Progress Energy

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

Ref: 10 CFR 50.90

September 4, 2007
3F0907-03

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

Subject: Crystal River Unit 3 – License Amendment Request #295, Revision 0: Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times Response to Request for Additional Information (TAC No. MD5241)

References:
1. NRC to Crystal River Unit 3 Letter dated July 18, 2007, "Crystal River Unit No. 3, Request for Additional Information Regarding License Amendment Request No. 295 Rev. 0, Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times (TAC No. MD5241)
2. Crystal River Unit 3 to NRC Letter dated April 13, 2007, "Crystal River Unit-3 – License Amendment Request #295, Revision 0, Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times Limiting Time"

Dear Sir:

On July 18, 2007, the Nuclear Regulatory Commission (NRC) issued a Request for Additional Information (RAI, Reference 1) regarding License Amendment Request (LAR) #295 (Reference 2). 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 RAI.

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,

J. A. Franke
Director Site Operations
Crystal River Nuclear Plant

JAF/seb/dar

Attachments: A. Response to Request for Additional Information
B. List of Regulatory Commitments

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

A001
NRR

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Crystal River Nuclear Plant
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STATE OF FLORIDA

COUNTY OF CITRUS

Jon A. Franke states that he is the Director Site Operations, 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
Director Site Operations
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 4th day of Sept., 2007, by Jon A. Franke.

Signature of Notary Public
State of Florida

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

(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 #295, REVISION 0

EXTENSION OF ALLOWED OUTAGE TIME TO SEVEN DAYS AND ELIMINATION OF SECOND COMPLETION TIMES

ATTACHMENT A

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Request for Additional Information Response

On July 18, 2007, the Nuclear Regulatory Commission (NRC) issued a Request for Additional Information (RAI, Reference 1) concerning License Amendment Request (LAR) #295, Revision 0 (Reference 2). Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc., hereby provides the following response to this RAI.

NRC Request

1. **In the submittal with regards to the quality of the CR-3 probabilistic risk assessment (PRA), the licensee does not state the scope of their model (i.e., internal events, external events, fires, seismic, etc., as well as at-power, transition, shutdown). The quantitative results presented are compared to the acceptance guidance of Regulatory Guides (RGs) 1.174 and 1.177, which are based on a full scope risk analysis. The submittal later states that the licensee does not have a fire PRA model. The licensee is requested to clarify the scope of their risk analysis used for this application.**

FPC Response

1. *The Probabilistic Safety Assessment (PSA) model used in this calculation is the Crystal River Unit 3 (CR-3) PSA Model of Record. This model is an at power (Mode 1) internal events model that includes internal flooding events. This model does not include fire events, seismic events, shutdown risk, and transition operation from power to shutdown states. All Nuclear Energy Institute peer certification Facts and Observations have been addressed in this model.*

   A gap assessment for the CR-3 PSA Model of Record and the American Society of Mechanical Engineers PRA Standard was completed in May 2007. Items identified were generally focused on assumptions and uncertainty requirements of the base model.

   The events not included in the model, fire, seismic, shutdown risk and transition operation from power to shutdown states, have been qualitatively or quantitatively assessed as well. These are discussed in the responses to Questions 2, 6 and 7.

NRC Request

2. **In the submittal (Attachment A, page 15 of 27), the licensee states that the only potentially significant external events that could influence the risk are fires and severe weather. This conflicts with the conclusions of the topical report BAW-2295-A Rev. 1, which was the basis for the staff approval of TSTF-430, on which this amendment request is based. BAW-2295-A concluded that the impact of the proposed changes had a negligible impact on external events risk. The basis for the licensee's conclusion identifies a review of the CR-3 Individual Plant Examination (IPE) and supporting data. Since external events, including fires, were not in the scope of the IPE, the basis for this conclusion is not clear. The licensee is requested to provide its disposition of external events (other than fires and severe weather) not included in the risk assessment results (such as seismic events, external flooding, high winds, transportation accidents), and a basis for its conclusions regarding the low-risk significance of each of these events for this application.**

FPC Response

2. *The conclusion in the technical report BAW-2295-A that external events are negligible is still valid. The Individual Plant Examination of External Events (IPEEE) can provide insight into external events. To ensure these results are still valid, the following additional evaluation has been performed.*

Based on the IPEEE, seismic events, transportation accidents, and external floods all have a frequency below 1E-06 per year. At this frequency, with an assumed Conditional Core Damage Probability (CCDP) of 1, and a minimal exposure time, there is negligible impact.

$$(1E\text{-}06\ yr^{-1}) * (7\ days) * (1/365\ yr/days) = \sim 1E\text{-}08$$

Due to the higher frequency of tornados (greater than 1E-06/yr), the high wind evaluation warrants additional analysis using the IPEEE information and the current PRA model. Table 2-1 provides a summary of high wind events quantified in the IPEEE updated with current model runs that considered the Allowed Outage Time (AOT):

Table 2-1 – Tornado and Unit 1 Stack Fall Impact

Event	IPEEE Frequency [1] (1/yr)	CDF with No AOT in Effect [2]	CDF Train A AOT in Effect [2,4]	CDF Train B AOT in Effect [2,5]	Total 7 Day ICCDP for Limiting Train[8]	4 Day Increase ICCDP for Limiting Train[6]
Tornado frequency for F1 case and greater	2.1E-05	3.4E-09	6.7E-09	5.3E-09	6.3E-11	3.6E-11
Loss of Category 1 structures due to tornado[3]	6.3E-08	6.3E-08	6.3E-08	6.3E-08	0	0
Unit 1 stack falling on EFW tank enclosure due to high winds	6.5E-06	2.6E-08	1.1E-07	1.0E-07	1.6E-09	9.2E-10

Table Notes:

1. The initiating frequencies for tornado events are taken from section 5.1.1 of the IPEEE. The initiating event frequency for Unit 1 stack falling on Emergency Feedwater (EFW) tank enclosure is from section 5.1.2 of the IPEEE.
2. Base Core Damage Frequency (CDF) is calculated by taking the CCDP of the Loss of Offsite Power initiating event without credit given for recovery of offsite power.
3. The CCDP for loss of Category 1 structures due to a tornado is assumed to be 1. Due to the low initiating event probability, no further analysis has been performed.
4. Train A Low Pressure Injection (LPI) CDF runs have the Train A Decay Heat Removal (DH), Building Spray (BS), Decay Heat Closed Cycle Cooling (DC), and associated Raw Water (RW-DC) pumps failed.
5. Train B LPI CDF runs have the Train B Decay Heat Removal, Building Spray, Decay Heat Closed Cycle Cooling, and associated raw water pumps failed.
6. The Incremental Conditional Core Damage Probability (ICCDP) increase is calculated using the following equation: ICCDP = (4 days) * (1 yr / 365 days) * ("Limiting Train CDF" – "Base CDF")
7. Values in this table were generated using a truncation of 1E-12 for quantified results. Some cutsets may not have been recovered due to the large number of cutsets generated. If recoveries are not applied, the results will be conservative.
8. The ICCDP increase is calculated using the following equation:
   ICCDP = (7 days) * (1 yr / 365 days) * ("Limiting Train CDF" – "Base CDF")

With the exception of fire and severe weather, external events contribute less than 1E-08 ICCDP for an increase of 4 days per AOT entry for each train of DH, DC, BS, and associated RW-DC trains. Based on these results, it is reasonable to state that these external events have a negligible impact on risk for this application.

NRC Request

3. In the submittal, the licensee identified general assumptions relevant to their risk analysis, including that "During performance of the AOT, the corresponding opposite train equipment and diesel are considered to be protected." The term "protected" should be more clearly defined as to what specifically is prohibited while the equipment is protected. The specific scope of equipment should be stated, and appropriate regulatory commitments identified.

FPC Response

3. When a corresponding opposite train is deemed "protected," administrative controls are implemented to minimize the possibility of the loss of use of this equipment as described in Administrative Instruction 0500, "Conduct of Operations Department Organization and Administration." There is no planned maintenance or discretionary equipment manipulation on protected equipment or its power supplies. The scope of affected equipment is equipment within the protected train whose loss of function would result in the inability to ensure safe shutdown. This includes pumps, valves, required instrumentation and power supplies of the systems required for decay heat removal capability, inventory control, power availability, reactivity control, pressure control and containment.

To ensure safe shutdown, the following commitment is made for extended (greater than 72 hours) preplanned outages on the LPI, BS, DC or RW-DC System (see Attachment B):

> *"The Remote Shutdown Panel, the Appendix R Cooler and the opposite train of LPI, BS, DC, RW-DC, EFW, Auxiliary Feedwater System, Emergency Feedwater Initiation and Control System, HPI, and their power supplies (AC and DC) will be administratively designated as "protected" (i.e., no planned maintenance or discretionary equipment manipulation)."*

NRC Request

4. In the submittal (Attachment E, Section 5.3), the licensee states that 96 additional hours of unavailability are assumed for each train of the affected systems, and that the unavailability would be incurred simultaneously for each train of the systems. No basis has been provided to justify either of these assumptions based on maintenance practices and past component maintenance history. This assumption is also inconsistent with the topical report BAW-2295-A Rev. 1. BAW-2295-A identified an increase from 80 to 93 hours of unavailability per train per year. Further, the licensee identified a compensatory measure (Attachment A, page 17) to schedule concurrent outages of raw water and decay heat systems, "if possible." This appears to conflict with an assumption of concurrent outages, since it identifies that such concurrent scheduling may not be possible in some circumstances. The licensee should justify these assumptions and clarify the intent of the compensatory

measures regarding concurrent outages, or provide more realistic risk analyses accounting for non concurrent outage scheduling.

FPC Response

4. *The increased average yearly unavailability for extending this AOT is expected to be less than or equal to the 13 hours presented in BAW-2295-A. The PRA calculations used 96 hours (the difference from 3 to 7 days) to bound results. The risk analysis has been re-performed with the 13 hour increased unavailability assumption. Results are presented in Table 4-2.*

   The Reference 2 submittal assumed DH, BS, and RW-DC would all be removed from service at the same time for maintenance associated with the extended AOT. This concurrent maintenance assumption has been removed in the results presented in Table 4-2. Table 4-1 provides modified test and maintenance terms calculated for the full AOT (7 days) and the expected average yearly value (current plus 13 hours). The expected average yearly value is based on the 13 hour increase provided in BAW-2295-A.

Table 4-1 – Unavailability Values for Non-Concurrent Maintenance Evaluation

Description	Nominal Model Value	Full AOT Values	Expected Values
DHP-1A Train in Maintenance	1.03E-02	2.13E-02 [1]	1.18E-02 [3]
RB Spray Train A in Maintenance	9.4E-03	2.04E-02 [1]	1.09E-02 [3]
RWP-3A in Maintenance	8.6E-03	1.96E-02 [1]	1.01E-02 [3]
DC Train A in Maintenance	4E-03	1.92E-02 [2]	5.48E-03 [3]
DHP-1B Train in Maintenance	1.03E-02	2.13E-02 [1]	1.18E-02 [3]
RB Spray Train B in Maintenance	9.4E-03	2.04E-02 [1]	1.09E-02 [3]
RWP-3B in Maintenance	8.6E-03	1.96E-02 [1]	1.01E-02 [3]
DC Train B in Maintenance	4E-03	1.92E-02 [2]	5.48E-03 [3]

Table Notes:
1. These Full AOT values are calculated by adding 4 days of unavailability per year to the current PRA model unavailability value. This represents the 4 day increase from 3 days to 7 days.
2. These Full AOT values are calculated by using 7 days of unavailability per year. The current value is less than 3 days per year, so the addition of 4 days would not have encompassed the full 7 day period.
3. The expected values for mean duration are calculated by adding 13 hours of unavailability per year to the current PRA model unavailability value.

Table 4-2 compares BAW-2295-A results to the current CR-3 PRA model results. Current results are provided with and without the concurrent maintenance assumption. A mean duration increase of 25 hours is used. BAW-2295-A evaluation used 13 hours. Results shown in Table 4-2 show the removal of the concurrent maintenance assumption does not significantly impact the results. The PRA calculations result in lower risk values than those presented in BAW-2295-A. Concurrent maintenance will still be performed whenever practical.

Table 4-2 – Comparison of BAW-2295-A to Current Model Results

Risk Measure	BAW-2295-A Values for CR3	Current Model Results, without Concurrent Maintenance Assumption
Base CDF (/yr)	8.18E-06	4.70E-06 [4]
4 Day Increase ICCDP (Single AOT Risk)	4.6E-07	3.4E-07 [1,4]
Total 7 Day ICCDP (Single AOT Risk)	8.0E-07	6.0E-07 [4,5]
Incremental Yearly CDF (Mean expected duration) (/yr)	4.4E-07	1.9E-07 [2,4]
Incremental Yearly CDF (full AOT) (/yr)	2.3E-06	1.67E-06 [2,4]
Base LERF (/yr)	1.84E-07	3.69E-07
Incremental CLERP (Single AOT Risk)	1.5E-08	4.03E-10 [6]
Incremental Yearly LERF (Mean duration) (/yr)	9E-09	[3]
Incremental Yearly LERF (full AOT) (/yr)	4.5E-08	[3]

Table Notes:

1. Based on Train B Out Of Service (OOS) which gives results that are more conservative. These results use a 4 day period which is equal to the change from 3 to 7 days. All maintenance events were at nominal values. If compensatory measures limit maintenance on Emergency Feedwater System (EFW), Auxiliary Feedwater System (AFW), Emergency Feedwater Initiation and Control System (EFIC), High Pressure Injection (HPI), Appendix R Cooler, and their power supplies, the resulting value would be 2.5E-07.
2. Quantified using values from Table 4-1.
3. The LERF values have not been re-quantified based on the Conditional Large Early Release Possibilities (CLERP) value. These values are the Incremental LERF values and will not provide additional insight.
4. Model truncation of 1E-11 has been used to quantify level 1 results.
5. Based on Train B OOS which gives results that are more conservative. These results use a 7 day period. All maintenance events were at nominal values. If compensatory measures limit maintenance on EFW, AFW, EFIC, HPI, Appendix R Cooler, and their power supplies, the resulting value would be 4.4E-07.
6. Based on Train B OOS which gives results that are more conservative. These results use a 7 day period. All maintenance events were at nominal values.

NRC Request

5. The submittal did not analyze emergent component failures (i.e., unplanned maintenance) and the potential impact on risk from increased probability of common cause failure. Since the applicability of the extended AOT is not limited to planned maintenance activities, the risk analysis should address emergent repairs per RG 1.177 Section 2.3.3.1 and Appendix A Section A.1.3.2.

FPC Response

5. *Random failure rates were left at their nominal value for risk metrics evaluations. Unplanned testing and maintenance contributions were included in the delta CDF and delta Large Early Release Frequency (LERF) calculation. The expected increase in unavailability was added to the existing term to ensure higher levels of planned maintenance and unplanned portions were not lost.*

The CCDP values were originally calculated with test/maintenance terms set to zero for the opposite trains of LPI, BS, DC, and RW-DC. Additionally, the test/maintenance terms were set to zero for the Appendix R Cooler, for all trains of EFW, AFW, EFIC, and HPI, and for their power supplies. To evaluate the possible impact of these test and maintenance events, the ICCDP change was re-quantified with average maintenance included for all components.

Table 5-1 – Impact of Test and Maintenance Cases

Risk Measure	Nominal Testing Maintenance	Reduced Test and Maintenance
Total 7 Day ICCDP (Single AOT Risk)	6.0E-07[3]	4.4E-07[3]
4 Day Increase ICCDP (Single AOT Risk)	3.4E-07[1]	2.5E-07[1]

Table Notes:
1. Based on Train B OOS which gives results that are more conservative. This result uses a 4 day period which is equal to the change from 3 to 7 days and is described in Table 4-2.
2. Model truncation of 1E-11 has been used to quantify level 1 results.
3. Based on Train B OOS which gives results that are more conservative. This result uses a 7 day period and is described in Table 4-2.

Common Cause Impact:

Unplanned failures and maintenance can result in AOT entries when the status of the complementary equipment of the same type and function may not be known. In these cases, the common cause term is larger than the nominal value. To evaluate common cause contribution, ICCDPs have been calculated for the DH, DC, and RW-DC trains. BS was not evaluated for this sensitivity because BS does not affect the level 1 results and does not impact the level 2 results significantly. Therefore, the common cause contribution for BS is considered to be negligible when compared to the level 1 results of DH, DC, and RW-DC.

Each train is evaluated for common cause impact by setting the pump for the desired train as failed, and increasing the common cause term for the pump to its largest respective common cause beta value. This results in conservative values, but provides some insight into the maximum possible impact.

A sensitivity of equipment recoveries was not performed because equipment recoveries are not credited in the base PRA model.

Table 5-2 – Common Cause[2]

Failed Component	Description	Model Value	Beta Value	Base CDF	New CDF	Total 7 Day ICCDP[3]	4 Day Increase ICCDP Change[1]
DHP-1A	DHP-1A,1B CCF[4]	1.02E-04	5.01E-02	4.70E-06	5.94E-05	1.05E-06	5.99E-07
DHP-1B	DHP-1A and DHP-1B fail to start	1.02E-04	5.01E-02	4.70E-06	6.16E-05	1.09E-06	6.24E-07
DCP-1A	DCP-1A,1B CCF	3.52E-05	1.52E-02	4.70E-06	3.39E-05	5.60E-07	3.20E-07
DCP-1B	DCP-1A,1B CCF	3.52E-05	1.52E-02	4.70E-06	4.72E-05	8.15E-07	4.66E-07
RWP-3A	RWP-3A,3B CCF	2.97E-05	3.55E-02	4.70E-06	4.90E-05	8.50E-07	4.85E-07
RWP-3B	RWP-3A,3B CCF	2.97E-05	3.55E-02	4.70E-06	6.21E-05	1.10E-06	6.29E-07

Table Notes:
1. This result uses a 4 day period which is equal to the change from 3 to 7 days.
2. Model truncation of 1E-11 has been used to quantify level 1 results.
3. This result uses a 7 day period and base model alignments.
4. Common Cause Failures.

The large CCF ICCDPs calculated are about twice as large as the point estimate value with nominal common cause values used. The impact of the common cause term is expected to be minimal for several reasons:

- *The unplanned maintenance unavailability is a small fraction of the overall pump unavailability.*
- *Only a small percentage of unplanned maintenance events last longer than three days, the current Technical Specification window. If the failed component is fixed within three days, there is no increase in risk to the public.*
- *An equipment failure or deficiency requires an assessment to determine if this failure has a common impact on redundant or similar equipment. Only a small percentage of common cause events are expected to be unidentified within the first day of the outage window which is less than the original Technical Specification window. If it is identified that an existing failure has a common impact on redundant or similar components, these components would be declared inoperable. This would require entry into a different Technical Specification.*

NRC Request

6. The submittal discusses the development of a fire risk sensitivity study and presents its results. The staff requests additional clarifications regarding the fire risk of the proposed changes:

FPC Response

6. To respond to this RAI, a new fire assessment was needed. A full fire PRA is not available for evaluation of the AOT. However, sufficient information exists in the IPEEE to support this submittal. The IPEEE cable routing database was used in conjunction with the current PRA model. The initial screen numbers will be used where all unprotected cables are assumed to fail if fire suppression is unsuccessful. Since detailed fire modeling is not credited, this will provide an alternate, conservative fire approach to assess the risk of fire as it relates to the 7 day AOT for LPI and BS.

Assumptions:

- *The cable spreading and control room are excluded from the analysis because both trains are impacted by fire, so the delta risk is expected to be zero with the simplified approach being used.*
- *Testing and Maintenance events were left at their nominal values.*
- *Fire wrap credit is given for the historical IPEEE evaluated condition.*
- *Fire suppression credit is given for the historical IPEEE evaluated condition.*
- *Fire ignition frequencies from the historical IPEEE are used.*
- *All equipment that is not fire wrapped is assumed to fail for all fires when suppression fails.*

The fire compartments were generally run at a truncation of approximately 1E-08. Some compartments would not quantify at 1E-08 because they contained a large amount of equipment that is assumed to fail. For these compartments, the truncation was raised to yield results.

Results were generated by adding up the CDF for all the fire compartments. Three CDFs were generated: a base CDF with nominal test and maintenance fraction, and two cases with Train A and B AOT removed from service. An ICCDP was calculated using a delta CDF and the 4 day increase. Train B yields a larger result with a change of ICCDP of 3.7E-07. The total 7 day ICCDP is 6.5E-07.

The following fire areas showed they were sensitive to fire risk in conjunction with a DH, DC, BS, and RW-DC outage:

- *The opposite train DH, DC, and RW-DC rooms.*
- *The opposite train MCC rooms (480 and 4160).*
- *Non-Safety 4160V and 480V rooms.*
- *The opposite train battery and charger rooms.*
- *The opposite train Inverter rooms.*
- *Remote Shutdown Room.*

Spurious actuations were not evaluated in the quantified fire assessment. However, they can be addressed qualitatively. Based upon a review of the PRA cutset results and the model, the opening of the pressurizer PORV would be the primary spurious event of concern when the DH system is unavailable. Although, the actual risk of this spurious event due to fire is not quantified, the impact can be mitigated by performing actions to protect cables associated with the PORV and the PORV block valve.

The List of Regulatory Commitments has been revised to reflect the above information. The new list is provided in Attachment B.

a. The submittal does not discuss the role of the LPI subsystem and its cooling water support systems for mitigation of fire events for achieving a safe shutdown. The licensee should discuss the anticipated role of these systems given a fire in the plant.

LPI is a function of the Decay Heat Removal System. The Appendix R Fire Study states the safe shutdown requirement for the Decay Heat Removal System is as follows:

The functions of the Decay Heat Removal System are:

1) *Provide decay heat removal and temperature monitoring for cold shutdown; and*
2) *Assure RC outlet isolation until reaching hot shutdown conditions; and*
3) *Isolate the Borated Water Storage Tank from the Reactor Building sump.*

The PRA has an additional requirement for Decay Heat Removal to provide a suction source and cool suction water from the emergency sump to makeup pumps for long term Feed and Bleed cooling.

b. **The fire zones of interest are identified as those which contain circuits of the systems for which the AOT extension is requested. However, fires which disable other systems resulting in a demand for the LPI and supporting systems would also be impacted by the AOT extension. For example, fire-induced failure of emergency feedwater components would increase the likelihood of requiring core cooling by bleed-and-feed once through cooling, which may ultimately require the use of LPI systems for long term decay heat removal. The licensee should justify the selection of fire scenarios.**

   The IPEEE fire analysis has been re-evaluated to address all originally scoped compartments. This ensures possible interactions with support and redundant systems have been captured.

c. **The submittal states that fires which impact "manual actions that are credited in the Fire Study" are expected to have a minimal risk impact. It is not understood to what the term "Fire Study" refers, nor are the manual actions identified in any way, therefore the basis for this statement is not clear. The licensee should clarify the intent of this assumption and assure it has a proper basis.**

   The "Fire Study" referred to is the CR-3 Appendix R Safe Shutdown Analysis. The manual actions were those actions in the CR-3 Appendix R Safe Shutdown Analysis. The fire analysis has been re-evaluated using IPEEE information instead of the "Fire Study" presented in the previous submittal.

d. **The development of the "compensated frequency" adjusts for "equipment which will not be operated without special precautions." It is not understood to what "special precautions" refers, to what equipment these precautions apply, and whether these are associated with any licensee commitments specific for this application. The licensee should clarify the scope and intent of this assumption.**

   Compensated frequency was an adjustment to the initiating event frequency for each fire compartment to reflect what equipment would be operating. If a motor was not going to be energized during the AOT, its contribution was removed from the analysis. This assumption is not used in the re-evaluated fire analysis.

   "Special precautions" was used to refer to fire watches. This is no longer credited in the fire analysis.

e. **The licensee applied a 0.1 conditional core damage probability given a fire and identified this as a conservative assumption. However, no specific basis has been provided to justify this value as conservative. Further, the final results of the fire sensitivity study using this assumption do not demonstrate low sensitivity to fire risk, which would indicate that the use of a conservative screening value may be inappropriate. The licensee should justify the use of 0.1 as the conditional core**

damage probability, or consider applying a more realistic value to better establish a less conservative estimate of fire risk during the extended AOT.

The IPEEE fire analysis has been re-evaluated with model quantification to generate estimates based on the current PRA model. Although detailed fire modeling has not been performed, this analysis (discussed in the response to 6, above) still provides conservative results. Initial screening values were used with all unprotected equipment in rooms assumed to fail because they contained a fire.

f. The results of the sensitivity study show that fire risk dominates the risk impact of the extended AOT, and the acceptance guidance of RG 1.177 of 5E-7 incremental conditional core damage probability is exceeded by almost a factor of 10. The licensee has not drawn any conclusions from these results, nor have specific compensatory measures been identified to offset this risk. (The licensee has proposed an undefined "periodic" fire watch in the decay heat pump room and seawater rooms, which contain the equipment subject to the extended AOT. However, in their fire analysis the decay heat pump rooms have a compensated frequency of "0" since the pumps are not running, so this compensatory fire watch does not directly offset any specific source of the fire risk.) The licensee should discuss the implications of the fire risk sensitivity study to the overall conclusions of the risk impacts associated with the extended AOTs, and consider implementing specific compensatory measures to address fire risk as a significant contributor to the risk profile.

As indicated in the response to 6.d. above, compensated frequency is no longer used in the fire analysis. As such, these fire watches are now compensatory measures that will directly offset some of the risk and not an assumption of the analysis. The compensatory measures, identified in the List of Regulatory Commitments, have been revised based on the re-evaluated fire analysis discussed in the response to Question 6, above. The revised List of Regulatory Commitments is included as Attachment B to this RAI response.

NRC Request

7. The submittal did not identify whether the risk analyses provided point estimates of the mean or actual means, nor was there any discussion of uncertainty analyses to support the calculations. The licensee is requested to address PRA model and parametric uncertainty using the guidance of RG 1.174 Section 2.2.5 and RG 1.177 Section 2.3.5.

FPC Response

7. The results provided in the submittal and in this RAI response are point estimates. Additional uncertainty analysis has been performed.

UNCERT Parameter Analysis

Parameter uncertainty was evaluated using the Risk and Reliability Workstation's UNCERT software. The results are provided in Table 7-1.

Table 7-1 – Parameter Uncertainty Results

Risk Measure	Base Model	Train B AOT	Total 7 Day ICCDP (Single AOT Risk)[3]	4 Day Increase ICCDP (Single AOT Risk)[1]
CDF (/yr)	4.70E-06	3.61E-05	6.0E-07	3.4E-07
Mean	4.67E-06	3.61E-05	6.0E-07	3.4E-07
5th Percentile	1.62E-06	1.22E-05	2.0E-07	1.2E-07
50th Percentile	3.65E-06	2.75E-05	4.6E-07	2.6E-07
95th Percentile	1.05E-05	8.10E-05	1.4E-06	7.7E-07

Table Notes:
1. Based on Train B OOS which gives results that are more conservative. This result uses a 4 day period which is equal to the change from 3 to 7 days. All maintenance events were at nominal values.
2. Model truncation of 1E-11 has been used to quantify level 1 results.
3. Based on Train B OOS which gives results that are more conservative. This result uses a 7 day period. All maintenance events were at nominal values and base model alignments.

Small LOCA Parameter Uncertainty

When reviewing the CCDP cutsets, small Loss of Coolant Accident (LOCA) was identified as a dominant initiator for cutsets including AOT failures. To evaluate the sensitivity of the initiating event frequency, the change in ICCDP has been recalculated at the 95th percentile.

Table 7-2 – Comparison of Small LOCA Uncertainty

Risk Measure	Current Model Results with Nominal Small LOCA	Current Model Results with 5th Percentile Small LOCA	Current Model Results with 95th Percentile Small LOCA
Small LOCA Initiator Value	5.0E-04	1E-04	1E-03
Base CDF (/yr)	4.70E-06	3.48E-06	6.22E-06
LPI Train B CDF (/yr)	3.61E-05	2.58E-05	4.89E-05
Total 7 Day ICCDP (Single AOT Risk)	6.0E-07[3]	4.3E-07[3]	8.2E-07[3]
4 Day Increase ICCDP (Single AOT Risk)	3.4E-07[1]	2.4E-07[1]	4.7E-07[1]

Table Notes:
1. Based on Train B OOS which gives results that are more conservative. This result uses a 4 day period which is equal to the change from 3 to 7 days. All maintenance events were at nominal values.
2. Model truncation of 1E-11 has been used to quantify level 1 results.
3. Based on Train B OOS which gives results that are more conservative. This result uses a 7 day period. All maintenance events were at nominal values and base model alignments.

Component Parameter Uncertainty

Component failure rates are not always at their nominal failure rate. When one train of BS, DH, DC and RW-DC is removed from service, the opposite train becomes important for mitigating plant events that require DH. The impact of varying failure rates for these individual pumps has been evaluated in Table 7-3. With the individual pumps at the 95th percentile value, there is no significant increase in risk. Train B AOT was used to evaluate this case because it yields larger numbers.

Table 7-3 – Comparison of Individual Component Uncertainty

Component	Risk Measure	Results with Nominal Value	Results with 5th Percentile Value	Results with 95th Percentile Value
DHP-1A	Basic Event Value	1.98E-03	6.43E-04	4.34E-03
	Base CDF (/yr)	4.70E-06	4.65E-06	4.78E-06
	LPI Train B CDF (/yr)	3.61E-05	3.51E-05	3.78E-05
	4 Day Increase ICCDP (Single AOT Risk)	3.4E-07 [1]	3.3E-07 [1]	3.6E-07 [1]
	Total 7 Day ICCDP (Single AOT Risk)	6.0E-07 [3]	5.8E-07 [3]	6.3E-07 [3]
DCP-1A	Basic Event Value	2.05E-03	9.76E-04	3.66E-03
	Base CDF (/yr)	4.70E-06	4.66E-06	4.75E-06
	LPI Train B CDF (/yr)	3.61E-05	3.53E-05	3.73E-05
	Incremental change CCDP (Single AOT Risk)	3.4E-07 [1]	3.4E-07 [1]	3.6E-07 [1]
	Total 7 Day ICCDP (Single AOT Risk)	6.0E-07 [3]	5.9E-07 [3]	6.2E-07 [3]
RWP-3A	Basic Event Value	9.26E-04	3.93E-04	1.77E-03
	Base CDF (/yr)	4.70E-06	4.68E-06	4.72E-06
	LPI Train B CDF (/yr)	3.61E-05	3.57E-05	3.67E-05
	Incremental change CCDP (Single AOT Risk)	3.4E-07 [1]	3.4E-07 [1]	3.5E-07 [1]
	Total 7 Day ICCDP (Single AOT Risk)	6.0E-07 [3]	5.9E-07 [3]	6.1E-07 [3]

Table Notes:
1. Based on Train B OOS which gives results that are more conservative. This result uses a 4 day period which is equal to the change from 3 to 7 days. All maintenance events were at nominal values and base model alignments.
2. Model truncation of 1E-11 has been used to quantify level 1 results.
3. Based on Train B OOS which gives results that are more conservative. This result uses a 7 day period. All maintenance events were at nominal values and base model alignments.

Loss of Offsite Power Parameter Uncertainty

A Loss of Offsite Power sensitivity was performed which showed results to be insensitive to changes in the initiating event value.

Assumption of Expected Unavailability

The expected increase in unavailability due to the increased AOT was 13 hours. This assumption of expected unavailability has a direct impact on delta CDF and delta LERF results. The level 1 model is more sensitive than the level 2 model. In the level 1 model, the delta CDF has been calculated using an average yearly increase of 26 hours. The 26 hour increase is based on historical data for maintenance and surveillances, including rebuilding pumps online.

Table 7-4 provides modified test and maintenance terms calculated for the 13 hour and 26 hour expected average yearly assumption. Table 7-5 provides the results which show values that are still lower than those identified in the topical report. The delta CDF increased linearly to the increased hours.

Table 7-4 – Unavailability Values for Expected Increase Unavailability Sensitivity

Description	Nominal Model Value	Mean Duration Increase of 13 Hours	Mean Duration Increase of 26 Hours
DHP-1A Train In Maintenance	1.03E-02	1.18E-02 [1]	1.33E-02 [2]
RB Spray Train A in Maintenance	9.4E-03	1.09E-02 [1]	1.24E-02 [2]
RWP-3A in Maintenance	8.6E-03	1.01E-02 [1]	1.16E-02 [2]
DC Train A in Maintenance	4E-03	5.48E-03 [1]	6.97E-03 [2]
DHP-1B Train in Maintenance	1.03E-02	1.18E-02 [1]	1.33E-02 [2]
RB Spray Train B in Maintenance	9.4E-03	1.09E-02 [1]	1.24E-02 [2]
RWP-3B in Maintenance	8.6E-03	1.01E-02 [1]	1.16E-02 [2]
DC Train B in Maintenance	4E-03	5.48E-03 [1]	6.97E-03 [2]

Table Notes:
1. The mean duration values are calculated by adding 13 hours of unavailability per year to the current PRA model unavailability value.
2. The mean duration values are calculated by adding 26 hours of unavailability per year to the current PRA model unavailability value.

Table 7-5 – Comparison of BAW-2295-A to Current Model Results

Risk Measure	BAW-2295-A Values for CR3 with Mean Duration Increase of 13 Hours	Mean Duration Increase of 13 Hours	Mean Duration Increase of 26 Hours
Base CDF (/yr)	8.18E-06	4.70E-06	4.70E-06
Modified Unavailability CDF (/yr)	Not provided	4.89E-06	5.09E-06 [1]
Incremental Yearly CDF (Mean duration) (/yr)	4.4E-07	1.9E-07	3.9E-07

Table Notes:
1. Quantified using values from Table 7-4.
2. Model truncation of 1E-11 has been used to quantify level 1 results.

Assumption of EFIC Failure Without Room Cooling

The EFW system provides long term decay heat removal and therefore, plays an important role as a redundant system to the DH system. On a loss of EFIC room cooling the failure mode of the EFW valves is uncertain. Although these valves fail open on a loss of power, there is a possibility when the control cabinets over heat with power still available the EFW valves may fail to the closed position. The PRA model assumes a 50% probability of failure for EFW valves on a loss of cooling to the EFIC control panels. The impact of varying the assumed state of EFW valves on a loss of EFIC room cooling has been evaluated in Table 7-6. With the individual valves assumed to fail on a loss of EFIC room cooling, there is not a significant increase in risk. Train B AOT was used to evaluate this case because it yields larger numbers.

Table 7-6 – EFIC Failures on Loss of Room Cooling

Risk Measure	Results with Nominal Values	EFW Valves Failing Open on Loss of EFIC Room Cooling	EFW Valves Failing Closed on Loss of EFIC Room Cooling
EFIC failure on a Loss Of Room Cooling	50%	0%	100%
Base CDF (/yr)	4.70E-06	3.57E-06	5.81E-06
LPI Train B CDF (/yr)	3.61E-05	2.33E-05	4.75E-05
Total 7 Day ICCDP (Single AOT Risk)	6.0E-07[3]	3.78E-07[3]	8.0E-07[3]
4 Day Increase ICCDP (Single AOT Risk)	3.4E-07[1]	2.16E-07[1]	4.6E-07[1]

Table Notes:
1. Based on Train B OOS which gives results that are more conservative. This result uses a 4 day period which is equal to the change from 3 to 7 days. All maintenance events were at nominal values and base model alignments.
2. Model truncation of 1E-11 has been used to quantify level 1 results.
3. Based on Train B OOS which gives results that are more conservative. This result uses a 7 day period. All maintenance events were at nominal values and base model alignments.

Transition Risk Completeness Uncertainty
A transition model that can be used to evaluate the impact of removing DH from service has not been developed.

Although maintenance is not normally planned during transition modes, it is possible for unplanned unavailability to exceed the AOT limit prompting a shutdown with one train of DH unavailable. Increasing the AOT from 3 to 7 days would decrease the frequency of this occurrence and presents a decrease in risk to the public.

Shutdown Risk Completeness Uncertainty
A shutdown model that can be used to evaluate the impact of removing DH from service has not been developed. This risk can only be considered on a qualitative level. This would decrease the risk impact if it were performed.

Fire Risk Completeness Uncertainty
The fire contribution for external events is based on routing information from the IPEEE. Using the IPEEE information to perform fire analysis provides a level of insight but does not provide a thorough fire assessment.

NRC Request

8. The submittal did not address the truncation level used for this analysis. The licensee is requested to provide this information consistent with RG 1.177 Section 2.3.3.4.

FPC Response

8. The original submittal results were quantified with a 1E-12 truncation. DH, DC, and RW-DC events are in the top 50 cutsets. However at this truncation level, the number of cutsets is found to increase above the capability of the recovery program when LPI is removed from service. This means some cutsets for these cases do not recover. The CCDP values generated for these cases are higher than they would be otherwise. This

causes delta CDF and CCDP to be conservative. Since this recovery failure has a conservative bias, re-quantification of results at a lower truncation is unnecessary, provided results could be obtained. However, results for some cases can not be quantified at a truncation level of 1E-12. These cases are run at an increased truncation level, the lowest the code would support.

The truncation levels are presented in the footnotes for tables containing results.

NRC Request

9. The submittal (Attachment A, page 16) ends the first bulleted compensatory measure with a phrase "These protective measures ..," which is not completed. The licensee is requested to provide the missing information.

FPC Response

9. This is the last sentence of the first of five bullets in that paragraph. The complete sentence should be:

> *"These protective measures will continue after the proposed AOTs have been implemented."*

This is immediately followed by the second of five bullets in that paragraph.

NRC Request

10. The submittal (Attachment A, page 16) identifies a commitment to "avoid an AOT... that results in 'Higher Risk' (Orange Color Code)." The licensee is requested to clarify the intended meaning of "avoiding an AOT," and to specify the basis for the "Higher Risk (Orange Color Code)" in terms of risk impact.

FPC Response

10. To avoid an AOT that results in "Higher Risk" (Orange Color Code) during an extended outage (greater than 72 hours) of the LPI, BS, DC or RW-DC System means CR-3 will not plan any maintenance that results in "Higher Risk" (Orange Color Code) when operating under those conditions. This is why it was identified as the first of the Regulatory Commitments identified in Attachment F of the submittal. The Orange Color Code is an administrative guideline that indicates risk significant SSCs have been removed from service, and that more stringent risk management actions need to be taken. It is defined when the ICCDP is greater than or equal to 1.0E-05 for a 7 day period or greater than or equal to 1.0E-06 for a 36 hour period. Under these conditions, the Work Week Manager and Superintendent Shift Operations determine what necessary actions should be taken to protect redundant trains and to return the SSC to service.

NRC Request

11. RG 1.177 Section 2.3.7.2 identified the key components of a configuration risk management program. The licensee has not identified whether the program applied under plant procedure CP-253 satisfies all aspects of RG 1.177, specifically:

- **The scope of structures, systems, and components (SCCs) included.**
- **Applicability if additional SSCs become inoperable or non-functional while in a risk-informed AOT.**
- **Consideration of external events and level 2 issues.**

The licensee is requested to clarify this issue.

FPC Response

11. CP-253, "Power Operation Risk Assessment and Management," applies to all on-line work activities and manages the resultant risk from these activities. The scope of SSCs included in CP-253 is consistent with that identified in Regulatory Guide 1.177. Specifically, this scope is all SSCs modeled in the CR-3 PRA in addition to all SSCs considered to be highly safety significant that are not modeled in the PRA. Work activities are screened for impact through quantitative and qualitative methods. The quantitative assessment is performed using the Equipment Out of Service (EOOS) program, a computer model that has the PSA as its basis. This assessment method provides risk impact by reviewing various combinations of EOOS in accordance with the Work Week Schedule. The qualitative assessment is performed on the equipment that is not modeled in the PSA and also takes into account other aspects that are not modeled, such as containment integrity and plant environmental conditions. Containment integrity is listed as one of the key safety functions, which address level 2 impacts. Qualitative assessments for external events address predicted adverse weather conditions such as hurricanes, internal flooding and maintenance external to the plant (i.e., maintenance to fossil plants at the Crystal River Energy Complex). When emergent work or an unplanned degradation is identified, a Senior Reactor Operator review is performed and an EOOS risk assessment (quantitative assessment) is performed if the affected component(s) are part of the EOOS model. If the component(s) are not part of the EOOS model, a qualitative risk assessment will be performed and documented in accordance with CP-253. Additional assessments against AOTs of Technical Specifications, Fire Protection Plan and the Offsite Dose Calculation Manual are also made, as applicable.

References:

1. *NRC to Crystal River Unit 3 Letter dated July 18, 2007, "Crystal River Unit No. 3, Request for Additional Information Regarding License Amendment Request No. 295 Rev. 0, Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times (TAC No. MD5241)*
2. *Crystal River Unit 3 to NRC Letter dated April 13, 2007, "Crystal River Unit-3 – License Amendment Request #295, Revision 0, Extension of Allowed Outage Time to Seven Days and Elimination of Second Completion Times Limiting Time"*

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 / LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #295, REVISION 0

EXTENSION OF ALLOWED OUTAGE TIME TO SEVEN DAYS AND ELIMINATION OF SECOND COMPLETION TIMES

ATTACHMENT B

LIST OF REGULATORY COMMITMENTS

List of Regulatory Commitments

The following table identifies those actions committed to by Florida Power Corporation (FPC) in this document. Any other actions discussed in the submittal represent intended or planned actions by FPC. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Supervisor, Licensing and Regulatory Programs of any questions regarding this document or any associated regulatory commitments.

Commitment	Due Date
CR-3 will perform procedure CP-253, "Power Operation Risk Assessment and Management," which requires both a deterministic and probabilistic evaluation of risk for the performance of all maintenance activities. This procedure uses the Level 1 PSA model to evaluate the impact of maintenance activities on core damage frequency. CR-3 will not plan any maintenance that results in "Higher Risk" (Orange Color Code) during an extended outage (greater than 72 hours) of the LPI, BS, DC or RW-DC System.	During extended (greater than 72 hours) preplanned outage on the LPI, BS, DC or RW-DC System
The Remote Shutdown Panel, the Appendix R Cooler and the opposite train of LPI, BS, DC, RW-DC, EFW, Auxiliary Feedwater System, Emergency Feedwater Initiation and Control System, HPI, and their power supplies (AC and DC) will be administratively designated as "protected" (i.e., no planned maintenance or discretionary equipment manipulation).	During extended (greater than 72 hours) preplanned outage on the LPI, BS, DC or RW-DC System
CR-3 will not initiate an extended preventive maintenance outage (greater than 72 hours) on the LPI, BS, DC or RW-DC System if adverse weather, as designated by Emergency Preparedness procedures, is anticipated.	During extended (greater than 72 hours) preplanned outage on the LPI, BS, DC or RW-DC System
When extended maintenance (greater than 72 hours) is scheduled on a train of the LPI or BS System, CR-3 will limit transient combustibles in the decay heat pump vault of the opposite train and establish a periodic fire watch of the decay heat pump vault of the opposite train.	During extended (greater than 72 hours) preplanned outage on the LPI or BS System
When extended maintenance (greater than 72 hours) is scheduled on a train of the DC or RW-DC System, CR-3 will limit transient combustibles in the seawater room and establish a periodic fire watch in the seawater room.	During extended (greater than 72 hours) preplanned outage on the DC or RW-DC System
When extended maintenance (greater than 72 hours) is scheduled on a train of the LPI, BS, DC or RW-DC System, CR-3 will limit transient combustibles and establish a periodic fire watch in the fire zones containing routed cables associated with the pressurizer PORV and PORV Block Valves.	During extended (greater than 72 hours) preplanned outage on the DC LPI, BS, DC or RW-DC System