160,954 | $222,569 | $172,569 | No |
| 10 | $50,000 | $385,886 | $335,886 | $450,632 | $400,632 | No |
| 38 | $50,000 | $156,934 | $106,934 | $171,950 | $121,950 | No |
| 3 | $350,000 | $307,249 | ($42,751) | $321,637 | ($28,363) | No |
| 6 | $400,000 | $258,749 | ($141,251) | $271,541 | ($128,459) | No |
| 5 | $500,000 | $510,643 | $10,643 | $558,333 | $58,333 | No |
| 17 | $500,000 | $357,503 | ($142,497) | $405,846 | ($94,154) | No |
| 11 | $250,000 | $116,334 | ($133,666) | $127,190 | ($122,810) | No |
| 15 | $300,000 | $322,867 | $22,867 | $372,754 | $72,754 | No |
| 4 | $250,000 | $393,159 | $143,159 | $423,033 | $173,033 | No |
| 35 | $700,000 | $3,388,897 | $2,688,897 | $4,023,617 | $3,323,617 | No |
| 51 | $100,000 | $1,004,230 | $904,230 | $1,099,976 | $999,976 | No |
| 49 | $150,000 | $2,237,648 | $2,087,648 | $2,685,178 | $2,535,178 | No |

**RAI 5.a**

Progress Energy does not adequately support why SAMAs were not developed for the 4.16 kV Switchgear Bus Rooms 3A and 3B. On the one hand Progress Energy uses the IPEEE to justify SAMA 49 to upgrade fire barriers in Battery Charger Room 3A, but on the other hand argues that the IPEEE is not an appropriate basis for considering a SAMA to upgrade fire barriers in the switchgear bus rooms. Provide an assessment of a SAMA to reduce fire risk in Switchgear Bus Rooms 3A and 3B.

***Response***

*As discussed in the response to RAI 5.a, it was recognized that changes to 4.16kV Switchgear Bus Rooms 3A and 3B, similar to those proposed for SAMA 49, would be cost beneficial when using the IPEEE results and the external events multiplier of 12. The averted cost-risk and net values for these plant enhancements can be estimated in the same manner as described for SAMA 49 in the response to RAI 3.d.*

*4.16kV Switchgear Bus Room 3A*

*For 4.16kV Switchgear Bus Room 3A, the averted cost-risk can be estimated by multiplying the internal events Maximum Averted Cost Risk (MACR) by the ratio of the 4.16kV Switchgear Bus Room 3A fire CDF to the internal events CDF:*

$$\$341,000 * 7.31E\text{-}06 / 4.95E\text{-}06 = \$503,578$$

*It is assumed that the cost of implementation (COI) for improving the fire barriers in 4.16kV Switchgear Bus Room 3A is the same as what was estimated for Battery Charger Room 3A in*

*SAMA 49 ($150,000). This estimate can be used with the $503,578 averted cost-risk calculated above to quantify the net value:*

***Net Value for Improving the Fire Barriers in 4.16kV Switchgear Bus Room 3A***

| Averted Cost-Risk | COI | Net Value |
|---|---|---|
| $503,578 | $150,000 | $353,578 |

*The net value for this SAMA is the averted cost-risk minus the COI, or $353,578 ($503,578 - $150,000 = $353,578).*

*As documented in Section E.7.2 of the ER, the point estimate based cost-risk values can be multiplied by 2.18 to account for the impact of the use of the 95th percentile PRA results. For 4.16kV Switchgear Bus Room 3A, this results in an averted cost-risk of $1,097,800 (2.18 * 503,578 = $1,097,800) and a net value of $947,800 ($1,097,800 - $150,000 = $947,800).*

*4.16kV Switchgear Bus Room 3B*

*For 4.16kV Switchgear Bus Room 3B, the averted cost-risk can be estimated by multiplying the internal events MACR by the ratio of the 4.16kV Switchgear Bus Room 3B fire CDF to the internal events CDF:*

*$341,000 * 6.79E-06 / 4.95E-06 = $467,756*

*It is assumed that the cost of implementation (COI) for improving the fire barriers in 4.16kV Switchgear Bus Room 3B is the same as what was estimated for Battery Charger Room 3A in SAMA 49 ($150,000). This estimate can be used with the $467,756 averted cost-risk calculated above to quantify the net value:*

***Net Value for Improving the Fire Barriers in 4.16kV Switchgear Bus Room 3B***

| Averted Cost-Risk | COI | Net Value |
|---|---|---|
| $467,756 | $150,000 | $317,756 |

*The net value for this SAMA is the averted cost-risk minus the COI, or $317,756 ($467,756 - $150,000 = $317,756).*

*As documented in Section E.7.2 of the ER, the point estimate based cost-risk values can be multiplied by 2.18 to account for the impact of the use of the 95th percentile PRA results. For 4.16kV Switchgear Bus Room 3B, this results in an averted cost-risk of $1,019,708 (2.18 * 467,756 = $1,019,708) and a net value of $869,708 ($1,019,708 - $150,000 = $869,708).*

**RAI 5.d.i**

Progress Energy did not adequately support why procedures and training could not be identified as an effective means of reducing the risk of basic event APWNR01R. Describe how it is that "no such weaknesses" were identified in the power restoration procedures at CR-3 despite the fact that this event has a failure probability of 0.6?

***Response***

*APWNR01R is not a Human Performance Error, but is a non recovery factor for loss of offsite power. It is a discrete integration of the time-dependent failures and the non-recovery distributions using raw data from Technical Update, Losses of Off-Site Power at U.S. Nuclear Power Plants - 2002, Electric Power Research Institute, TR-1008052, April 2003.*