NEXTera ENERGY DUANE ARNOLD

September 23, 2009

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

Duane Arnold Energy Center
Docket 50-331
License No. DPR-49

Clarification of Response to Request for Additional Information Regarding Severe Accident Mitigation Alternatives for Duane Arnold Energy Center

References:

1. Letter, Richard L. Anderson (FPL Energy Duane Arnold, LLC) to Document Control Desk (USNRC), "Duane Arnold Energy Center Application for Renewed Operating License (TSCR-109)" dated September 30, 2008, NG-08-0713
2. Letter, Charles Eccleston (USNRC) to Richard L. Anderson (FPL Energy Duane Arnold, LLC), "Request for Additional Information, Including a Revision to RAI 3.H, Regarding Severe Accident Mitigation Alternatives for the Duane Arnold Energy Center (TAC NO. MD9770)" dated June 25, 2009.
3. Letter, Richard L. Anderson (NextEra Energy Duane Arnold, LLC) to Document Control Desk (USNRC), "Response to Request for Additional Information Regarding Severe Accident Mitigation Alternatives for Duane Arnold Energy Center" dated July 9, 2009, NG-09-0514.
4. Letter, Charles Eccleston (USNRC) to Richard L. Anderson (FPL Energy Duane Arnold, LLC), " Clarification of Response to the Request for Additional Information Regarding Severe Accident Mitigation Alternatives Analysis for License Renewal of the Duane Arnold Energy Center (TAC NO. MD9770)" dated August 24, 2009.

By Reference 1, FPL Energy Duane Arnold, LLC submitted an application for a renewed Operating License, including an Environmental Report. By Reference 2, the U.S. Nuclear Regulatory Commission (NRC) Staff requested additional information regarding the Severe Accident Mitigation Alternative analysis provided in the Environmental Report. By Reference 3, NextEra Energy Duane Arnold (f/k/a FPL Energy Duane Arnold LLC) provided the responses to the Staff's request for additional information. By Reference 4, the U.S. Nuclear Regulatory Commission (NRC) Staff requested clarification to the responses provided in Reference 3.

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The enclosure to this letter contains the NextEra Energy Duane Arnold, LLC, responses to the Staff's requests for clarification of the responses to their request for additional information.

This letter contains the following new commitment:

> Any conforming changes, resulting from our responses, will be transmitted in the annual update of our License Renewal Application.

If you have any questions or require additional information, please contact Mr. Kenneth Putnam at (319) 851-7238.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on September 23, 2009.

Christopher R Costanzo

Christopher R. Costanzo
Vice President, Duane Arnold Energy Center
NextEra Energy Duane Arnold, LLC

Enclosure: DAEC Clarification of Previous Responses to NRC SAMA-Related Request for Additional Information

cc: Administrator, Region III, USNRC
Project Manager, DAEC, USNRC
Senior Resident Inspector, DAEC, USNRC
Project Manager, NRR - License Renewal
License Renewal Environmental Project Manager, USNRC
M. Rasmusson (State of Iowa)

SAMA RAI 1.b

NRC Requested Clarification:
Station Blackout (SBO) is stated to contribute 34% of the core damage frequency (CDF) while Loss of Offsite Power (LOSP) is stated to contribute 37% of the CDF. Discuss the reasons for the large fraction of LOSP CDF events that involve SBO.

DAEC Response:
The SBO event at DAEC impairs the plant's ability to cool the containment and to provide low pressure injection sources because it is a single unit site that does not have mitigating features found at other single unit and multiple unit plants. For example, the DAEC does not have: an alternate offsite emergency AC power source such as a dedicated line from a nearby hydroelectric dam; additional diesel generators for crosstie capability beyond the two safety related Emergency AC Diesels; nor high pressure core spray pumps with separate dedicated diesel generators. Consequently, most of the risk associated with a loss of offsite power event is created by the simultaneous loss of onsite emergency power.

SAMA RAI 1.e.i

NRC Requested Clarification:
The response indicates that Probabilistic Risk Assessment (PRA) Rev. 3A was completed in March 1995 with a CDF of 3.3E-05 per year, and that PRA Rev. 3B was completed in January 1996 with a CDF of 1.5E-05 year. However, DAEC Individual Plant Examination (IPE) RAI responses (DAEC letter dated June 26, 1995 provide a CDF of 1.5E-05 per year for PSA Rev. 3 and indicate a date of December 1994. Clarify this inconsistency and indicate any necessary changes to the identification of major changes made to the DAEC PRA in response to RAI 1.e.ii.

DAEC Response:
PRA 3A (aka PRA 3) was completed in March 1995, not December 1994, with a CDF of 3.3E-05.
PRA 3B was completed in January 1996 with a CDF of 1.5E-05.

The CDF of 1.5 E-05/year mentioned in the June 26, 1995 letter was the value of a sensitivity case run in May of 1995, which used the 3A PRA model and took into account control building modifications to improve flood response.

SAMA RAI 1.i

NRC Requested Clarification:
The response describes a self-assessment of the DAEC PRA program conducted in 2004. Describe this self-assessment in more detail, including the depth to which the PRA itself was reviewed.

DAEC Response:
The focus of the 2004 PRA self assessment was to ensure the DAEC PRA complied with applicable standards and to identify any potential program enhancement opportunities. It did not delve into the validity of the model.

All recommended enhancements dealt with programmatic elements. No model changes were recommended or implemented.

SAMA RAI 2.a

NRC Requested Clarification:
The response indicates that a peer review of the Level 2 PRA was conducted in 2007. Describe this review, including the scope, the reviewers, the findings, and the overall conclusions. Specifically address any review findings regarding the Level 2 sequence binning, release characterization, or need to update MAAP3.0B calculations.

DAEC Response:
The 2007 DAEC PRA peer review took place after the Environmental Report was prepared.

The review was conducted by industry experts knowledgeable of the requirements of the ASME PRA Standard with experience in performing activities related to the PRA Elements they reviewed. The review used the NEI 05-04 process and the ASME PRA Standard as clarified by the NRC in RG 1.200 Rev 1. This review was a full-scope review of all the technical elements of the internal events, at power PRA, and PRA Configuration Control.

As such, the review was a Gap analysis to identify actions to enhance the DAEC PRA so it conforms to the new ASME Standard. The review concluded that the sequence binning and release characterization met the ASME PRA Standard. The reviewers suggested that DAEC upgrade from MAAP3.0B to 4.x. To date, DAEC has not implemented this change.

SAMA RAI 2.b

NRC Requested Clarification:

- It is stated that source term category (STC) release timing (early <6 hours, intermediate 6-24 hours, and late >24 hours) is defined based on the time interval from accident initiation to general emergency declaration (which is given in Table 3.4.4-1.) The times given in this table for all of the intermediate STCs and one late STC (LL/L) do not agree with this definition. Address these discrepancies.

- The response indicates that the (Cesium Iodide) CsI release fraction decreases as the bins go from High (H) to very low (LL). As indicated in the RAI, this is not true for a number of the bins. Describe in more detail how sequences were assigned to release bins. Explain how sequences with high release fractions (>0.1) were assigned to the moderate bin, and how sequences with moderate release fractions (< 0.1, >0.01) are assigned to the low bin. Provide a description of the dominant sequences contributing to each release bin.

DAEC Response:
Item 1
The previous response to this RAI, should be revised to state that: "The early, intermediate, and late releases are defined in terms of the time interval from recognition of the emergency action level to the time of release; the time intervals are <6 hours, 6-24 hours, and ≥ 24 hours, respectively." The remainder of the NextEra July 2009 response is valid.

The release times in Table 3.4.3-2 and the Level 3 modeled emergency declaration times in Table 3.4.4-1 agree with this revised definition, except for the three late STCs H/L, M/L, and L/L. As described in the NextEra July 2009 response, the Level 3 analysis used the MAAP core damage time as an analog for the emergency declaration time. The Level 2 analysis gives general emergency times for the aforementioned late STC representative sequences of "~0" (after scram), which is in agreement with the revised time binning definition above. By selecting the general emergency times equal to the later core damage time in the Level 3 calculations for these late STC sequences, there is less time for evacuation and the dose-risk results are, as demonstrated in the NextEra July 2009 response, slightly conservative.

Item 2
The severe accident sequences were assigned to a H (high), M (medium), L (low), or LL (low-low) release magnitude bin by considering three fundamental variables, initial containment failure mode, water availability and reactor building effectiveness. Taken in the aggregate, the CsI release fraction decreases as the bins go from High (H) to very low (LL). If timing is also considered, this does not hold (i.e., the release fraction does not continually decrease from H early to LL late).

For those severe accident sequences where the containment remains substantially intact, the source term escape fraction was estimated to be 2E-4, for the initial hour of the release. For those sequences where there was no reactor building filtration or holdup effectiveness, the CsI maximum leakage escape fraction to the environment was 0.0048 in 24 hours, and the sequences were assigned to the L release bin. For those sequences where the reactor building was effective in removing some of the material, the CsI release fraction was estimated to be between 0.0001 and 0.001 and the sequences were assigned to the LL (low-low) release bin.

When energetic failures of the containment occur near the time of RPV failure, it was assumed (based on a spectrum of MAAP analyses) that a sufficient fraction of CsI was airborne and would result in a large CsI release. These sequences were assigned to the H release bin.

Sequences with previous containment failures followed by a drywell head failure were assigned to the H release bin.

Containment isolation failure sequences were treated conservatively and were assigned to the H release bin.

ATWS sequences where the wetwell initially fails resulted in the drywell head staying intact. These ATWS sequences were assigned to the M or L release bins depending on whether the reactor building was intact or not. ATWS sequences where a large or medium LOCA was present were assigned to the H release bin. ATWS sequences with loss of injection prior to containment failure were binned similarly to accident sequences involving loss of inventory makeup in which the reactor pressure remains high.

Binning of sequences where the containment flooding was successful was dependent upon the effectiveness of the reactor building, condenser, and/or turbine building. If these were not effective the sequence was assigned to the H release bin. If these were effective, the sequence was assigned to the M release bin.

Sequences involving wetwell airspace or wetwell vent failures were treated as scrubbed releases, and were assigned to the L or LL release bin.

Sequences where RPV injection failures initially involve less severe containment failures (e.g., wetwell failures) but lead to temperature induced drywell failures, were assigned to either the L or M release bin, depending upon the reactor building effectiveness. Releases during these events are considered small until the time the containment is dry. Afterwards, as the temperature increases late in the sequence, material located on the containment surfaces is released into the airspace, increasing the release magnitude. If the reactor building was not effective, the sequences were assigned to the M release bin, otherwise the sequences were assigned to the L release bin since the release would be reduced by an order of magnitude.

The MAAP cases referenced in Table 3.4.3-2 do not include the containment event nodal effects on source term magnitude, such as if water is available to the molten debris, or reactor building effectiveness. The Level 3 analysis used the MAAP cases which represented the greatest release fraction and would be conservative for the release fraction of that release bin.

SAMA RAI 3.d

NRC Requested Clarification:
The listing of conservatisms in the DAEC fire PRA model includes the statement that the DAEC Individual Plant Examination of External Events (IPEEE) fire analyses do not credit fire suppression or fire brigades. Clarify if this remains true for the current model. If so, describe the basis for the reduction in the Divisions I and II 4kV essential switchgear rooms CDF from that given in the IPEEE to the current values quoted in response to RAI 3.f.

DAEC Response to SAMA RAI 3.d
In the current model, if the IPEEE fire analysis core damage frequency for the fire compartment under consideration is 1E-7 or less, neither fire suppression nor the fire brigade is credited. Conversely, if the core damage frequency of the fire compartment exceeds 1E-7, then Fire Growth and Damage event trees are used. In the latter cases, the Fire Growth and Damage event trees do credit fire suppression and fire brigades depending on the type of fire and its initiation.

The use of the improved realism provided by the Fire Growth and Damage event trees resulted in reductions in CDF for the Divisions I and II 4KV essential switchgear rooms.

SAMA RAI 3.e

NRC Requested Clarification:
Provide a description of the dominant seismic CDF sequences beyond those provided in the response, i.e., down to a CDF of 1 E-08 per year.

DAEC Response:
1. G7-06 (5.03E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.7 and 0.9 g. This earthquake results in loss of offsite power and a failure to scram. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. This failure of HPCI leads to core damage at a high RPV pressure (Accident Class IVT).

2. G7-05 (4.64E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.7 and 0.9 g. This earthquake results in loss of offsite

power. The reactor is scrammed successfully. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. Without high pressure injection, RPV depressurization is required but fails, leading to core damage at high RPV pressure (Accident Class IA).

3. G7-12 (4.12E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.7 and 0.9 g. This earthquake results in wide-spread failure of safe shutdown equipment. Core damage occurs due to loss of coolant injection with the primary containment potentially damaged (Class G).

4. G8-12 (3.80E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.9 and 1g. This earthquake results in wide-spread failure of safe shutdown equipment. Core damage occurs due to loss of coolant injection with the primary containment potentially damaged (Class G).

5. G7-03 (3.31E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.7 and 0.9 g. This earthquake results in loss of offsite power. The reactor is scrammed successfully. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. Without high pressure injection, RPV depressurization is successfully initiated and low pressure injection is successful, but long term containment heat removal is lost leading to core damage induced high containment pressure (Accident Class IIT).

6. G7-04 (3.19E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.7 and 0.9 g. This earthquake results in loss of offsite power. The reactor is scrammed successfully. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. Without high pressure injection, RPV depressurization is successfully initiated but low pressure injection fails leading to core damage (Accident Class ID).

7. G4-05 (2.47E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.2 and 0.3 g. This earthquake results in a loss of offsite power. The reactor is scrammed successfully. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. Without high pressure injection, RPV depressurization is required but fails, leading to core damage at high RPV pressure (Accident Class IA).

8. G3-05 (2.34E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.12 and 0.2 g. This earthquake results in a loss of offsite power. The reactor is scrammed successfully. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. Without high pressure injection, RPV depressurization is required but fails, leading to core damage at high RPV pressure (Accident Class IA).

9. G5-10 (2.26E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.3 and 0.5 g. This earthquake results in the turbine lube oil tank failing and causing a major fire in the turbine building. The fire causes a station blackout. The reactor is scrammed successfully. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. Without high pressure injection, RPV depressurization is required but fails, leading to core damage at high RPV pressure (Accident Class IB2).

10. G6-05 (2.24E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.5 and 0.7 g. This earthquake results in a loss of offsite power. The reactor is scrammed successfully. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. Without high pressure injection, RPV depressurization is required but fails, leading to core damage at high RPV pressure (Accident Class IA).

11. G6-06 (1.92E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.5 and 0.7 g. This earthquake results in loss of offsite power and a failure to scram. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. This failure of high pressure injection leads to core damage at a high RPV pressure (Accident Class IVT).

12. G5-05 (1.79E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.3 and 0.5 g. This earthquake results in a loss of offsite power. The reactor is scrammed successfully. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. Without high pressure injection, RPV depressurization is required but fails, leading to core damage at high RPV pressure (Accident Class IA).

13. G8-06 (1.73E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.9 and 1g. This earthquake results in loss of offsite power and a failure to scram. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. This failure of high pressure injection leads to core damage at a high RPV pressure (Accident Class IVT).

14. G2-10 (1.71E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.05 and 0.12 g. This earthquake results in the turbine lube oil tank failing and causing a major fire in the turbine building. The fire causes a station blackout. The reactor is scrammed successfully. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. Without high pressure injection, RPV depressurization is required but fails, leading to core damage at high RPV pressure (Accident Class IB2).

15. G8-05 (1.58E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.9 and 1g. This earthquake results in a loss of offsite power. The reactor is scrammed successfully. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. Without high pressure injection, RPV depressurization is required but fails, leading to core damage at high RPV pressure (Accident Class IA).

16. G3-10 (1.38E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.12 and 0.2 g. This earthquake results in the turbine lube oil tank failing and causing a major fire in the turbine building. The fire causes a station blackout. The reactor is scrammed successfully. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. Without high pressure injection, RPV depressurization is required but fails, leading to core damage at high RPV pressure (Accident Class IB2).

17. G9-06 (1.18E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude beyond 1g. This earthquake results in loss of offsite power and a failure to scram. High pressure systems HPCI and RCIC are conservatively excluded as a

success path in the seismic model. This failure of high pressure injection leads to core damage at a high RPV pressure (Accident Class IVT).

18. G8-03 (1.14E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.9 and 1g. This earthquake results in loss of offsite power. The reactor is scrammed successfully. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. Without high pressure injection, RPV depressurization is successfully initiated and low pressure injection is successful, but long term containment heat removal is lost leading to core damage induced high containment pressure (Accident Class IIT).

19. G8-04 (1.09E-08): This seismic-induced core damage sequence is initiated by a seismic event with a magnitude between 0.9 and 1g. This earthquake results in loss of offsite power. The reactor is scrammed successfully. High pressure systems HPCI and RCIC are conservatively excluded as a success path in the seismic model. Without high pressure injection, RPV depressurization is successfully initiated but low pressure injection fails leading to core damage (Accident Class ID).

SAMA RAI 3.f

NRC Requested Clarification:
The failures listed in the Tables on pages 24 and 25 appear to be random failures that occur in the fire or seismic event. Identify and describe any of the listed failures that occur as a direct result of the initiating event. If none are included discuss why not.

DAEC Response:
None of the failures listed on pages 24 and 25 of the NextEra July 2009 response occur as a direct result of the initiating event.

The component importance measures listed on Page 24 are based on the failure of components within the damage sequence which would normally mitigate the impact of the initiating event if they did not fail. The maintenance terms listed on Page 25 are treated likewise.

Failures due to fire initiating events are not combined as a component importance measure since they are flagged as failed in the model when the damage frequency is calculated for that specific fire.

Failures due to seismic initiating events are system based point estimates of the potential damage evaluated for different seismic hazards.

The breakdown of initiating events for the seismic model and the fire model were provided on pages 22 and 23 respectively of the NextEra July 2009 response. As mentioned on page F-20 of the Environmental Report, the Seismic Margins Assessment performed at DAEC identified the turbine lube oil storage tank as the only potential seismic outlier. Treatment of this tank is discussed in the NextEra response to RAI 5d.

SAMA RAI 4.a

NRC Requested Clarification:
Clarify how the population growth rates were determined from the data, i.e., least squares fit, exponential fit, linear fit.

DAEC Response:
An exponential fit of the data was used to determine the population growth rates.

SAMA RAI 4.c

NRC Requested Clarification:
Confirm that the DAEC core inventory provided in Table 3.4.3-1 was based on a DAEC-specific analysis rather than a generic analysis, and the source (computer code) of the information.

DAEC Response:
The core inventory was based on a DAEC specific analysis conducted by General Electric during the power up-rate process. The ORIGEN2 code was utilized.

SAMA RAI 4.d

NRC Requested Clarification:
- It is noted on page F-25 of the original submittal that all releases were at ground level. Confirm that this is in error and that the statements in Sections 3.4.3 and 8.4 and the response to this RAI are correct.

- The response indicated that releases for STC M/I are from the stack via the wetwell vent. The release fractions for this STC given in Table 3.4.3-2 for Cs and I are 0.17. This seems high for releases from the wetwell vent which would be scrubbed by the suppression pool. Discuss this.

DAEC Response:
Item 1
The text on page F-25 of the original submittal is in error and should not have been included. It will be removed as part of the Annual License Renewal Update.

Item 2
As described in the July 2009 response to RAI 2b (subheading RELEASE FRACTIONS), Level 3 release fractions were obtained from the MAAP .TAB output files. The MAAP simulation for the analyzed STC M/I sequence (3A02) did not include scrubbing from the suppression pool. The Level 2 analysis reports the CsI release fraction to the environment for this sequence as 0.017 and notes that it was adjusted to account for a decontamination factor (DF) of 10 in the pool. The Level 2 manual adjustment for the DF in the pool was not incorporated in the Level 3 analysis. If this adjustment were made (i.e., reducing the Level 3 Cs and I release fractions for the M/I STC only), the population dose and cost risks would decrease from 19.8 person-rem and $76,700 to 18.3 person-rem and $68,200.

SAMA RAI 5.d

NRC Requested Clarification:

- Explain why failure of the turbine Lube Oil tank is not included in the list of important components nor in the important external event sequences provided in response to RAI 3.e and 3.f. Provide more information on this failure including: its modeling in the seismic analysis, the identified weakness in its support structure, and the potential for a cost-beneficial if uncertainties (increase of factor of 10 in seismic CDF) are considered.

- Review each of the dominant fire and seismic core damage sequences provided in the response to RAI 3.e to identify potential SAMAs that might address fire- or seismic-induced damage, such as adding/improving fire suppression systems, minor cable relocations, and minor structural modifications.

DAEC Response:

Item 1

The list of important components was based on their ability to mitigate initiating events, not on their failure as a direct result of an initiating event incurring additional risk. The sequences containing the tipping of the turbine lube oil tank resulting in a fire that occur at the lower seismic levels were not part of the dominant external event sequences listed in July 2009 response to RAI 3.e. Consequently, the turbine lube oil tank is included in neither the list of important components nor the important external event sequences.

The seismic walkdowns performed in support of the DAEC IPEEE and subsequent stress calculations showed the turbine lube oil support structure would likely buckle at the safe shutdown earthquake level. The turbine lube oil tank is on a five foot support structure located in a vault in the basement of the turbine building. The model assumes a low fragility for the support structure failing in an earthquake causing the tank to tip over, breaching the vault wall, spilling the oil into the turbine building, and the oil igniting causing a major turbine building fire. This condition is modeled as an event tree node for each seismic magnitude initiating events listed in NextEra's July 2009 response to RAI 3f.

A method to improve the seismic fragility of the support structure would be to add stiffeners to decrease the probability of a tank tip event. As described in the NextEra July 2009 response to the NRC RAIs, if the lube oil tank were modified to completely eliminate the risk, the design, fabrication, and installation of the stiffeners would have to cost less than $23,000 to be cost effective. This maximum benefit is below the minimum cost for a hardware change established by the Expert Panel ($100,000). If the CDF were assumed to be an order of magnitude greater than the maximum benefit of the modification would increase to $230,000, which would warrant consideration of the SAMA beyond Phase I screening. However, making such an assumption is contrary to the methodology used to assess all other SAMAs.

Item 2

The review of the dominant fire and seismic core damage sequences provided in the NextEra July 2009 response to RAI 3.e did not identify any potential SAMAs. The major sequences are an extreme magnitude earthquake, multiple fires in multiple compartments, fires in the control room, or fires in the essential and non-essential switchgear rooms.

SAMA RAI 5.f.v

NRC Requested Clarification:
As described, SAMP 704 involves using a portable AC generator to supply existing battery chargers. This does not provide an additional battery charger or an additional source of DC power called for by SAMA 3. It would appear that SAMP 707, identified in response to RAI 5.f.vi, utilizes a portable DC power supply. Address this.

DAEC Response:
The DAEC has three chargers, one for each battery and a swing charger which can be used for either battery. SAMP 704 allows for the use of these chargers in conjunction with a diesel generator to charge the batteries. Given that the chargers themselves do not appear on the important component list, and given that each battery has essentially two chargers, the procurement of an additional charger was not deemed necessary to meet the intent of this SAMA, which is to provide a method of charging the batteries with a diesel driven battery charger.

SAMP 707 does indeed rely upon a portable battery.

SAMA RAI 5.f.xiii

NRC Requested Clarification:
Clarify if the guidelines and their usage would effectively resulted in all elements of the two SAMAs being implemented.

DAEC Response:
The two SAMAs referenced in the request for clarification are:
- SAMA 158, prohibit any work in the switchgear room supporting the operating river water train during any maintenance on the river water system; and
- SAMA 159 - post a fire watch in the switchgear room supporting the operating river water train, or stage temporary hoses for implementation of AOP-410, Total Loss of River Water.

The two referenced work guidelines are:
- WPG-2, "On-Line Risk Management Guideline" and
- OMG-7, "Outage Risk Management Guidelines".

The referenced guidelines, in conjunction with other planning documents, have effectively resulted in reducing the risks associated with work in the plant. Work is not scheduled on select equipment and entry into certain areas is prohibited when a train of an important system is out of service. These prohibitions and considerations are not limited to work on the river water system, but extend to all important systems. Though their implementation reduces risk, no credit is taken for actions stemming from either guideline when calculating the CDF.

When applying the work guidelines, SAMA 158 is still implemented while SAMA 159 is not. However, the essential switchgear room's fire risk has decreased since the IPEEE as a result of the application of the Fire Growth and Damage event trees. (See clarification of RAI 3.d).

SAMA RAI 6.a

NRC Requested Clarification:
The response dismisses SAMA 41 on the basis that the reactor water cleanup (RWCU) system is not capable of being an alternate means of injection, and doesn't address the concern that the benefit was potentially under-estimated. Note that the response to RAI 6.f for this SAMA indicates that the expert panel considered that RWCU could be used for injection. Address this discrepancy. Also provide an appropriate estimate of the benefit associated with this modification.

DAEC Response:
SAMA 41 was assessed two different ways, which led to the responses in July 2009.

The cost of $1,300,000 provided in the NextEra July 2009 response to RAI 2f, was based on using the system's current pumps. These pumps develop only a small differential pressure so the source of water would have to be near reactor pressure. Thus the NextEra July 2009 response to RAI 2a stating we thought SAMA 41 was for decay heat removal rather than as an injection point.

The addition of pumps and piping capable of using a source of water with a low suction pressure would increase the cost of SAMA 41 by at least $2,700,000. The total cost of providing the capability of injecting water via the reactor water cleanup system would exceed $4,000,000. As this is in excess of the maximum attainable benefit, SAMA 41 is not cost beneficial.

Table 7.1.3-1 "DAEC Phase II SAMA Analysis" and Table 8.2-1 "DAEC Sensitivity Evaluation" will be revised as part of the Annual License Renewal Update.

SAMA RAI 6.c

NRC Requested Clarification:
The response indicates that the evaluation of SAMA 117 (increasing boron concentration or enrichment) assumed elimination of mechanical failures but not the human error associated with standby liquid control (SLC) injection. This SAMA would do just the opposite; it would allow more time for the operator to manually inject SLC, thereby reducing the operator error contribution. Mechanical failures could still occur. Provide an appropriately revised evaluation for this SAMA.

DAEC Response:
NextEra concurs that SAMA 117 would reduce human errors rather than mechanical failures.

If instead of reducing the CDF by 6.6% (Table 7.1.3-1 of Appendix F of the Environmental Report) using enriched boron could reduce the CDF by 9.9% (the worth of the operator's failure to inject stand-by liquid control early from Table 5.1-1), the benefit of SAMA 117 (Table 8.2-1) would increase to about $200,000 while the upper bound benefit would increase to about $500,000. Given that the cost estimate of implementing the SAMA is $400,000, this SAMA warrants further evaluation.

Consequently, the reassessment of SAMA 117 has been entered into the site's corrective action program.

SAMA 117 is not aging related.

SAMA RAI 6.d

NRC Requested Clarification:

- Describe the River Water System inlet valve control system in more detail including the portions in the normal base model, the backup controls added to the normal base model to create the revised base model to evaluate SAMA 164, and the function and modeling of the handswitches whose failures were eliminated to determine the SAMA benefit.

- The response to this RAI indicates that the normal base case has a CDF of 1.097E-05 per year. This is different from that given in the ER (1.08E-05 per year on p. F-10, and 1.09E-05 per year on p. F-14) and in response to the question during the May 27 conference call to discuss the RAIs (1.08E-05 per year.) Explain the various values quoted and confirm that consistent model quantifications were used when determining the benefit (due to reductions in population dose, offsite economic risk, and CDF) of each SAMA.

DAEC Response:

Item 1

The River Water System supplies water to the ESW/RHRSW basins and supplies makeup water to the Cooling Tower basins for evaporative losses. The valves can be opened in the following four ways:

1. Normally, one supply valve, controlled by an electro-pneumatic (E/P) converter, throttles the valve to maintain a specified Cooling Tower basin level, while the other valve is closed.
2. Either or both valves can be manually opened in the control room by a hand switch.
3. Each valve, controlled by a solenoid that is separate from the E/P converter, will open automatically when the ESW/RHRSW basin levels are low.
4. Each valve, controlled by a solenoid that is separate from the E/P converter, will open automatically when the Cooling Tower basin levels are low.

The normal base PRA model only credits the use of the E/P converter and the ability to manually open the River Water supply valves (i.e., methods 1 and 2). The revised base PRA model includes the automatic opening of the valves when the Cooling Tower basin levels are low. Including this automatic function reduced the worth of the handswitches thus eliminating SAMA 164 from consideration.

Item 2

A CDF of 1.097E-05 per year is obtained when the CDF of each individual sequence is calculated and the results are summed. This is the base case value used to evaluate the SAMA benefit.

Quantifying the base model after creating a "single top" yields a CDF of 1.08E-05 per year (p F-10). This lower value, as compared to that obtained from adding the results of individual sequences, is expected since some sequences have redundant cutsets.

The 1.09E-05 per year number, found in the Table of F-15, is in error. It should be 1.08 E-05. This will be corrected as part of the 2009 License Renewal update.

SAMA RAI 6.e

NRC Requested Clarification:
For those cost estimates that did not use the lower bound values, indicate what the expert panel estimates included, such as inflation, contingencies, and replacement power.

DAEC Response:
The expert panel discussed a conceptual design and degree of complexity to implement the SAMA. They then chose a similarly complex design modification that had already been implemented at DAEC. The actual cost of the project was then selected as the cost of the SAMA.

No adjustment was made due to inflation. Contingencies were not included when estimating the SAMA. Replacement power is not included in the cost of a design package; therefore it was not included in the SAMA estimates.