NEXTera ENERGY DUANE ARNOLD

February 12, 2013

NG-13-0056
10 CFR 50.90

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

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

Response to Request for Additional Information, License Amendment Request to Adopt National Fire Protection Association Standard 805, Performance-Based Standard For Fire Protection For Light Water Reactor Generating Plants

References:
1) License Amendment Request (TSCR-128): Transition to 10 CFR 50.48(c) - NFPA 805, Performance-Based Standard For Fire Protection For Light Water Reactor Generating Plants (2001 Edition), NG-11-0267, dated August 5, 2011 (ML11221A280)
2) Clarification of Information Contained in License Amendment Request (TSCR-128): Transition to 10 CFR 50.48(c) - NFPA 805, Performance-Based Standard For Fire Protection For Light Water Reactor Generating Plants (2001 Edition), NG-11-0384, dated October 14, 2011
3) Electronic Communication, ME6818 - Duane Arnold Adoption of NFPA-805 – Request for Additional Information – Round 2, dated December 5, 2012 (ML12340A450)
4) Electronic Communication, ME6818 – DAEC NFPA-805 Adoption – Record of Revisions to RAI Items, Expected Response Schedule, Participants List for Meetings on 17-18 December 2012, dated December 19, 2012 (ML12355A072)

In the Reference 1 letter, as clarified by Reference 2, NextEra Energy Duane Arnold, LLC (hereafter NextEra Energy Duane Arnold) submitted a License Amendment Request for the Duane Arnold Energy Center (DAEC) pursuant to 10 CFR 50.90. Subsequently, the NRC Staff requested, via Reference 3, additional information regarding that application.

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Per discussions with the Staff, as documented in Reference 4, the requested information will be provided in two submittals, one on February 12 and one on March 6, 2013. Attachment 1 to this letter contains the responses due on February 12, 2013.

This additional information does not impact the 10 CFR 50.92 evaluation of "No Significant Hazards Consideration" previously provided in the referenced application.

This additional information does not make changes to any existing commitments and makes the following new commitment.

RAI Response Number	Description
Probabilistic Risk Assessment 58.01	A full update of the NFPA 805 fire PRA application will be performed prior to transition. Periodic updates will occur when they are necessary or when significant changes have been made. The interval between updates will be no longer than five years

If you have any questions or require additional information, please contact Tom Byrne at 319-851-7929.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on February 12, 2013

Richard L. Anderson
Vice President, Duane Arnold Energy Center
NextEra Energy Duane Arnold, LLC

Attachment: Response to Request for Additional Information, License Amendment Request to Adopt National Fire Protection Association Standard 805, Performance-Based Standard For Fire Protection For Light Water Reactor Generating Plants

cc: NRC Regional Administrator
NRC Resident Inspector
NRC Project Manager
M. Rasmusson (State of Iowa)

Attachment to NG-13-0056

Response to Request for Additional Information, License Amendment Request to Adopt National Fire Protection Association Standard 805, Performance-Based Standard For Fire Protection For Light Water Reactor Generating Plants

68 pages follow

DAEC RAI PRA 12.01

In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 12 and provided the results of a sensitivity study of eliminating the credit for the control power transformer (CPT) for Scenarios 02B-F01 and 03A-D12. Credit for CPT is denoted by footnote 2 of Table 4.0-1 of the FSR which then appears to indicate that CPT credit was also used for scenario 02D-B01. Revision 4 of the FSR added footnote 6 to the Table indicating that this later scenario was not used in the final quantification. Explain why this scenario was not used in the final quantification.

RESPONSE:

The development of a fire PRA is an iterative process where conservative assumptions may over estimate the risk associated with a fire scenario. The naming scheme for the scenario indicates this scenario was a postulated hot gas layer scenario in PAU 02D. Per Appendix C, page C-1, of the Fire Scenario Report (0493080001.003), the initial assumptions in the hot gas layer scenario included the heat release rate contribution of two cable trays. Upon further review of the postulated scenario, PAU 02D does not contain fixed ignition sources. Postulated transient fires in the PAU do not include secondary combustibles based on the transient fire zone of influence. Based on Table C-2 of the Fire Scenario Report, it was estimated that 1500 KW was required for a hot gas layer to be caused by a transient fire. With the absence of secondary combustibles, the heat release rate associated with postulated 98th percentile transient fires is 317 KW (NUREG/CR-6850 Table G-1). Therefore, the postulated scenario was overly conservative and removed from the quantification.

DAEC PRA Question 02.01

In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 02. Expand on the response for the disposition of F&O 4-17 on supporting requirement (SR) PRM-B9. The disposition of this F&O addresses the modeling initiation and actuation logic for the main steam isolation valves (MSIVs) fail to close. Address the broader finding concerning inclusion of all the systems that are required for initiation and actuation of a system, including the presence of the conditions needed for automatic actuation (e.g., low vessel water level) and permissive and lockout signals that are required to complete actuation logic. Confirm that the initiation and actuation logic for all systems are appropriately included in the FPRA and that this inclusion meets the requirements of SR PRM-B9.

RESPONSE:

SR PRM-B9 requires that the systems analysis portion of the fire PRA be performed in accordance with HLR SY-A and HLR SY-B with the clarification that the SRs under HLR SY-A and HLR SY-B are to be addressed in the context of fire scenarios accounting for fire damage to equipment and cables. F&O 4-17 was associated with Part 2 SRs SY-B10 and SY-B15.

In the June 2010 Fire PRA Peer Review, the review team lacked confidence that system fault tree logic had been appropriately updated for fire since the most recent peer review for the FPIE PRA had not occurred prior to the start of the Fire PRA project. A major update of the internal events PRA was begun in 2008 to address findings from the 2007 peer review and to create a model that would serve as a basis for the Fire PRA being developed at the same time. Included in the documentation set for the update project is a notebook for the Common Actuation System (DAEC-PSA-SY-05.19, Rev 3.) This notebook documents development of fault tree logic related to instruments that provide actuation signals to the safety-related high and low pressure injection systems. Included systems are the High Pressure Coolant Injection system, the Reactor Core Isolation Cooling system, the Automatic Depressurization System, the Core Spray system, the Low Pressure Coolant Injection system, LPCI Loop Selection Logic, and the Emergency Diesel Generators.

Information relating to conformance to high level requirements SY-A and SY-B is provided in the Common Actuation Notebook. Regarding SY-B10 and SY-B15, which are the subject of F&O 4-17, fault tree logic for actuation instrumentation includes dependencies on sensor inputs for reactor pressure vessel parameters such as low level and high pressure; and, manual initiation of associated systems given failure of automatic initiation failure.

Therefore, although only the addition of logic associated with MSIV failure to close is mentioned in the disposition of finding F&O 4-17 in Table V-3 of the NFPA 805 License Amendment Request, actuation logic for other systems is indeed contained in the Fire PRA. This logic was developed in accordance with supporting requirements SY-B10

and SY-B15, or alternatively, with lesser detail when conditions specified in SY-A15 are met.

The fire PRA includes the cables for the combination of automatic actuation signals. The cables for these signals are grouped as "super components" (e.g., RHR SYS LOGIC 1R1201). The PRA accounted for fire damage to the signals by failing a surrogate component of the system that the signal provides input. For example, the super component RHR SYS LOGIC 1R1201 is mapped to the basic event for RHR Pump A (DRHR—CRMPP229A-1R--). In general the "super components" are mapped to several basic events. Table A-1 of the Fire Model Development Report (0493080001.002) provides the component to basic event mapping and Table I-1 provides the component and cable data relationships. These data relationships ensure that fire damage to equipment and cables for these signals are properly addressed in the Fire PRA consistent with SR PRM-B9.

The Fire PRA treatment for the containment isolation signal is an outlier because cables for the signal were not selected. The fire PRA treatment of the containment isolation signal is discussed in Section 4 of the Fire Model Development Report (0493080001.002). Additionally, further information regarding the treatment is provided in the response to PRA RAI 47.01.

DAEC RAI PRA 06.01

In your letter dated May 23, 2012 (ADAMS Accession No. ML12146A094) you responded to PRA RAI 06. PRA RAI 06 discussed the multi-compartment analysis (MCA) from 02E to 02B, and your response stated the following:

"When considering a hydrogen recombiner fire event, the fire is most likely to propagate to 02B through the east recombiner vault door and heating, ventilation, and air conditioning (HVAC) room door. The target set outside the HVAC room door in 02B does not contain FPRA cables."

a. The response did not address potential impact of fire propagation through the east recombiner door. Discuss the analysis of this potential pathway for this scenario.

b. In addition, since the fire area is a control rod drive (CRD) module area, discuss whether a potential impact on the CRD modules exists and was considered in the risk analysis. Describe the impact on the results of this MCA analysis.

RESPONSE:

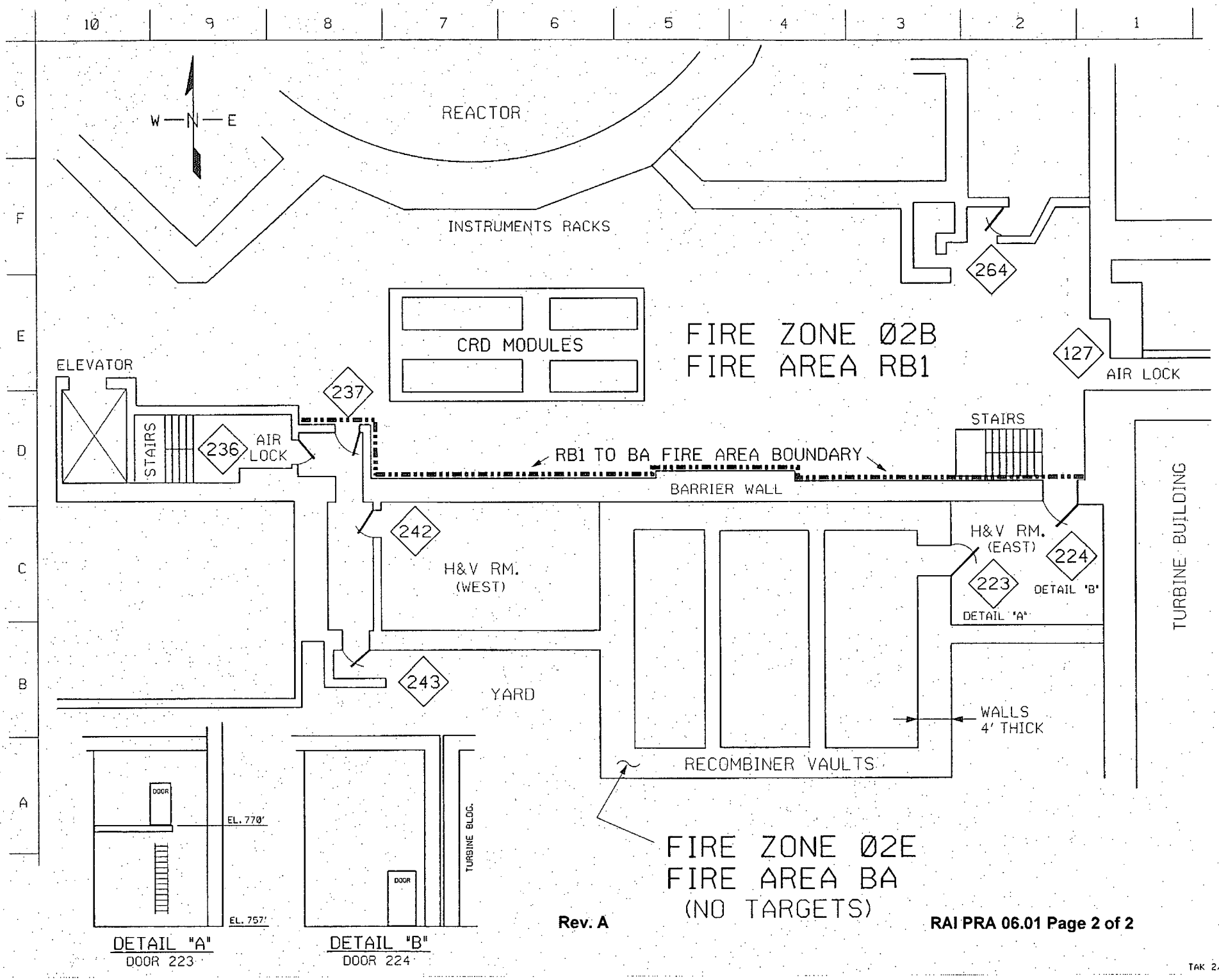
a. FHA-400 provides discussion of the construction of PAU 02E. PAU 02E is the Offgas Recombiner Room. The offgas condenser and recombiners are contained in a concrete vault and there are HVAC equipment rooms on the east and west sides of the vault. Access to the vault is via door 223 (location C2 on attached sketch) between the vault and east HVAC room. A door in each HVAC room opens into PAU 02B. The north wall of PAU 02E is continuous with PAU 02B.

   There are two multi compartment interactions considered for a hydrogen recombiner fire between PAUs 02E and 02B that involve doors.

   1. A hydrogen recombiner fire that is postulated to propagate into the east HVAC room via the failure of vault door 223 and subsequently into PAU 02B via failure of the east HVAC room door 224 (location C1 on attached sketch). There are no fire PRA targets outside the east HVAC room door.

   2. A hydrogen recombiner fire that is postulated to propagate into the west HVAC room via failure of the four foot thick vault wall and subsequently into PAU 02B via failure of the west HVAC room door 242 (location C7 on attached sketch) and the emergency airlock egress door 237 (location D8 on attached sketch). There are no fire PRA targets outside the west HVAC room door.

b. In the MCA for PAUs 02E and 02B, impacts to the CRD modules were not postulated. A fire propagation pathway does not exist from the doors to the CRD modules. Additionally, HGL concern does not exist because of the large volume of PAU 02B.



Rev. A

TAK 2/5/13

DAEC RAI PRA 10.01

In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) your responded to PRA RAI 10. The response to part (b) of PRA RAI 10 states "An examination of the impairment log, which has been electronic since 2008, reveals that the vast majority of impairments on these systems are not for corrective maintenance but for..." There is no clear statement if any of the fire protection systems credited in the license amendment request (LAR) have experienced outlier behavior relative to system unavailability or not. Describe whether the credited systems have experienced outlier behaviors and, if so, provide an assessment of the impact of this behavior on the FPRA total and delta risk results.

RESPONSE:

In the response to PRA RAI 10, Duane Arnold maintained that the Fire PRA satisfied CCI for FSS-D7. CCI recognizes that credited systems use generic estimates of total system unavailability and seeks confirmation that:

a) the credited system is installed and maintained in accordance with applicable codes and standards and,
b) the credited system is in a fully operational state during plant operation.

The Fire PRA uses the generic estimates from NUREG/CR-6850 (page P-6) for both wet pipe and deluge/pre-action systems. A comparison to these estimates is extremely difficult as unavailability and reliability calculations have not been performed. Additionally, NUREG/CR-6850 does not provide sufficient information to provide a comparison for outlier behavior (i.e., NUREG/CR-6850 does not provide generic unavailability data). Duane Arnold is not in a position to state unequivocally that the credited systems have not experienced outlier behavior and hence has not claimed that the Fire PRA has met the CCII criteria for outlier behavior which states:

c) the system has not experienced outlier behavior relative to system reliability.

Under the NFPA 805 Monitoring Program Duane Arnold will capture and track actual system data for credited systems.

Duane Arnold only credits suppression systems for multi-compartment interactions which are not a significant risk contributor to total plant risk. Therefore, the total plant risk is not sensitive to the uncertainty in suppression system unreliability and unavailability.

The Turbine Building Fire Area is the area where suppression systems have been credited for multi-compartment interactions in the fire risk evaluations. The analyses where the systems are credited do not include VFDR's; therefore, the Fire PRA delta risk is not sensitive to the uncertainty in suppression system unreliability and unavailability.

DAEC RAI PRA 14.01

In your letter dated May 23, 2012, (ADAMS Accession No. ML12146A094) you responded to PRA RAI 14. The response included added cable spreading room (CSR) analysis using NUREG/CR-6850 FIF and modeling from the CSR Risk evaluation report. This modeling considered prompt suppression, plant trip, offsite power, Division 1 and Division 2 availability, alternate shutdown capability (ASC) availability, and ASC mitigation. Provide further justification for the following modeling considerations:

a. For plant trip where there is no data, provide additional support for why the 50/50 split is assumed to be a point estimate for the case.

b. For offsite power, the point estimate case assumes 13% of the time offsite power will be lost based on offsite power cables routed in approximately 85 routing points of the total of 660 in the nuclear safety capability assessment (NCSA) database. The upper bound estimate is taken as 20%. Provide further support for the validity of these postulated split fractions.

c. The likelihood that Division 1 paths will be available from the control room (CR) depends on the resolution of PRA RAI 20. For the likelihood that the ASC will be impacted by the fire, a 50/50 split is used for both the point estimate and the upper bound. Discuss the reasonableness of this assumption, including fire modeling or fire modeling assumptions if applicable.

d. Furthermore, it appears that the CSR risk analysis credits prompt suppression for transient fires. However, per NUREG/CR-6850, Attachment P, prompt suppression can only be credited for hot work fire scenarios in which a continuous fire watch is present; this credit does not apply for transient fires. Provide justification on the application of credit for prompt suppression or reconsider the analysis following NUREG/CR-6850 guidance.

e. In the CSR Risk Report, the LERF was estimated to be 30% of CDF based on the CDF/LERF ratio from the full power internal events (FPIE) PRA. Use of the FPIE PRA to estimate FPRA LERF for this fire scenario is not applicable; rather the FPRA LERF should consider the FPRA model. Reconsider the LERF analysis for this fire scenario using a justified analysis and provide revised results.

RESPONSE:

In lieu of providing additional justification for the provided CSR analysis, an evaluation of the CSR was performed using the same methods employed throughout the fire PRA. These methods include postulating fires, identifying target sets, and quantifying the results. The evaluation of the CSR is documented in the new report 0493080001.007, DAEC Cable Spreading Room Fire Scenario and Quantification Report.

The evaluation assumes at least a plant trip for CSR fires, identifies the contribution from the loss of offsite power, assumes ASC is not available, does not credit prompt suppression for general transient fires, and quantifies LERF. Based on this evaluation, the CSR CDF is estimated as 1.26E-6/yr and the LERF is estimated as 3.69E -7/yr.

Fires in the CSR are not postulated to result in MCR abandonment because of habitability; however, fires may result in the loss of Division 1 and Division 2 equipment. Therefore, the delta risk for the CSR is based on those fire scenarios that may result in the loss of a sufficient set of controls that may result in the operators using ASC. The delta CDF for the CSR is calculated as 5.23E-10/yr and the delta LERF is calculated to be 3.91E-10/yr.

Based on the results of the evaluation, the CSR CDF and LERF increased from those included in the fire PRA. The CSR CDF increased from 5.7E-7/yr to 1.26E-6/yr and the CSR LERF increased from 1.7E-7/yr to 3.69E-7.yr. Delta risk from the CSR was not included in the original fire risk evaluation for CB1 because postulated fires did not result in the loss of a sufficient set of controls requiring ASC. Therefore, this evaluation calculates delta risk not previously included for CB1.

DAEC RAI PRA 16.01

In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 16. F&O 5-37 and 5-38 identify apparent weaknesses in identifying cues and associated instrumentation in the FPRA evaluations. The response to PRA RAI 16 only addressed the specific operator actions identified and stated that credit for these actions was not, in fact, credited. The disposition of related F&O 1-3 cites Table 3.3-1 for a listing of instrumentation relied upon for operator actions. However, this Table does not include the instrumentation specifically cited in the F&O, i.e. hotwell level and filter differential pressure (dp). It would appear that hot well level is needed for actions where PCS injection is occurring. For example, Table 3.3-1 includes operator action DCNDSTCNOP02----HE-- OPERATOR FAILS TO OPEN HOTWELL MAKEUP BYPASS LINE with RPV level the only cue identified.

a. Provide further information concerning the need for hotwell level and filter dp instrumentation.

b. Does Table 3.3-1 only include cues and instrumentation that appear in the safe-shutdown list because they are relied on to support safe-shutdown paths? If not, what is included in Table 3.3-1?

c. In addition, for several operator actions concerning loss of room cooling, the cue is stated as "Environmental cue given room heatup given loss of room cooling." If instrumentation is being relied on for the cue, describe whether it has been verified to be available for applicable fire scenarios.

d. Cues and instrumentation may be credited in non-safe-shutdown operator actions in the FPRA that do not appear in the safe-shutdown list. Describe how the affects of fire scenarios on this instrumentation are evaluated for credit in the FPRA.

RESPONSE:

The fire PRA identifies the following types of cues:

1. Instrumentation identified on the Safe Shutdown Equipment List (SSEL),

2. Environmental cues which do not require instrumentation such as a changes in control room lighting caused by a loss of offsite power, and

3. Procedure steps located within procedures that are initiated based on the available indications.

For example, the cues for actions related to maintaining RPV level are addressed by items 1 and 3. These actions are cued on low RPV level which is on the SSEL. Additional cues are provided by the RPV Control EOP (EOP 1) which is entered on the low RPV level cue. Between the RPV level cue and the procedure steps within the RPV

Level Control leg of EOP 1, the operators would have sufficient cues for the use of multiple injection systems, including Condensate.

a. Regarding hotwell level and filter dp instrumentation:

- Hotwell level is the identified cue for the internal events version of the operator action DCNDSTCNOP02----HE--, OPERATOR FAILS TO OPEN HOTWELL MAKEUP BYPASS LINE. As discussed above, the fire PRA credits RPV level along with procedural cues to provide cues for RPV injection.

- Filter dP is one of the multiple identified cues for the internal events version of the operator action DCBHV-NNOPFTSHV-HE--, OP FAILS TO START STANDBY CB HVAC TRAIN. In addition to filter dP, Control Room heat up was identified as a cue. Control room heatup is the environmental cue credited in the fire PRA.

b. The fire PRA identified a minimum set of instruments for operator action cues. The minimum set selected and identified in Table 3.3-1 of the Fire Model Development Report are safe shutdown instruments. The safe shutdown instruments are credited because these instruments provided sufficient cues along with environmental and procedural cues for the operator actions credited in the fire PRA .

c. The operator actions that identify room heatup as an environmental cue do not rely on instrumentation. These actions are:

- DCBHV-NNOPDORFANHE--: OP FAILS TO PROP OPEN CB DOORS OR START APP R FANS - On a loss of Control Building HVAC, the operators in the Control Room which is located in the Control Building would be able to feel the Control Room environment heating up. Therefore, an instrument cue is not required for the action.

- DCBHV-NNOPFTSHV-HE--: OP FAILS TO START STANDBY CB HVAC TRAIN – On a loss of Control Building HVAC, the operators in the Control Room which is located in the Control Building would be able to feel the Control Room environment heating up. Therefore, an instrument cue is not required for the action.

- DHPCI-CNOPOPENDRHE--: Operator Fails to Ventilate HPCI Room – For non fire events, the identified cue is Station Blackout conditions. For fire events, the Station Blackout condition would be the same cue. Therefore, an instrument cue is not required for the action.

- DPHVACNNOPPHDORSHE--: OPERATOR FAILS TO OPEN PUMPHOUSE DOORS – Operator rounds are credited to detect the increased temperature in the Pumphouse. The system window for the action is 2.25 hours. The cue is assessed as poor and additional delay time is included to account for operator rounds.

d. The fire PRA credited operator actions using the safe shutdown list instruments. In many instances the cue was assessed as degraded, because the cue for an action was considered inferred based on a safe shutdown list instrument. Instruments not on the safe shutdown list were not credited in the fire PRA.

DAEC RAI PRA 20.01

In your letter dated May 23, 2012, (ADAMS Accession No. ML12146A094) you responded to PRA RAI 20 and described an analysis of a fire in the corner of the CSR. Cables for only motor control center (MCC) 1B34 and 1B44 are mentioned and they are not both in the zone of influence (ZOI) of the postulated transient fire.

a. Explain why this situation is not a variance from deterministic requirement (VFDR) for which any retained risk should be summed with other retained VFDR risks.

b. Confirm that there are no other Division 1 cables other than those for MCC 1B34 in the CSR.

c. Provide justification for not considering the fire induced failure of cables for MCC 1B34 and other (than cables for MCC1B44) division 2 cables in the ZOI.

RESPONSE:

a. For Fire Area CB1, which includes the CSR, alternate shutdown capability is the credited safe shutdown strategy. VFDRs for fire area CB1 are associated with operator actions to establish alternate shutdown that do not take place at a primary control station. The operator action to switch control of MCC 1B44 occurs at the primary control station 1C388 as identified in Table G-1 of the LAR. Therefore, while the cables associated with MCC 1B34 and MCC 1B44 are located in the CSR, the situation is not a VFDR.

b. Division 1 cables associated with MCC 1B34 and control building chiller 1VCH001A are the only Division 1 cables located in the CSR. There are Division 2 cables associated with Division 1 components in the CSR. These have been reviewed and it was confirmed that a sufficient set of Division 1 components are available for the credited functions (e.g., reactor inventory, decay heat removal, AC and DC power, etc).

c. In clarification to the response to PRA RAI 20, the postulated transient fire scenario included the failure of both Division 1 and Division 2 components with cables within the ZOI of a postulated transient fire in the location of the 1B34 cables.

DAEC RAI PRA 31.01

In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 31. The disposition of F&O 1-1 in the LAR states that each multiple spurious operation (MSO) disposition was added to Table G-1 of the fire model development report (FMDR). In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 47 and stated that functional failure and MSO were considered in modeling the containment isolation valves (CIVs). However, review of Table G-1 noted that it did not include MSOs for CIVs. Clarify this discrepancy and provide the CIVs which may be missing from Table G-1 based on the disposition of F&O 1-1.

RESPONSE:

Table G-1 of the FMDR (0493080001.002) dispositions the BWROG NEI 00-01 MSO list consistent with the scenario description. The disposition to F&O 1-1 and the response to PRA RAI 31 discuss BWROG MSO scenarios 4B, 4C, and 4D with respect to containment over pressure. Given this, a specific MSO item for CIVs is not included in Table G-1.

The response to PRA RAI 47 provided a discussion of the fire PRA modeling of CIVs. MSO of CIVs is included in the PRA. The CIV discussion and fault tree logic is included in the DAEC Level 2 report, DAEC-PSA-L2-15 Appendix C.

The following spurious combinations are included in the fault tree with logic modeling the spurious events.

- AO 4300 and AO 4301 – Torus Purge Line Valves
- AO 4306 and AO 4308 – Torus Purge Line Valves
- AO 4311 and AO 4313 – Torus Purge Line Valves
- AO 4306 and AO 4307 – DW Purge Line Valves
- AO 4302 and AO 4303 – DW Purge Line Valves
- AO 4311 and AO 4312 – DW Purge Line Valves
- AO 3704 and AO 3705 – DW Floor Drain Valves
- AO 3728 and AO 3729 – DW Equipment Drain Valves

The following spurious combinations result in isolation failure only if there is a pipe break in the system. The fault tree logic does not include spurious events for these valves; however, the cable data is mapped to the fail to close events. Therefore, the

events for the valves were failed when cable damage is postulated. Appendix A and Appendix I of the FMDR provide the data relationship for basic events and cables.

- MO 4423 and MO 4424 – Main Steam Line Drain Valves
- MO 2700 and MO 2701 – RWCU Suction Valves
- MO 2000 and MO 2001 – RHR Loop A Drywell Spray Valves
- MO 1902 and MO 1903 – RHR Loop B Drywell Spray Valves
- MO 2069 and MO 2012 or MO 2015 – RHR Loop A Torus Suction Valves
- MO 1989 and MO 1913 or MO 1921 – RHR Loop B Torus Suction Valves
- MO 2005 and MO 2006 – RHR Loop A Torus Spray Valves
- MO 1932 and MO 1933 – RHR Loop B Torus Spray Valves
- MO 2005 and MO 2007 – RHR Loop A Test Line Valves
- MO 1932 and MO 1934 – RH Loop B Test Line Valves
- MO 1908 and MO 1909 – SDC Suction Valves
- MO 2100 and MO 2147 – CS Loop A Torus Suction Valves
- MO 2120 and MO 2146 – CS Loop B Torus Suction Valves
- MO 2112 and MO 2132 – CS Test Line Valves
- MO 2238 and MO 2239 – HPCI Steam Supply Valves
- MO 2321 and MO 2322 – HPCI Torus Suction Valves
- MO 2290A and MO 2290B – HPCI Exhaust Vacuum Breaker Valves
- MO 2400 and MO 2401 – RCIC Steam Supply Valves
- MO 2516 and MO 2517 – RCIC Torus Suction Valves

The fault tree logic for some of the above valve combinations includes a basic event for the conditional probability of the valve initial position. This event would not be applicable for fire induced spurious events and is a non conservatism in the model logic for fire events. However, fire induced spurious failure of these valve combinations resulting in isolation failure is negligible in the PRA because of the low probability of the pipe break events (see DAEC-PSA-L2-15 Appendix C).

DAEC RAI PRA 32.01

In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 32.

a. The response does not mention MO-2010. Clarify if V19-048 is modeled in the FPRA as open and requiring operator action to close, and, also, describe whether it is a model surrogate for MO-2010.

b. The response states that closure of this valve is not a required recovery action (RA) to meet the risk criteria for CR abandonment, and is therefore only credited for defense-in-depth (DID). The response also stated that closure of V19-048 is prescribed for a fire in RB1 and RB3. Describe the risk impact if V19-048 cannot be closed.

RESPONSE:

a. The RHR cross tie motor operated valve, MO-2010, is modeled in the PRA with failure modes of "fail to open" and "fail to remain open." The RHR cross tie manual, V19-0048, is modeled in the PRA with the failure mode "fail to remain open." Any one of these failure modes would fail the RHR cross tie function.

   For Fire Areas RB1 and RB3, the operator action to close V19-0048 would fail the RHR cross tie function. To simulate this action, MO-2010 is included in the Fire PRA as a surrogate for the action and MO-2010 is assumed to fail in the closed position for all fires initiating in Fire Areas RB1 and RB3.

b. V19-0048 is required to be closed in Fire Areas RB1 and RB3 for fires that may result in the inability to isolate the RHR cross tie line from the Control Room when spurious valve operation may result in flow diversion. Report 0027-0042-000-004 Duane Arnold Energy Center Fire Risk Evaluations (FRE Report) documents the VFDRs for MO-2010 and potential flow diversions, as well as the postulated fires in which MO-2010 and flow diversion is a possibility.

   In RB1, VFDR SSA-RB1-03 is associated with MO-2010 and VFDR SSA-RB1-13 and VFDR SSA-RB1-30 are associated with potential flow diversions. From Section 2.1.2 of the FRE report for RB1, fire scenario 02D-D01 may result in damage to MO-2010 and cause flow diversions. From Table 3-1, the CDF for fire scenario 02D-D01 is 6.54E-09/yr and the delta CDF is 5.83E-09/yr. From Table 3-2, the LERF for fire scenario 02D-D01 is 1.07E-09/yr and the delta LERF is 9.21E-10/yr. Therefore, the risk impact if V19-0048 cannot be closed is negligible compared to total plant risk.

   In RB3, VFDR SSA-RB3-04 is associated with MO-2010 and VFDR SSA-RB3-07 is associated with potential flow diversions. From Section 2.1.2 of the FRE report for RB3, there is not a fire scenario associated with VFDR SSA-RB3-07. Therefore, there is not risk impact if V19-0048 cannot be closed.

DAEC RAI PRA 39.01

In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 39 and stated "None of the fire response actions or the FPIE actions retained for the Fire PRA were considered complex enough to justify the use of the upper bound ASEP curve." ASME PRA standard (ASME/ANS RA-Sa-2009) SR HR-G3 requires that when estimating human error probabilities (HEPs) the impact of ten specifically defined plant and scenario specific shaping factors should be evaluated in order to meet CC-II. The accident sequence evaluation program (ASEP), which was used for analysis of cognition error when time available for recovery is less than 60 minutes, does not consider these shaping factors. It Is not clear whether ASEP, as it was applied, yields more conservative results than cause-based decision tree method (CBDTM), which was used for cognition error when time available for recovery is greater than 60 minutes. Application of ASEP to this specific set of cognitive errors using the lower bound diagnosis curve (Figure 7-1) provided in NUREG/CR-4772, "Accident Sequence Evaluation Program Human Reliability Analysis Procedure," appears to be too optimistic. Page 7-1 of NUREG/CR-4772 states that: "critical parameter which operating personnel must commit to memory, use the lower bound values in Figure 7-1.... only if those parameters can be classified as skill-based behavior per Table 2-1, otherwise use the nominal values." (The ASEP approach defines skill-based actions as being highly practiced). Use of the lower bound rather than the nominal curve to determine these values for FPRA human failure events (HFEs) is questionable as fire RAs are more complex and less practiced than the internal event RAs addressed by ASEP, even when offset with time reductions of 10 and 20 minutes (for CR and ex-CR actions). Either, demonstrate how SR HR-G3 is being met using ASEP to determine the HEP for this set (i.e., under 60 minutes available) of cognitive errors, show that treatment of these errors would have negligible impact on the FPRA, or determine the impact of this treatment of cognitive error on fire CDF and LERF, and Δ CDF and Δ LERF.

RESPONSE:

As stated in the RAI, the accident sequence evaluation program (ASEP) does not consider performance shaping factors directly. However, the typical cognitive error assessment method for DAEC Fire PRA actions with limited time available for recovery (Trec values less than 60 minutes) was not ASEP alone but ASEP + CBDTM. CBDTM was used to develop cognitive error probabilities because it includes detailed evaluations of performance shaping factors and the operator-plant and operator-procedure interfaces. However, it was concluded that the CBDTM did not adequately limit credit for short term actions, so the ASEP result for cognitive error was added to the CBDTM result for cognitive error in order to reflect the reduction in credit appropriate for actions with limited time.

SR HR-G3 is considered to be met by application of the CBDTM. Only two actions had cognitive error assessments based on ASEP alone:

- DCBHV-NNOPFTSHV-HE--, OP FAILS TO START STANDBY CB HVAC TRAIN

- DWELLWDNOPELLWTRHE--, OP FAILS TO MAXIMIZE WELL WATER TO MAINTAIN CONDENSER

The assessment of DCBHV-NNOPFTSHV-HE-- did not include CBDTM. If CBDTM were used instead of ASEP, the HEP would increase from 2.8E-3 to 3.4E-3 (a 21% increase). The fire PRA version of this action (FCBHV-NNOPFTSHV-HERX) does not appear in the fire PRA cutsets.

DWELLWDNOPELLWTRHE-- had a CBDTM assessment but the HEP was calculated based on ASEP alone. The HEP for this action would increase from 1.6E-3 to 1.9E-3 if CBDTM were used (a 19% increase). Use of CBDTM for this action does not result in a noticeable increase in the total fire risk.

DAEC RAI PRA 41.01

In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 41 and provided a general discussion on dampers. In the LAR it is noted for engineering evaluation ID FPE-B97-052, "FIRE BARRIER EVALUATION 10A TO 12B," Attachment C page 18, that some ventilation ducts have had the fire dampers sealed open. However, Table C-3 of the updated FSR takes credit for a fire damper between PAU 10A and PAU 12B. Discuss the FPRA modeling assumptions and impacts of sealed open versus unsealed fire dampers between these two fire areas.

RESPONSE:

As discussed in PRA RAI 41, the ventilation communication paths between fire zones was included in the multi compartment analysis (MCA) by allowing hot gas to migrate to the fire zone communicated with. If the ventilation path does not include a fire damper or includes a sealed open fire damper the hot gas is assumed to migrate to the fire zone communicated with. If the ventilation path does include a fire damper the hot gas migrates to the fire zone communicated with given failure of the fire damper.

In the case of the ventilation interaction between PAU 10A and 12B, Per FPE-B97-052 the ventilation ducts pass from the essential switchgear rooms through PAU 10A to PAU 12B. There are no openings into the ducts from PAU 10A. The inclusion of the damper interaction in the MCA is not accurate. The MCA interaction from 10A to 12B included consideration of penetration failures which encompass the closed ductwork. Because of the interaction configuration and that the interaction was not included in the quantification, the inclusion of the damper interaction and damper failure probability does not change the results.

A review of plant drawings and operating practices was performed to ensure no other instances were identified in which an abandoned or non functional damper is included as a MCA interaction between two PAUs.

DAEC RAI PRA 43.01

In your letter dated May 23, 2012, (ADAMS Accession No. ML12146A094) you responded to PRA RAI 43 and noted that high pressure core injection (HPCI) and reactor core isolation cooling (RCIC) are not credited in the NSCA; therefore it is necessary to lower reactor pressure for inventory control.

a. Discuss the assessment performed for fire-related affects, on the depressurization function for essential switchgear room fire MCAs.

b. Discuss the specific reason that HPCI and RCIC failed during these scenarios. Was it the loss of DC power due to battery depletion, fire damage to DC power, fire damage to HPCI and RCIC or other reasons?

c. Discuss how the ability to depressurize the reactor is modeled for these fire scenarios, including dependencies on DC power and operator actions.

RESPONSE:

a. For the essential switchgear room fire areas, CB2 and CB3, there is not a VFDR associated with the depressurization function. Appendix C of the Fire Scenario Report documents the MCA which includes the essential switchgear rooms. The MCA for the essential switchgear rooms resulted in these interactions being screened based on CDF (see Table C-3 PAU 10E and 10F entries).

b. HPCI and RCIC were assumed failed in several scenarios in the essential switchgear rooms for a number of potential different reasons:

   - Postulated fire scenarios that result in a consequential station blackout because of postulated cable damage and random failures do not credit offsite power recovery. Therefore, HPCI and RCIC are failed after battery depletion.

   - Cables for the turbine exhaust vacuum breaker isolation valves (MO 2290A and MO 2290B which are in series) are located in each essential switchgear room. Spurious closure of an isolation valve due to postulated cable damage in a fire scenario was assumed to result in HPCI and RCIC failure.

   - HPCI and RCIC logics have DC power dependencies for both divisions. If any of these cables were postulated damaged in a fire scenario then HPCI and RCIC were assumed to be failed.

c. The depressurization model includes relief valve failures and DC power failures. The model includes automatic and manual initiation. One division of DC power is required for automatic or manual initiation. As discussed in part (a) of this response, a VFDR is not associated with the depressurization function; therefore, postulated fire damage alone in an essential switchgear room does not result in failure of the depressurization function.

DAEC RAI PRA 44.01

In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 44 which included the following text:

"In conclusion, the delta risk calculations in the fire risk calculations would only be affected if:

a) the larger zone of influence included additional VFDRs, or b) the larger zone of influence included a VFDR in question that had not been affected in that particular fire scenario because of a smaller zone of influence."

Clarify what b) means.

RESPONSE:

The application of the NUREG/CR-6850 transient heat release rate resulted in a larger zone of influence which could result in additional target damage. A noticeable change in the delta risk calculations would occur only if:

a. The larger zone of influence resulted in additional targets that included VFDRs not included as part of the initial target set, or

b. The larger zone of influence resulted in additional targets that when included with a VFDR (i.e., part of the initial target set) a noticeable risk increase resulted.

An example of the second case would be a scenario where the initial target set included a VFDR for HPCI cables but did not included RCIC cables. In this case the delta risk may not be very large because alternate high pressure injection was available in the variant case. Now as a result of a larger zone of influence, the new target set now includes the VFDR for HPCI and RCIC cables. In this case the delta risk may be larger because the variant case would not have high pressure injection.

DAEC RAI PRA 47.01

In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 47 and noted that cables were not traced for the containment isolation signal (CIS) system, but were traced for CIVs. NUREG/CR-6850, Section 3.5.2.2, provides high level guidance on addressing the CIS. With respect to modeling the containment instrumentation and systems which are important for containment modeling, address the following:

a. Describe the failure modes modeled for CIS if fire damages these cables. Describe whether this has an impact on the CIVs "failure to receive isolation signal" probability modeled for random failures.

b. CIS impact appears to be manually excluded for battery room fires according to the FSR. Explain this modeling assumption.

c. In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 57 which provides a basic event "Containment Isolation Signal Fails." Describe whether this is a system-level basic event which models impacts on multiple CIVs as discussed in NUREG/CR-6850, Section 3.5.2.2.

d. In addition, a review of the instrumentation available to operators in Table 3.3-1 of the fire model development report did not find drywell pressure. Clarify if drywell pressure instrumentation fire impacts are included in the FPRA. If not, discuss why not.

RESPONSE:

a. Consistent with NUREG/CR-6850 Section 3.5.2.2, the CIS is modeled as a "dummy" component for the system logic signals. If fire damage was assumed for cables associated with the CIS then the automatic CIS was assumed failed. That is, only manual isolation of CIVs was credited. The PRA logic models the CIS and the CIVs independently (see Appendix C.2 of DAEC-PSA-L2-15, Rev. 3). Therefore, failure to receive an automatic isolation signal does not affect the random failure probability of the CIVs. This modeling is used for the CIVs listed in Table C.2-2 of DAEC-PSA-L2-15, Rev. 3.

   The CIVs listed in Table C.2-3 of DAEC-PSA-L2-15, Rev. 3 do not include explicit modeling of the CIS. These valves are in closed loop systems and require piping failure for a release path. Given this, fire induced isolation failure of these valve combinations resulting is negligible in the PRA because of the low probability of the pipe break events (see DAEC-PSA-L2-15 Appendix C).

b. As discussed above, the CIS is a "dummy" event in the fault tree logic that represents the redundant signal and divisions associated with the CIS. Therefore, the CIS was assumed available for a battery fire in the battery rooms (PAUs 10B and 10D) given the loss of a single DC power supply.

c. Consistent with NUREG/CR-6850 Section 3.5.2.2, the CIS is a "dummy" event for the system wide logic signals. Fire impacts on the CIS are included for the CIVs listed in Table C.2-2 of DAEC-PSA-L2-15, Rev. 3.

   CIVs listed in Table C.2-3 of DAEC-PSA-L2-15, Rev. 3 are discussed in the response to part "a" of this response.

d. Drywell pressure was not a credited cue for operator actions when considering fire impacts. Fire impact on drywell pressure instrumentation is included for system automatic initiation signals. Response to RAI PRA 02.01 provides further discussion on the modeling of fire impacts on these signals.

DAEC RAI PRA 48.01

In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 48.

a. The response to the RAI appears to contain a conflict. In the first paragraph it states "Process monitoring of suppression (or torus) pool level is required for the ASCS" while the second paragraph states that since the monitoring of suppression pool level is not called for in AOP 915,"... the Fire PRA does not credit suppression pool level instrumentation for any of the operator actions." Clarify why suppression pool level monitoring is not included in AOP 915, and therefore not included in the PRA, yet is included as required for the ASCS in the updated final safety analysis report (UFSAR). Review of Table G-1 in the LAR also noted that there is a primary control station action to monitor torus level (CB1 LI 4363A).

b. Further, the torus pressure instrumentation was noted to be monitored in Attachment C of the LAR for process monitoring. However, a review of Table G-1 in the LAR does not show torus pressure as being involved in any RAs or activities occurring at a primary control station. Describe if torus pressure is modeled in the FPRA. If not, describe why not.

RESPONSE:

a. The UFSAR reflects the design and licensing of the ASCS per Appendix R requirements and guidance. The minimum level of process monitoring for the NFPA 805 NSCA was selected based on guidance similar to that provided in NRC Information Notice 84-09. There is no VFDR in any fire area, including the CB1 fire area, that require torus level indication as an operator cue to perform a recovery action credited by the Fire PRA. The primary control station action in Table G-1 of the LAR is the action to transfer torus level indication to the primary control station. The action does not require torus level as an operator cue to perform the transfer.

b. It was identified during the first audit at DAEC that torus pressure indication was included in the minimum set of processing monitoring in error. RWA01648614-10 is tracking removal of torus pressure indication from the minimum set of process monitoring. Per transition Project Instruction PI-03-002 "Component Selection & Logic Development", the minimum set of instruments is based on the guidance provided in IE Information Notice No 84-09, which does not include torus pressure.

Torus pressure indication is not available at the primary control station. There are no control room abandonment VFDRs that require torus pressure indication as an operator cue to perform a recovery action credited by the Fire PRA.

Torus pressure indication is modeled in the Fire PRA and is credited as an operator cue for non-abandonment operator actions. It is not credited as an operator cue for any non-abandonment recovery actions.

DAEC PRA 55.01

In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 55 and stated:

"Given the recovery actions to establish alternate shutdown were treated as being completely dependent, an HRA was not required for those actions not required in the short term given the short term actions are most likely impacted by fire events."

Clarify what is meant by "an HRA". Describe if it is the assessment of the impact of fire on the non-fire human reliability analysis (HRA). Describe whether the HRA is considered for long term RAs for situations where short term RAs are successful.

RESPONSE:

Per the ASME/ANS RA-Sa-2009 standard, HRA is, "a structured approach used to identify potential human failure events and to systematically estimate the probability of those events using data, models, or expert judgment." In the context of fire PRA, HRA includes the assessment of the impacts of fire to those non-fire human failure events related to normal, emergency, and abnormal operating procedures, as well as those NFPA 805 fire recovery actions identified in Table G-1 of the LAR. Appendix E of the Fire Scenario Report (0493080001.003) describes the HRA process used to address the impacts of fire on non-fire human failure events.

For non-fire human failure events, the HRA dependency analysis considers the relationship between short term and long term actions.

The operator actions in Table G-1 are those fire recovery actions and Primary Control Station (PCS) actions required to establish alternate shutdown capability. In lieu of performing a detailed HEP calculation for each of these actions identified in Table G-1, the time available, location, complexity, and fire impacts of each action in AOP 915 was used in order to develop an understanding of the expected HEPs. Based on the information in AOP 915, the PCS actions and fire recovery actions were estimated to be bounded by the short term fire recovery action to locally start the standby diesel generator. That is the time available, location, complexity and fire impacts for the other actions would result in a lower HEP than that for the action to locally start the standby diesel generator.

A detailed HEP calculation was performed for the short term fire recovery action to locally start the standby diesel generator and the failure probability was estimated as 5E-2 (see Table E-5 of the Fire Scenario Report). As discussed above, this fire recovery action would bound the PCS actions when considering time available, location, complexity and fire impacts. As for the other fire recovery actions, each of these actions is not required in the first hour as these actions are required for functions related to the long term success of alternate shutdown. To ensure that the short term fire recovery action HRA bounded the long term fire recovery actions, a detailed HEP calculation was developed for the fire recovery action performed in support of establishing torus cooling

(locally close manual valve V-19-0048). The probability of this long term fire recovery action was estimated as 1E-3 (see Table E-5 of the Fire Scenario Report).

Each of the PCS and fire recovery actions was considered to be completely dependent on the remaining actions. That is, the individual HEP for a given action is applied and the remaining actions occurring within a cutset with the given action are set to 1.0. In order to be bounding, the action with the highest HEP (usually the action with the shortest system window) retains its individual HEP value. With the remaining actions having HEPs of 1.0 due to complete dependency, the operator error contribution to sequences containing these actions is equal to the HEP for the action with the highest HEP. Given the short term fire recovery action bounded the PCS and long term fire recovery actions, the fire PRA applied the short term fire recovery action failure probability as a single failure probability for the fire recovery actions to establish alternate shutdown.

DAEC RAI PRA 57.01

In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 57 and stated that the HRA for the Level 1 and Level 2 operator action to manually depressurize relies on reactor pressure vessel (RPV) level as the instrument cue which has been verified available in each fire area. However, experience shows (i.e., Fukushima) that if there is boiling occurring in the reference legs of the reactor, which may occur during a severe accident (in Level 1 portion of analysis) that water level instrumentation provides nonconservative water level indication. Describe the basis for relying on RPV level as the instrument cue in the Level 2 model even if the instrumentation is not impacted by the fire, but is potentially providing incorrect information due to the above condition.

RESPONSE:

The original response to 57 discusses that no credit is provided for AC power recovery, so fires that lead to station blackout conditions, as the Fukushima plants experienced after an earthquake and tsunami, are assumed to cause core damage. At Fukushima, station blackout conditions created elevated temperatures and pressures in the containments and resulted in boiling in the vessel level instrument reference legs after many hours with no injection. The response to RAI 57 dated April 23, 2012 states that to avoid core damage RPV depressurization must occur in the first 2 hours of the event or before reaching containment conditions that would cause boiling in the reference legs.

The model relies upon the RPV level instruments for human actions appropriately and conservatively when considering the Fukushima event.

DAEC RAI PRA 58.01

In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 58. In reviewing the response, PSAG-2 does not appear to require reviewing changes to the FPIE model for appropriate inclusion in the FPRA for the NPFA 805 application. Describe the process for reviewing changes to the FPIE for inclusion in the FPRA. Confirm that such a review has or will be done prior to transition and will be performed after transition on some periodicity.

RESPONSE:

PSAG-2 has been superseded by the following fleet PRA program procedures and site procedures that meet RG 1.200 Revision 2 "MU" supporting requirements:

- EN-AA-105, "*Probabilistic Risk Assessment (PRA) Program*"
- EN-AA-105-1000, "*PRA Configuration Control and Model Maintenance*"
- EN-AA-105-1000 (DAEC), "*PRA Configuration Control and Model Maintenance (DAEC) Specific Information*".

The general process for reviewing changes to the FPIE for inclusion in the FPRA is described in fleet procedure EN-AA-105-1000 and the site specific procedure EN-AA-105-1000 (DAEC).

Section 2.0, of the fleet procedure EN-AA-105 states, "Internal hazards such as internal flooding or internal fire events are treated as internal events." PRA application updates are addressed in Section 3.2, of site specific procedure EN-AA-105-1000 (DAEC) which states, "Model updates shall update the appropriate application and activities that are based on the PRA in accordance with the guidance provided by the PSAGs and PTGs."

A full update of the NFPA 805 fire PRA application has not been performed but will be performed prior to transition. Periodic updates will occur when they are necessary or when significant changes have been made. The interval between updates will be no longer than five years.

DAEC RAI PRA 60.01

a. For a number of VFDRs, a PRA quantification had not been performed. Clarify if the criteria used to determine that a VFDR did not require a PRA quantification was the ZOI, or whether there were other considerations in the determination such as multiple concurrent shorts, fire zones, etc.

b. If a fire scenario involves more than one VFDR, describe whether the delta risk for the ignition source is simply the sum of the individual VFDR delta risks, or does it include synergistic affects from cables and equipment which may all be simultaneously failed by one fire.

c. In addition, the Fire Risk Evaluation report notes that scenarios that do not result in CR abandonment were not considered as part of the delta risk calculations. Explain why the non-abandonment scenarios are not included in the delta risk calculation.

d. In your letter dated April 23, 2012, (ADAMS Accession No. ML12117A052) you responded to PRA RAI 60 and indicated that RAs required to establish alternate shutdown capability were included in the FPRA, however, in LAR, Attachment G, it indicates that some RAs have been specifically included in the FPRA, while others have not been included. Clarify this apparent discrepancy. In addition, describe how the RAs were included in the delta-risk calculations for the CR abandonment scenario?

RESPONSE:

a. Section 5.3 of Report Number 0027-0042-000-004 Duane Arnold Energy Center Fire Risk Evaluations provides a high level description of the fire risk evaluations. Section 2.1.2.1 of each attachment describes the effect of each VFDR on the fire PRA. Some VFDRs did not affect the fire PRA. This determination was made considering:

- Multiple concurrent hot shorts were required for the VFDR and the subject cables were routed such that a postulated ignition source would not include all of the cables. This could be because the cables were in separate fire zones. For example, this is the case for some VFDRs in Fire Areas CB1, RB1 and RB3.

- Cables for redundant equipment were required for the VFDR and the subject cables were routed with sufficient separation such that a postulated ignition source would not include all of the cables. This could be because the cables were in separate fire zones. For example, this is the case for several VFDRs in Fire Area RB1.

- A proposed modification makes the VFDR condition compliant (i.e., Fire Area TB1).

b. When a fire scenario included more than one VFDR, the delta risk calculation includes the synergistic affects from the cables and equipment associated with each VFDR included in the fire scenario. Table 2-2 in the attachments to Report Number

0027-0042-000-004 Duane Arnold Energy Center Fire Risk Evaluations identifies each VFDR associated with the fire scenario.

c. The credited safe shutdown method is the Alternate Shutdown Capability (ASC). Therefore, the applicable VFDRs associated with the fire area are applicable to establishing ASC. Per AOP 915, ASC may be required when conditions exist that may threaten CR habitability or when functions required to achieve or maintain cold shutdown are compromised by fire. This approach is consistent with guidance to establish VFDRs based on actions taken that are not at the Primary Control Station (PCS) (FAQ 07-0030, RG 1.205 Regulatory Position 2.4)

The fire PRA evaluated fire scenarios to determine if conditions in the CR may threaten habitability or if fire damage may result in the loss of a sufficient set of controls. If a fire resulted in one of these conditions, CR abandonment was postulated and ASC was relied upon. For these cases, a delta risk calculation was performed to evaluate the VFDRs associated with ASC, using the guidance from FAQ 07-0030 and RG 1.205. To account for potential fires that could challenge shutdown from the PCS, CCDP adjustments were made to account for the potential impact of fire damage, as discussed in the response to RAI PRA 64.

If a fire scenario did not result in one of the conditions described above, then ASC was not required and shutdown from the CR was evaluated. In these cases there is not a delta risk calculation, because a VFDR is not associated with the fire scenario. That is, ASC is not relied upon.

However, the non-abandonment cases were considered as part of the risk calculation for the fire area, and were used to gain risk insights for CR fires.

d. LAR (ML11221A280) Attachment G Step 5 describes the process of evaluating the reliability of RAs. The evaluation depends on if a recovery action is modeled specifically in the fire PRA or if a recovery action is not modeled specifically because the risk associated with the recovery action is bounded by the treatment of additional risk associated with the VFDR. The result of Step 5 was that no specific recovery actions were added to the fire PRA. As indicated in Attachment W of the LAR (ML11221A280), RAs were reviewed and evaluated but not modeled specifically.

The delta risk calculations for CR abandonment scenarios were performed by applying an appropriate CCDP that reflected the applicable VFDRs for the variant case and comparing the variant case to an appropriate CCDP that reflected the compliant case. For the variant case, the risk associated with the RAs was bounded by equipment failures; therefore, RAs were not explicitly included in the delta risk calculations for the CR abandonment scenarios.

Report Number 0027-0042-000-004 Duane Arnold Energy Center Fire Risk Evaluations, Attachment – Fire Area CB1 describes the delta risk calculations for the CR abandonment scenarios. Additionally, the response to RAI PRA 64 provides additional information related to the evaluation of RAs for CR abandonment scenarios.

DAEC PRA RAI 65

As described in LAR Attachment U, the internal events peer review was originally performed in December 2007 using the combined PRA standard, ASME/ANS RA-Sa-2005, and RG 1.200, Revision 1. The subsequent focused PRA peer review was conducted in March 2011 using the most current combined PRA standard, ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2. As stated in the Oct. 14, 2011 LAR supplement (ADAMS Accession No. ML1128702452), the scope of the 2011 peer review focused on the SR associated with upgrades, updates, or previous F&Os and not all the SRs previously assessed as 'MET' during the 2007 full scope peer review were reassessed. Provide a self-assessment of the PRA model for the RG 1.200, Revision 2 clarifications and qualifications and indicate how any identified remaining gaps were dispositioned.

RESPONSE:

The SRs within the scope of the March 2011 peer review were assessed against the most current combined PRA standard, ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2 including all related clarifications and qualifications. The remaining "Met" SRs **NOT** in the scope of the 2011 Focused Peer Review were reviewed and no new gaps were identified relative to this most current combined standard.

The GAP assessment compared the SRs in ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2 to the SRs reviewed in ASME/ANS RA-Sa-2005, and RG 1.200, Revision 1. The comparison determined if there were any differences between these two sets of SRs that were not reviewed during the focused peer review. There were none.

A copy of the self-assessment is provided.

ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2 GAP Assessment of Met SRs **NOT** within the scope of the March 2011 peer review.					
SR	**2007 Capability Assessment**	**2007 Basis for Assessment reviewed against** ASME/ANS RA-Sa-2005, and RG 1.200, Revision 1	**Rev 1 vs Rev 2**	**GAP**	**2009 SR**
IE-A1	Met	Primarily used existing list of known initiators and plant specific initiators.	SAME	None	IE-A1
IE-A3	Met	Challenges account for plant experience. However, see A2 for suggestion to evaluate loss of AC bus and CRD as possible initiators.	SAME	None	IE-A3
IE-A3a	Met: (CC I/II)	Generic analysis of similar plants is considered.	SAME	None	IE-A4
IE-A5	Met	The DAEC PRA incorporates events that have occurred at conditions other than at-power, including events generating a scram during controlled shutdown conditions. Evidence of that can be found in Table 2-1 and Appendix A3 of the Initiating Event Analysis notebook.	SAME	None	IE-A7
IE-B2	Met	The Duane Arnold categories were based a review of categories used in 4 existing PRAs, categories from EPRI study on plant transients, PSA procedures guide, Wash 1400 and the UFSAR and a review of Duane Arnold specific initiators. The categories appear reasonable. A further enhancement would be to better define the "general transient" category. See IE-B1 SR for related F&Os.	SAME	None	IE-B2
IE-B4	Met	Initiating events were found to be properly grouped.	SAME	None	IE-B4
IE-C1	Met	Document 1249309D-002 "Initiating Event Analysis" documents use of both generic and plant specific data in generating Initiating Event Frequencies for DAEC PRA model throughout document.	SAME	None	IE-C1
IE-C2	Met	Document 1249309D-002 "Initiating Event Analysis" shows use of Bayesian updating in Table 2.3, however the updating is using a non-informative prior. The use of an informative prior is NA. This SR is considered Met.	SAME	None	IE-C4
IE-C5	Met: (CC I/II)	No evidence found for use of time trend analysis in initiating events. This SR not applicable for Category II	SAME	None	IE-C7
IE-C11	Met: (CC I/II)	1249309D-002 "Initiating Event Analysis" Section 3.3 for RPV provides the comparison of other data sources as justification for choice of RPV frequency. SR Met for Category I and II.	SAME	None	IE-C13
IE-C12	Met: (CC I/II)	ISLOCA found to be adequately analyzed.	SAME	None	IE-C14
IE-D1	Met	Review of initiating events documentation.	SAME	None	IE-D1
AS-A1	Met	The accident sequence methodology explicitly models systems and operator actions, employs event tree analysis, and uses acceptable means (PRAQuant) for quantification	SAME	None	AS-A1
AS-A2	Met	Event tree notebook describes key safety functions associated with each modeled initiating event	SAME	None	AS-A2
AS-A3	Met	Systems required to meet key safety functions found for those initiating events modeled.	SAME	None	AS-A3
AS-A4	Met	The necessary operator actions to achieve the defined success criteria are discussed as appropriate.	SAME	None	AS-A4
AS-A5	Met	The accident sequence model appears to be consistent with the plant-specific: system design, EOPs, abnormal procedures, and plant transient response.	SAME	None	AS-A5
AS-A6	Met	The DAEC PRA Event Tree notebook includes detailed accident scenario descriptions, including timing of events. The event-tree models reflect the accident scenario descriptions.	SAME	None	AS-A6
AS-A7	Met: (CC I/II)	Possible accident sequences for each initiating event are delineated to a high level of detail in the DAEC PRA. A CC III could not be assigned to this SR because the Excessive LOCA initiating event and resulting accident sequences discussed in Section 11 of the notebook are not explicitly incorporated in the DAEC PRA Quantification.	SAME	None	AS-A7
AS-A8	Met	End states are documented in the DAEC PRA Event Tree notebook. Tables 2-4 and 2-5 of the notebook summarize accident scenario end states / core damage classes.	SAME	None	AS-A8
AS-A9	Met: (CC II)	The DAEC PRA uses plant-specific and generic thermal hydraulic analyses to determine accident progression parameters.	SAME	None	AS-A9
AS-A10	Met: (CC II)	The DAEC PRA includes in the accident sequences modeled by the event trees sufficient detail that significant differences in requirements on systems and operator responses are captured.	SAME	None	AS-A10
AS-A11	Met	Transfers between event trees are properly documented in the DAEC PRA Event Tree notebook.	SAME	None	AS-A11

ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2 GAP Assessment of Met SRs **NOT** within the scope of the March 2011 peer review.					
SR	**2007 Capability Assessment**	**2007 Basis for Assessment reviewed against** ASME/ANS RA-Sa-2005, and RG 1.200, Revision 1	**Rev 1 vs Rev 2**	**GAP**	**2009 SR**
AS-B1	Met	The General Transient Event Tree Notebook section was reviewed in detail. The success criteria was noted to be 2 SRVs open (ET Notebook page 3-9). The Success Criteria notebook page 4-6 identified the success criteria as 3 SRVs open and noted no plant specific analysis was performed. A pressure relief function failure is not processed as a LLOCA for non-ATWS sequences. Justification is provided on page 3-10 of the Event Tree notebook. This is acceptable for Category II, but not Category III. Loss of Instrument Air was found to cause loss of MC and Condensate Make-up to Feedwater. This dependency is modeled in the system models.	SAME	None	AS-B1
AS-B2	Met	Review of the LOOP with Stuck Open Relieve Valve Event Tree found the following: RCIC and HPCI alone are not adequate for injection. This is correct. The PRA Model assumes SRVs are not challenged with HPCI and RCIC available. Alternate injection systems are not credited without recovery of offsite power in two hours. Depressurization is needed for low pressure injection. The event tree review indicates dependencies are properly modeled.	SAME	None	AS-B2
AS-B5	Met	Reviewed Loss of Off-Site Power and Large LOCA outside Containment Event Trees. No issues identified.	SAME	None	AS-B5
AS-B6	Met	Reviewed SBO/LOOP and ATWS Event Tree notebooks. No issues found. Excellent use of Calculations to support SBO timing assumptions (Strength). Event Tree Notebooks provide excellent description of the Event Trees and event tree timing assumptions (Strength).	SAME	None	AS-B7
AS-C1	Met	Event trees are adequately developed.	SAME	None	AS-C1
SC-A1	Met	Core damage would not occur until after the core was more than two-thirds uncovered and the water level was not being recovered. This meets the intent of the standard	SAME	None	SC-A1
SC-A2	Met: (CC II/III)	core damage would not occur until after the core was more than two-thirds uncovered and the water level was not being recovered. This meets the intent of the standard	SAME	None	SC-A2
SC-A4	Met	Success criteria found for those initiating events modeled.	SAME	None	SC-A3
SC-B1	Met: (CC II)	Document 1249309D-003 "Level 1 PSA Success Criteria" References several GE documents that are applicable to DAEC and are used as realistic generic sources for success criteria.	SAME	None	SC-B1
SC-B3	Met	Document 1249309D-003 "Level 1 PSA Success Criteria" and 1249309D-006 "LEVEL 1 PSA MAAP THERMAL HYDRAULIC CALCULATIONS" provide detail of analysis being performed and references to different evaluations and MAAP runs to support each success criteria.	SAME	None	SC-B3
SC-B5	Met	Document 1249309D-003 "Level 1 PSA Success Criteria" Section 4.1 Timely Reactor Depressurization discusses the results of the MAAP runs and compares the results to same analysis at other plants and additional analyses from NEDO and EPRI reports.	SAME	None	SC-B5
SC-C2	Met	Reviewed Success Criteria Notebook. CD definition is provided in Section 2.1 of SC Notebook with basis for the definition. Sources for the SC are referenced in notes related to the SC for each event tree (e.g., Table 3-1a of SC notebook).	SAME	None	SC-C2
SY-A6	Met	system boundaries defined.	SAME	None	SY-A6
SY-A7	Met: (CC I/II)	However, no justification for use of point value but no system development for initiating event ., even with equipment and human dependencies	SAME	None	SY-A7
SY-A8	Met	Documentation found in PTG-006, Step 2	SAME	None	SY-A8
SY-A11	Met	Noted evidence of different success criteria (SRVs needed for ATWS versus transients) are required for some systems to mitigate different accident scenarios	SAME	None	SY-A10
SY-A12	Met		SAME	None	SY-A11
SY-A12a	Met		SAME	None	SY-A12
SY-A12b	Met	Flow diversion in system notebooks based on 1/5 diameter which is conservative. Suggest less restrictive requirement, e.g., 1/3.	SAME	None	SY-A13
SY-A13	Met	Failure modes consistent with the level of modeling detail have been considered. PTG has it	SAME	None	SY-A14
SY-A15	Met: (CC I/II)	Some pre-initiators have been considered. Refer to HRA element for disposition of pre-initiators. Cat II	SAME	None	SY-A16
SY-A16	Met	Refer to HRA element for disposition of post-initiators. Cat II	SAME	None	SY-A17
SY-A17	Met	Component trips have been described in the notebooks (e.g. RHR); PTG-6 addresses this.	SAME	None	SY-A18
SY-A19	Met	Excessive heat loads are evaluated in Report.	SAME	None	SY-A21

ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2 GAP Assessment of Met SRs **NOT** within the scope of the March 2011 peer review.					
SR	**2007 Capability Assessment**	**2007 Basis for Assessment reviewed against** ASME/ANS RA-Sa-2005, and RG 1.200, Revision 1	**Rev 1 vs Rev 2**	**GAP**	**2009 SR**
SY-A21	Met	A sampling of basic events shows consistency in naming nomenclature.	SAME	None	SY-A23
SY-A22	Met	Repair methods are not used. The DAEC HRA, Supplement A lists an operator action "Operator fails to repair hardware in approx. 15 hours".	SAME	None	SY-A24
SY-B2	Met: (CC I/II)	No requirement to include intersystem common cause modeling	SAME	None	SY-B2
SY-B4	Met	The Beta analysis approach is applied. The data and the models are consistent.	SAME	None	SY-B4
SY-B5	Met	System supports are accounted for through the linked fault tree approach.	SAME	None	SY-B5
SY-B6	Met	System Dependency Notebook, HPCI System Notebook The HPCI/RCIC room heatup calculation exemplified compliance with this SR.	SAME	None	SY-B6
SY-B7	Met: (CC II)	Support system modeling is consistent with the modeling of the front line systems in this linked fault tree model.	SAME	None	SY-B7
SY-B10	Met	Review of SLC System Notebook and 125 VDC, both have dependency matrix that identifies these interfaces. See Table 3-1. This SR is Met	SAME	None	SY-B9
SY-B11	Met: (CC II/III)	RHR Notebook describes low pressure permissive require for LPCI initiation. Logic found in model. This SR is Met Category II	SAME	None	SY-B10
SY-B12	Met	Examples found of modeling inventories of air, power, and cooling to support mission time. This SR is met.	SAME	None	SY-B11
SY-B13	Met	ROOM COOLING TO NON-ESSENTIAL SWITCHGEAR modeled in AC System Notebook. This SR is Met	SAME	None	SY-B12
SY-B14	Met	Instrument Air System fails many components, confirmed under the single top model. This SR is met.	SAME	None	SY-B13
SY-B16	Met	SLC system requires action to initiate. Operator actions required to open doors to HPCI and RCIC room to prevent overheating. SR is MET.	SAME	None	SY-B15
SY-C1	Met	The DAEC PRA system notebooks provide sufficient documentation to facilitate PRA applications, upgrades, and peer review.	SAME	None	SY-C1
DA-A1	Met	Test, maintenance and recovery actions have been identified. Common cause grouping considers Beta level modeling.	SAME	None	DA-A1
DA-A3	Met	The parameters and associated failure data have been identified.	SAME	None	DA-A4
DA-B2	Met: (CC I/II)	No evidence was found in the DAEC PRA of outliers being included in a group of components of the same type.	SAME	None	DA-B2
DA-C2	Met	Reviewed Component Data Notebook. No discussion of plant specific data Plant specific data is used but the use is not incorporated into the Data Notebook. Reviewed PTG-007: Provides instructions for collecting and analyzing data.	SAME	None	DA-C2
DA-C4	Met	Reviewed Component Data Notebook. Reviewed DAEC Rev 5 PRA Component Failure Update Letter No NG-02-0869.pdf and DAEC Rev 5 PRA Component Failure Update Letter No NG-03-0142.pdf Maintenance rule data is used for unavailability data.	SAME	None	DA-C4
DA-C5	Met	Maintenance rule data used.	SAME	None	DA-C5
DA-C6	Met	EPIX was used as the source, which is plant specific.	SAME	None	DA-C6
DA-C7	Met: (CC II/III)	EPIX information used, which is plant specific.	SAME	None	DA-C7
DA-C8	Met: (CC II/III)	EPIX information used, which is plant specific.	SAME	None	DA-C8
DA-C9	Met: (CC I/II)	EPIX information used which is plant specific.	SAME	None	DA-C9
DA-C11	Met	Maintenance rule data is used.	SAME	None	DA-C11

ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2 GAP Assessment of Met SRs NOT within the scope of the March 2011 peer review.					
SR	2007 Capability Assessment	2007 Basis for Assessment reviewed against ASME/ANS RA-Sa-2005, and RG 1.200, Revision 1	Rev 1 vs Rev 2	GAP	2009 SR
DA-C11a	Met	Maintenance rule data is used which assures that cascading of support system unavailability to the front line system is not done.	SAME	None	DA-C12
DA-D1	Met: (CC II)	The DAEC model is heavily based on generic data. In later revisions (5 for example) plant specific values are calculated. This SR is MET	SAME	None	DA-D1
DA-D2	Met	Section H of the Component Data Notebook provides a fault tree by fault tree discussion of Data Based on Engineering Judgment or Review of Operating Experience with notes documenting the rational for use of expert judgment. Ties SR is MET	SAME	None	DA-D2
DA-E1	Met	A reasonable amount of information was provided to the review team to understand most aspects of the Data Analysis element.	SAME	None	DA-E1
IF-A1	Met	The Internal Flooding Analysis properly defines flood areas.	SAME	None	IFPP-A1
IF-A1a	Met	Internal Flooding Analysis properly defines flood areas.	SAME	None	IFPP-A2
IF-A3	Met	The development of flood areas included use of the UFSAR Section 2.4.2 "Floods" and UFSAR Section 3.4 "Water Level (Flood) Design" as well as UFSAR Sections 3.6 and 1.4. Also DAEC response to SOER 85-5, the DAEC HELB analysis and the DAEC Fire Hazards Analysis as well as studies from other plants. Plant walkdowns confirmed assumptions used in the development of flood areas. Plant General arrangement drawings were also used (Reference 4 of Section 7 lists drawings utilized).	SAME	None	IFPP-A4
IF-B1	Met	Page 1-7 notes systems, tanks, and potential flow paths are identified. The Internal Flooding assessment explicitly addressed the potential for back flow through drains. Potential back flow through drains was also addressed in the DAEC Response to SOER 85-05 (Attachment B of the Internal Flooding Analysis). The DAEC Response credited check valves and isolation valves in several areas. Flow paths through Ventilation ducts were explicitly addressed.	SAME	None	IFSO-A1
IF-B1b	Met	Areas with no significant flood sources have been screened out as documented in Section 3 of the Flooding Analysis.	SAME	None	IFSO-A3
IF-C1	Met	Propagation paths were identified in Section 3 of the Internal Flooding Analysis.	SAME	None	IFSN-A1
IF-C2	Met	Reviewed Internal Flooding Notebook: The IF Nbk includes the information requested in the SR. These features are discussed in the document text.	SAME	None	IFSN-A2
IF-C2a	Met	Reviewed Internal Flooding Notebook: The IF Nbk includes the information requested in the SR. These features are discussed in the appendices of the notebook	SAME	None	IFSN-A3
IF-C2b	Met	Reviewed Internal Flooding Notebook: The IF Nbk includes consideration of sumps/drains and considers the impacts of the sumps/drains on the scenarios.	SAME	None	IFSN-A4
IF-C3b	Met: (CC II)	Propagation is considered. Potential for door failures is considered.	SAME	None	IFSN-A8
IF-C3c	Met	Reviewed the IF Notebook An example of a calculation is provided on page A-3 of the calculation of flood timing to terminate the flood in the reactor building.	SAME	None	IFSN-A9
IF-C4	Met	Flood scenarios are discussed in the notebook for those events that are not screened out.	SAME	None	IFSN-A10
IF-C6	Met: (CC III)	Operator actions to prevent challenges to normal plant operations are not relied upon to screen out flood areas.	SAME	None	IFSN-A14
IF-D1	Met	New plant flooding initiators have been created for all the screened-in flooding scenarios.	SAME	None	IFEV-A1
IF-D3	Met: (CC II)	Flooding scenarios have been grouped only within similar plant response areas. Example: Reactor building flooding initiator includes HPCI, RCIC, corner rooms and torus area with worst case/similar plant response.	SAME	None	IFEV-A2
IF-D3a	Met: (CC III)	Flooding initiators are independently modeled and are not subsumed with other plant initiating groups	SAME	None	IFEV-A3
IF-D6	Met: (CC I/II)	Generic data has been used which accounts for maintenance related activities.	SAME	None	IFEV-A7
IF-E1	Met	Flooding scenarios are adequately represented by applicable accident sequence model.	SAME	None	IFQU-A1
IF-E6	Met	Quantified as per existing methodology	SAME	None	IFQU-A7
IF-E6a	Met	SR met.	SAME	None	IFQU-A8

ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2 GAP Assessment of Met SRs **NOT** within the scope of the March 2011 peer review.					
SR	**2007 Capability Assessment**	**2007 Basis for Assessment reviewed against** ASME/ANS RA-Sa-2005, and RG 1.200, Revision 1	**Rev 1 vs Rev 2**	**GAP**	**2009 SR**
IF-E7	Met	LERF is adequately treated in the Level 2 assessment.	SAME	None	IFQU-A10
QU-A1	Met	Evidence found in documentation that shows model integrates system analysis, initiating events, data, HRAs, and dependencies.	SAME	None	QU-A1
QU-A2a	Met	The accident sequence logic, top logic, frontline system, and support system fault tree models are linked together within CAFTA to form an integrated fault tree model. The fault tree contains all the necessary logic required to quantify the accident sequences defined by the event trees. The Level 1 CDF logic is linked into the Level 2 analysis. PRAQuant provides the platform for specifying: - Fault tree file and basic event database to be used - Individual accident sequences to be quantified - Quantification truncation limit to be applied to each sequence (different values may be applied to different sequences) - Flag files to be applied to each sequence quantification - Mutually exclusive file to be used in the quantification process. - Recovery files to be used in the quantification process.	SAME	None	QU-A2
QU-A2b	Met: (CC II)	Model uses a type code file which ensures that any event probabilities correlations performed (like Monte Carlo) would impact like basic events the same (i.e. all the like MOV's).	SAME	None	QU-A3
QU-A3	Met	The Level 1 CDF logic is linked into the Level 2 analysis. Accident sequences can be tabulated as shown by Table 3-2 of DAEC PRA Revision 5C Summary Document. Additionally cutset files can be produced to identify contributors to CDF. This SR is Met.	SAME	None	QU-A4
QU-A4	Met	Recovery action to be used in the quantification process and recoveries are found in the model fault tree. This SR is Met.	SAME	None	QU-A5
QU-B1	Met	Use industry standard software tools CAFTA, PRAQUANT, etc.	SAME	None	QU-B1
QU-B2	Met	Truncation is being performed at 1E-11, the CDF is approximately at 1E-5, which is 1E-6 below CDF. This is a sufficient truncation level per the convergence analysis This SR is met.	SAME	None	QU-B2
QU-B3	Met	Figure 3-13 of DAEC PRA Revision 5B Summary Document shows CDF vs. truncation limit and resulting in selected truncation level. This SR is Met.	SAME	None	QU-B3
QU-B4	Met	The "minimized" CDF and LERF results are used in risk applications via the software used CAFTA. SR is MET	SAME	None	QU-B4
QU-B5	Met	Software being used automatically identifies circular logic and no quantification can proceed without resolving the circular logic. Common circular logic traps (AC Power Diesels and Battery Chargers) have been appropriately resolved from inspection of model. There is no documentation of resolution of circular logic, it is considered part of model development. This SR is Met	SAME	None	QU-B5
QU-B6	Met	System Success accounted for in evaluation of accidents sequences. ONEFORALL software performs this automatically. SR is Met.	SAME	None	QU-B6
QU-B7a	Met	Mutually Exclusive events for maintenance activities has been identified and implemented in the model. This SR is Met.	SAME	None	QU-B7
QU-B7b	Met	Mutually Exclusive events for maintenance activities has been identified and implemented in the model via a list input to the software. Software removes cutset post process. The SR is Met.	SAME	None	QU-B8
QU-C3	Met	Event Tree Linking was discussed with PRA team and found SAT. SR is Met.	SAME	None	QU-C3
QU-D1b	Met	The results of the DAEC PRA show modeling and operational consistency.	SAME	None	QU-D2
QU-D1c	Met	The DAEC PRA quantification model does not include flags (eliminated in Rev 5C), and recoveries are explicitly modeled in the event trees. There is a mutually exclusive events file including maintenance events. The quantification gives logical results.	SAME	None	QU-D3
QU-D4	Met	A review of nonsignificant cutsets resulting from the DAEC PRA quantification indicated that the logic of the cutsets is correct. The samples of cutsets analyzed were in the 6.0E-10 and 6.0E-11 range. These samples were obtained from the DAEC PRA Level 1 quantification cutset file.	SAME	None	QU-D5
QU-D5b	Met	The importance of components and basic events resulting from the model quantification make logical sense.	SAME	None	QU-D7
QU-F2	Met	Quantification in the form of the Model Summary Documents provides most of the listed items this SR is Met.	SAME	None	QU-F2

ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2 GAP Assessment of Met SRs **NOT** within the scope of the March 2011 peer review.					
SR	**2007 Capability Assessment**	**2007 Basis for Assessment reviewed against** ASME/ANS RA-Sa-2005, and RG 1.200, Revision 1	**Rev 1 vs Rev 2**	**GAP**	**2009 SR**
LE-A1	Met	The DAEC LE Analysis contains several discussions of accident types, plant features, characteristics. and how they relate to release timings and magnitudes, including LERF. The CETs model the following physical characteristics at the time of Core Damage that could influence LERF: -Containment Isolation -Core Melt Arrested In-Vessel -Energetic Phenomena Post-Core Melt -Steel Containment Shell Failure -Containment Flooding -Containment Heat Removal -Containment Overpressurization (or Overtemperature) Failure -Suppression Pool Bypass -Release Mitigation in Reactor Building	SAME	None	LE-A1
LE-A2	Met	Section 8 of the Level 1 DAEC PRA and Section 5 of the Level 2 DAEC discuss accident sequence characteristics that impact LERF in table 5-1 SUMMARY OF THE CORE DAMAGE FREQUENCY BY ACCIDENT SEQUENCE SUBCLASS.	SAME	None	LE-A2
LE-A3	Met	The characteristics of the Level 1 event trees are transferred to the Level II event trees as logic (both failure and success criteria) and Level II dependencies on Level I are also considered. This SR is Met	SAME	None	LE-A3
LE-A4	Met	The characteristics of the Level 1 event trees are transferred to the Level II event trees as logic (both failure and success criteria) via a bridge tree built by hand and Level II dependencies on Level I are also considered. This SR is Met.	SAME	None	LE-A4
LE-A5	Met	Plant Damage States defined are consistent with the methods used in LE-A1 through A4 SRs.	SAME	None	LE-A5
LE-B1	Met: (CC II)	Noted evidence of and unique plant features, e.g., RHRSW cross-tie was used as the alternate injection source.	SAME	None	LE-B1
LE-B2	Met: (CC II)	Apparent use of combination of conservative and realistic assumptions used for non significant containment challenge	SAME	None	LE-B2
LE-B3	Met	Evidence found that model utilized supporting engineering analyses in accordance with the applicable requirements.	SAME	None	LE-B3
LE-C1	Met: (CC II)	Category II requirements appear to be satisfied.	SAME	None	LE-C1
LE-C2b	Met: (CC II/III)	No indication that repair is considered.	SAME	None	LE-C3
LE-C3	Met: (CC II)	Category II requirements appear to be satisfied.	SAME	None	LE-C4
LE-C4	Met: (CC II)	Level 1 systems and similar L2 models are used in the accident sequences.	SAME	None	LE-C5
LE-C5	Met	Level 1 systems and similar L2 models are used in the accident sequences.	SAME	None	LE-C6
LE-C6	Met	HRA approach is consistent across Level 1 and Level 2.	SAME	None	LE-C7
LE-C7	Met	Linked fault tree approach accounts for system dependencies consistently across Level 1 and Level 2.	SAME	None	LE-C8
LE-C8a	Met: (CC II/III)	Examined Appendix M of the L2 Notebook.	SAME	None	LE-C9
LE-C9a	Met: (CC II/III)	Very little credit is taken for success of inventory makeup after containment failure based upon the number of systems that are available.	SAME	None	LE-C11
LE-C9b	Met: (CC II)	Level 2 documentation adequately supports CCII.	SAME	None	LE-C12
LE-C10	Met: (CC II/III)	Level 2 documentation discusses analyses associated with the development of decontamination factors associated with scrubbing.	SAME	None	LE-C13

ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2 GAP Assessment of Met SRs **NOT** within the scope of the March 2011 peer review.					
SR	**2007 Capability Assessment**	**2007 Basis for Assessment reviewed against** ASME/ANS RA-Sa-2005, and RG 1.200, Revision 1	**Rev 1 vs Rev 2**	**GAP**	**2009 SR**
LE-D1a	Met: (CC II)	Table 3.3-2 tabulates postulated containment challenges for which the containment has been analyzed. Containment analysis is performed using MAAP. DAEC MAAP model includes several plant specific features such as incorporating HTLC, EOPs for venting, pool cooling, drywell sprays, failure size, etc. Generic information is also used in the analysis.	SAME	None	LE-D1
LE-D1b	Met: (CC II)	The containment ultimate capability analysis considers seals, penetrations, hatches, bellows, etc. (Reference Section 3.5 of notebook).	SAME	None	LE-D2
LE-D2	Met: (CC II)	The location of the failure is determined probabilistically based on the plant specific structural analysis for slow overpressurization. The containment failure size and location are used to calculate the release to the reactor building and ultimately to the atmosphere.	SAME	None	LE-D3
LE-D3	Met: (CC II)	The postulated failure probability is based on SNPS PRA. Interfacing system failure analysis is modeled as CET3: Class V, where containment is bypassed and a direct release path is established from the RPV to the reactor building. The Class V CET is used to evaluate two distinct core melt scenarios. LOCAs outside containment for which coolant makeup to the reactor vessel has failed leads to a core melt event with a direct release pathway from the vessel to the reactor building, and an interfacing LOCA or drywell bypass	SAME	None	LE-D4
LE-D6	Met: (CC II)	MAAP run is used to perform containment isolation analysis. The success of the containment isolation node (IS) is satisfied If the containment penetrations that communicate between the drywell (or wetwell) atmosphere and the reactor building (or environment) are "closed and isolated". The criteria used to satisfy this requirement of "closed or isolated" is that no line, hatch, or penetration has an opening greater than 2 inches in diameter. This implies that all containment penetrations are adequately sealed and isolated during the entire accident progression until either: (1) a safe stable state is reached; or, (2) the accident conditions exceed the ultimate capability of containment as determined in the plant specific evaluation.	SAME	None	LE-D7
LE-E1	Met	The DAEC PRA Level 2 model uses parameter values consistent with Level 1 human reliability and data analyses.	SAME	None	LE-E1
LE-E2	Met: (CC II)	The DAEC PRA Level 2 model uses realistic parameter estimates for significant accident sequences, and conservative estimates for non-significant accident sequences, when plant-specific calculations were not performed.	SAME	None	LE-E2
LE-E3	Met: (CC II)	SR met.	SAME	None	LE-E3
LE-F1a	Met: (CC II/III)	5B summary has breakdown of LERF by initiator, has operator actions by importance, etc but not by accident classes nor containment failure modes (LERF contributors as shown on Table 4.5.9-3). However, the level 2 quantification file can generate LERF per accident subclass, e.g., class IA, II, etc	SAME	None	LE-F1
LE-F1b	Met	Section 3.3 contains potential containment failure modes	SAME	None	LE-F2
LE-F2	Met: (CC II)	Extensive sensitivity analyses contained in Section 4.9 of the notebook	Deleted	None	n/a
LE-G1	Met	Documentation provides adequate detail for review and use by applications.	SAME	None	LE-G1
LE-G2	Met	Review of LE documentation provides for listed examples and more. SR is MET	SAME	None	LE-G2
LE-G3	Met: (CC II/III)	The Level 2 Model Summary document provides various lists of significant contributors to LERF and their individual contribution. This SR Meets Category II	SAME	None	LE-G3
LE-G4	Met	Model assumptions are discussed in the Level 2 Notebook section 3.6.1 and uncertainty is addressed in Appendix K. Discussion of sources of uncertainty was only general in nature.	Similar	None	LE-G4

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Identify any changes made to the FPRA since the full-scope peer review that are consistent with the definition of a "PRA upgrade" in ASME/ANS-RA-Sa-2009, "Addenda to ASME/ANS RA-S 2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, American Society of Mechanical Engineers/American Nuclear Society, New York, NY," as endorsed by Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Revision 2, March 2009." Specifically consider if the changes described in the LAR Table V-3 disposition of F&Os 4-22, 4-23, 4-25 and 4-32 are upgrades. If any such changes exist, describe what actions have been or will be implemented to address this review deficiency (i.e., lack of a focus-scope peer review when an upgrade occurs).

RESPONSE:

Appendix 1-A of the ASME/ANS RA-Sa-2009 standard provides guidance in determining when a change to the PRA satisfies the definition of PRA upgrade. Per the standard, the definition of PRA upgrade satisfies when one of three criteria:

1. New methodology
2. Change in scope that impacts the significant accident sequences or the significant accident progressions
3. Change in capability that impacts the significant accident sequences or the significant accident progressions

Changes in the Fire PRA since the full scope Fire PRA peer review are those changes that were performed to the internal events PRA model which the Fire PRA is built and those related to resolution of Fire PRA peer review F&Os.

A focused scope peer review of the internal events PRA model was conducted in March 2011 and is discussed in Attachment U of the LAR. No additional changes were made to the internal events PRA model after the focused scope peer review.

Changes in the Fire PRA to resolve peer review F&Os do not satisfy any of the criteria above. The methods employed in the fire PRA are consistent with those reviewed and recommended during the peer review. Additionally, the changes did not result in a change in the scope or capability of the Fire PRA. The changes did not result in significant changes to the risk insights from the Fire PRA.

F&Os 4-22, 4-23, and 4-25 relate to the requirement to characterize the factors that influence the timing and damage associated with ignition sources and target sets. The F&Os identified limitations in the method applied. The resolution of these F&Os was to apply the method used in the sensitivity study referenced in F&O 4-22. The sensitivity

study method included multi point treatment, fire growth, and fire severity such that the concerns raised by these F&Os could be addressed.

F&O 4-32 relates to concerns of the application of a severity factor for transient fire scenarios. As recommended by the F&O, the severity factor was removed and not replaced with another.

Section 1-A.3 of the ASME/ANS RA-Sa-2009 standard provides examples of PRA change classifications. A review of the change, classification, and rationale of these examples indicate that the changes to the Fire PRA since the full scope peer review do not satisfy the definition of PRA upgrade. Specifically, Example 32 discusses that use of multiple methods that have been peer reviewed is not classified as a PRA upgrade given that a new method was not being incorporated. As discussed above, F&Os 4-22, 4-23, and 4-25 were resolved by changing the method related to the F&O to an alternate method that was peer reviewed which resolved the F&O concerns. In resolution to the concern identified in F&O 4-32, the method was removed and not replaced by another.

DAEC RAI PRA 68

The disposition of the peer review F&O 2-8, states that common cause failure (CCF) for fire induced failures do not impact the results. How is the CCF probability in the FPRA treated when redundancy is reduced as a result of a fire (e.g., redundancy is decreased from N to N-1)?

RESPONSE:

Common cause treatment of remaining redundant components is needed only for component groups of more than two. That is, if fire damage occurs to one member of a two member group, then failure of the other member is sufficiently reflected in the independent failure rate for that component. For component groups larger than two, the PRA includes common cause treatment for each subset of a component group (e.g., for a group of three, a subset is included for two of three CCF and three of three CCF). Therefore, when fire damages one or more components in a component group, the reduced redundancy is accounted for with the subset CCF combinations in addition to the independent failure rates.

DAEC RAI PRA 70

FSR Section 5.1.5.1 states that for closed panels that are substantially sealed, damage is limited to the cabinet itself. Fire propagation from electrical cabinets is discussed extensively in FAQ 08-0042 (ADAMS Accession No. ML092110537). Summarize the guidelines for substantially sealed in FAQ 08-0042 are met.

RESPONSE:

As stated in Section 5.1.5.1 of the Fire Scenario Report, 0493080001.003, fire propagation from panels was not considered if the panel was substantially sealed. Table B-1 of the Overview and Documentation Roadmap report (0493080001.000) dispositions the guidance in FAQ 08-0042 as incorporated in the PRA (note that Table B-1 mistakenly references FAQ 07-0042 instead of FAQ 08-0042). Consistent with the guidance, fire propagation from panels was not considered if:

1. Based on visual inspection, the panel did not contain open vents or open penetrations such that the passage of air would not be readily allowed.

2. Based on visual inspection, the panel doors were secured by multiple mechanical fasteners that appeared not to be simple twist handle style door latches.

DAEC RAI PRA 71

Appendix A of the FSR for the CR transient fire appears to include a 0.1 conditional probability that combustibles are stored near the specific location in the MCR. Provide justification for the use of this factor.

RESPONSE:

The transient fire ignition frequency for the Main Control Room is derived from assigned transient influence factors of Very High, High, and High for Mech/Elect Maintenance, Occupancy, and Storage, respectively (see Table B-1 of report 0493080001.001 entry for PAU 12A). These influence factors may be appropriate for specific locations in the Main Control Room based on NUREG/CR-6850 guidance; however, they are not necessarily reflective of everywhere in the Main Control Room.

Transient fires in the Main Control Room were postulated near cable trays with fire PRA targets consistent with NUREG/CR-6850 Section 11.5.1.6. These transient fires are postulated in the electric panel area behind the Main Control Board. This area is different than the locations in front of the Main Control Board which includes the operators area, Shift Supervisor Office, General Office, Tagout Area, computer room, and kitchen. Specifically, the electric panel area contains cable trays.

The Control of Combustibles procedure, ACP 1412.2 provides guidance for the placement of combustible materials. Section 3.1 provides caution that combustibles should be located away from ignition sources and away from plant equipment and cable trays. The caution provides a rule of thumb of six feet from plant equipment and twelve feet from overhead cable trays.

The transient fire ignition frequency is apportioned to the postulated transient fires in the electric panel area based on the applicable transient area. However, the frequency was not considered to be equal to other areas of the Main Control Room. Because of the cable trays in the electric panel area, it was postulated that combustible materials would be less likely to be stored in these locations versus locations in the Main Control Room where cable trays are not located. Therefore, a 0.1 conditional probability was applied given these considerations.

DAEC RAI PRA 73

It is noted that the sensitivity studies described in response to PRA RAI 1 (ADAMS Accession No. ML12146A094) as well as the response to some of the other RAIs, for example: PRA RAIs: 8, 14, 20 and 35 (ADAMS Accession No. ML12146A094) may lead to the need for revised and updated FPRA and NFPA 805 LAR documentation. Describe the plans for developing the documentation for the updated models that meets the requirements of SR FQ-F1.

RESPONSE:

Update of the DAEC Fire PRA performed as a result of responding to the first and second requests for additional information is documented in 049308001.006, Rev. 1, "NFPA 805 RAI Model Update Quantification Report," dated October 2012. Tables are included in the report that update results presented in Attachment W of the NFPA 805 LAR. Revisions were also made to the Plant Partitioning and Fire Ignition Frequency Report (049308001.001), to the Fire Model Development Report (049308001.002), and to the Fire Scenario Report (049308001.003) as a result of responding to RAIs involving requantification of the Fire PRA.

Disposition of findings from the 2010 DAEC Fire PRA Peer Review is provided in Table V-3 of the LAR (ML11221A280). Findings related to SR FQ-F1 have been addressed and SR FQ-F1 is assessed as 'met' by DAEC's final capability assessment (Table V-1). Document updates performed to address questions from the first and second requests for additional information ensure that SR FQ-F1 is still met.

DAEC RAI PRA 74

Relative to F&O 5-26, concerning the assignment of transient influence factors, the guidance provided in Table 4.3-1 of the "Plant Partitioning and Fire Ignition Frequency Development Report, 493080001.01" appears, in some instances to differ from the guidance in Table 6-3 of NUREG/CR-6850. For example, in Table 6-3 the maintenance influence factor should be based on the number of preventive maintenance/corrective maintenance (PM/CM) work orders compared to the average number of work orders for a typical compartment. The maintenance factors were assigned based on the frequency with which Mechanical/Electrical or Hotwork is performed such as occasionally (quarterly), or frequently (weekly). Further, Table 6-3 requires that to have a medium storage influence factor, all combustible/flammable material is stored in closed containers placed in dedicated fire safe cabinets and if not in a fire safe cabinet it is considered to have a high influence factor. Medium storage influence factor Table 4.3-1 appears to require storage normally in sealed drums/cabinets. Provide further discussion of the procedure for assigning influence factors and how it is consistent with NUREG/CR-6850.

RESPONSE:

NUREG/CR-6850 Section 6.5.7.2 provides discussion on the assignment of transient influence factors. Table 6-3 provides additional description of the transient influence factors. In implementing the guidance provided in the discussion and table, judgment is needed to best reflect the relative transient characteristics of each fire PRA compartment.

A panel of plant personnel was assembled to review each fire PRA compartment and assign a rating for transient activity based on the guidance of NUREG/CR-6850. Section 4.3 and Appendix B of the Plant Partitioning and Fire Ignition Frequency Report, 0493080001.001, document the process used to assign transient influence factors.

Bullet 1 of Section 6.5.7.2 of NUREG/CR-6850 provides guidance when assigning the maintenance transient influence factor. The NUREG/CR-6850 guidance states, "...the analyst may ask the maintenance personnel to rate assign a rating number between 0 and 10 in terms of frequency of maintenance at a compartment..." This guidance was applied when assigning the maintenance transient influence factors. To aid the plant personnel in assigning a rating factor, the frequency of maintenance activities was presented in the terms of typical plant maintenance activities (i.e., weekly, quarterly, etc.) as described in Table 4.3-1 and Table B-1 of the Plant Partitioning and Fire Ignition Frequency Report. Therefore, the method used in assigning maintenance influence factors based on frequency is consistent with NUREG/CR-6850.

Bullet 3 of Section 6.5.7.2 of NUREG/CR-6850 provides guidance when assigning the storage transient influence factor. The NUREG/CR-6850 guidance states, "The amount, type, and frequency of the use of material maintained in these storage containers should be taken into account." Table 6-3 provides a description that classifies a "Medium" influence factor as one that can be assigned only to

compartments with dedicated fire safe cabinets. The descriptions in Table 6-3 do not seem to consider amount or frequency across each of the influence factor ratings. Therefore, the discussion in Section 6.5.7.2 was considered and a "Medium" factor was assigned to compartments in which storage may be described as "some storage, but normally in sealed drums/cabinets" (see Table 4.3-1 and Table B-1 of the Plant Partitioning and Fire Ignition Frequency Report). This allowed the panel to consider amount, type and frequency when assigning storage influence factors consistent with the guidance in Section 6.5.7.2 of NUREG/CR-6850.

DAEC RAI PRA 75

F&O 5-27 concerning the documentation of operator interviews was dispositioned by stating that the FPRA documentation was updated to include documentation of the operator interviews in Appendix E of the FSR. NRC staff review of Appendix E describing the operator interviews indicates that insufficient information is provided to conclude that the requirements of SR HRA-A4 are met. Specifically, the SR requires that interviews confirm the interpretation of procedures relevant to actions identified in SRs HRA-A1, HRA-A2 and HRA-A3 are consistent with plant operation and training. These SRs identify for each scenario: safe shutdown actions carried over from the FPIE PRA, new fire specific safe shutdown actions and new undesired actions that could result from failure of single instrument. Provide further discussion and documentation of meeting SR HRA-A4.

RESPONSE:

SR HRA-A4 requires that actions identified in SR HRA-A1, HRA-A2, and HRA-A3 be talked through with plant operators to confirm the interpretation of the procedures is consistent with plant operational and training practices. Appendix H of DAEC-PSA-HR-04, Rev. 5, documents in detail the operator interviews performed for credited operator actions; however, these did not consider fire impacts. Section E.2 of the Fire Scenario Report discusses potential fire impacts on operator actions. The operator interviews for the fire PRA documented in Appendix E of the Fire Scenario Report were used to justify the assumptions used for modeling actions for fire events. These assumptions include procedure use, cue delays, access delays, and manpower requirements.

Additionally, the operator interviews were used to confirm the result of the instrumentation review (see Appendix E of the Fire Scenario Report). During the talk through, the Main Control Board was reviewed and no single instrument failure was identified that would result in an undesired action.

DAEC RAI PRA 77

The disposition of F&O 5-34 indicated that the FPRA HEP consistency review was documented by Table E-4 added to the FSR. While this table appears to address the consistency of FPRA HEPs and FPIE HEP, the consistency of the FPRA HEP relative to each other as required by SR HR-G6 is not specifically included. Discuss how the consistency review requirement of SR HR-G6 is met for CCII.

RESPONSE:

The consistency review was performed to review fire PRA HEPs and FPIE HEPs, as well as fire PRA HEPs relative to each other. However, Table E-4 of the Fire Scenario Report did not include the multiple columns related to the fire PRA HEPs. The attached table includes the consistency review table in its entirety. The table lists the actions in decreasing order by fire PRA HEPs. This allows an ordered comparison of the relative error probability of the actions in view of the data columns which contain timing and PSF information associated with each action. The "Comments" column provides any necessary explanation regarding final HEPs that may not appear consistent when compared to other HEPs.

The table includes:

- Basic Event ID – HEP designator
- Description – Basic event description
- HEPf – Fire PRA HEP
- Comments – Consistency review comments
- Execution Location – Plant location the action is performed
- Stress Level – Execution stress based on plant response, workload, and performance shaping factors (PSFs)
- Workload/PSF – Assigned workload and PSFs
- T_{sw} – System window
- T_m – Manipulation time
- T_{delay} – Delay time
- T_{rec} – Recovery time
- SPAR-H Ratio – Represents the number of times the action can be completed based on T_{sw}, T_m, and T_{delay}

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
DADS--ANOPS1-WA-HE--	OP FAILS TO INITIATE RPV EMERGENCY DEPRESS (ML-WATER) - FPRA VERSION	1.0E+00	Inadequate time to execute action in FPRA.	Control Room	High	High Workload / Negative PSFs	12	0.5	10.5	1	3.0
DCNDSRCNOPMECVACHE--	OPERATOR FAILS TO ALIGN MECHANICAL VACUUM PUMP - FPRA Version	1.0E+00	For FPRA, no cue is available except the closure of MSIVs on low main condenser vacuum.	Control Room	Moderate	Low Workload / Negative PSFs	60	1	10	49	50.0
DCNDSTCNOP-1546-HE--	OP FAILS TO OPEN CONDENSATE BYPASS LINE MOV-1546 - FPRA VERSION	1.0E+00	Set to 1.0 in FPRA; was at 0.1 as a screening value in FPIE model.		n/a	n/a	n/a	n/a	n/a	n/a	n/a
DCNDSTCNOPREP---HE--	OPERATOR FT OPEN BYPASS VALVE IN CONTROL ROOM - FPRA VERSION	1.0E+00	Set to 1.0 in FPRA; was at 0.1 as a screening value in FPIE model.		n/a	n/a	n/a	n/a	n/a	n/a	n/a
D----CNOPLL-WA-HE--	OP FAILS TO MANUAL INITIATE INJECTION SYSTEM GIVEN AUTO START FAILURE (LLOCA-WA)- FPRA VERSION	1.0E+00	Insufficient time in FPRA for LLOCA.	Control Room	High	High Workload / Negative PSFs	8	2	10	-4	-1.0
DCSPRYCNOPPCSCSTHE--	OP FAILS TO ALIGN CORE SPRAY SUCTION TO CST - FPRA Version	1.0E+00	This HEP is assigned a 1.0 failure probability (no formal HRA calculation) because this is a non-proceduralized action; the valve is for full-flow testing and not for use during accident scenarios.		n/a	n/a	n/a	n/a	n/a	n/a	n/a

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
DESW--DNOPXTIE--HE--	OPERATOR FAILS TO CROSS TIE ESW TRAINS - FPRA Version	1.0E+00	This is a non-proceduralized action and is assigned a 1.0 failure probability (no formal HRA calculation).		n/a	n/a	n/a	n/a	n/a	n/a	n/a
DFEED-CNOPSTRTLAHE--	OPERATOR FAILS TO INITIATE FEEDWATER (LARGE LOCA OR ATWS) - FPRA Version	1.0E+00	Insufficient time to complete the action due to assigned cue delay in Fire scenarios.	Control Room	High	High Workload / Negative PSFs	12	2	11	-1	0.5
DFEED-CNOPVEL-TTHE--	OP FAILS TO CONTROL FEEDWATER FOLLOWING SCRAM - FPRA Version	1.0E+00	Insufficient time to complete the action due to assigned cue delay in Fire scenarios.	Control Room	Moderate	Low Workload / Negative PSFs	3	0.25	10	-7.25	-28.0
DFPROTDNOPALPSBOHE--	OP FAILS TO ALIGN FIRE PROTECTION PER AIP-404 (SBO) - FPRA Version	1.0E+00	Insufficient time to complete the action due to assigned cue delay in Fire scenarios and increase execution time due to potential access delays for ex-CR execution areas.	Control Room, Pumphouse and Cooling Towers	High	High Workload / Negative PSFs	330	70	250	-5	0.9
DFPROTDNOPAOP410HE--	OP FAILS TO ALIGN FP TO STILLING BASIN PER AOP 410 FOR LATE INJECTION - FPRA Version	1.0E+00	Set to 1.0 in FPRA; was at 0.1 as a screening value in FPIE model.		n/a	n/a	n/a	n/a	n/a	n/a	n/a
DHPCI-CNOPL8TRIPHE--	OP FAILS TO SHUTOFF HPCI GIVEN L8 TRIP FAILURE - FPRA Version	1.0E+00	1.0 in FPIE due to insufficient time. 1.0 maintained for FPRA.		n/a	n/a	n/a	n/a	n/a	n/a	n/a

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
DHPCI-CNOPLPTRP-HE--	OPERATOR FAILS TO BYPASS HPCI LOW RPV PRESS TRIP - FPRA Version	1.0E+00	This is a non-proceduralized action for ATWS, this action is assigned a screening HEP of 1.0 in both FPIE and FPRA.		n/a	n/a	n/a	n/a	n/a	n/a	n/a
DINAIRANOPN3312-HE--	OP FAILS TO TRANSFER 1K1 COMPRESSOR POWER FROM 1B3312 TO 1B4501 - FPRA Version	1.0E+00	1.0 in FPIE due to insufficient time. 1.0 maintained for FPRA.		n/a	n/a	n/a	n/a	n/a	n/a	n/a
DINAIRANOPT265--HE--	Operator fails to manually align standby dryer train - FPRA Version	1.0E+00	1.0 in FPIE due to insufficient time. 1.0 maintained for FPRA.		n/a	n/a	n/a	n/a	n/a	n/a	n/a
DPDFP-DNOPALNPMPHE--	Operator fails to align portable diesel fire pump for low pressure injection - FPRA Version	1.0E+00	Insufficient time to complete the action due to assigned cue delay in Fire scenarios and increase execution time due to potential access delays for ex-CR execution areas.	Control Room, Pumphouse, and Reactor Building	High	High Workload / Negative PSFs	330	70	250	-5	0.9
DRCIC-CNOPL8TRIPHE--	OP FAILS TO SHUTOFF RCIC GIVEN L8 TRIP FAILURE - FPRA Version	1.0E+00	Set to 1.0 in FPRA; was at 0.1 as a screening value in FPIE model.		n/a	n/a	n/a	n/a	n/a	n/a	n/a
DRHR--CNOPLP-INJHE--	OP FAILS TO OPEN LPCI INJ MOV-1904(-2004) GIVEN MOV-2004(-1904) FAILS - FPRA Version	1.0E+00	Set to 1.0 in FPRA; was at 0.1 as a screening value in FPIE model.		n/a	n/a	n/a	n/a	n/a	n/a	n/a
DRHR--CNOPSPRYL-HE--	Operator Fails to Initiate Drywell Spray (LARGE LOCA) - FPRA Version	1.0E+00	1.0 in FPIE. 1.0 maintained for FPRA.		n/a	n/a	n/a	n/a	n/a	n/a	n/a

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
DRHR--CNOPSPRYSIHE--	OP Fails to Initiate DW Sprays for Debris Cooling (SI Node) - FPRA Version	1.0E+00	1.0 in FPIE. 1.0 maintained for FPRA.		n/a	n/a	n/a	n/a	n/a	n/a	n/a
DRHRSWDNOPLTINJLHE--	OP FAILS TO ALIGN ALT LP INJECTION (LARGE LOCA) - FPRA Version	1.0E+00	Insufficient time in FPRA for LLOCA.	Control Room	High	High Workload / Negative PSFs	8	5	10	-7	-0.4
DRHRSWDNOPUNOUT-HE--	Operator Fails to Dial Back Flow on RHRSW Pump after One Trip - FPRA Version	1.0E+00	Set to 1.0 in FPRA; was at 0.1 as a screening value in FPIE model.		n/a	n/a	n/a	n/a	n/a	n/a	n/a
DRVRW-DNOPV42-12HE--	OPERATOR FAILS TO OPEN MANUAL VALVE V42-0012 - FPRA Version	1.0E+00	Set to 1.0 in FPRA; was at 0.1 as a screening value in FPIE model.		n/a	n/a	n/a	n/a	n/a	n/a	n/a
DADS--ANOP-LVL2-HE--	Operator Fails to Depress Before RPV Fails Given Operator Failed in Level 1 - FPRA Version	5.8E-01	Conditional value maintained for FPRA.	Control Room	n/a	n/a	n/a	n/a	n/a	n/a	n/a
D-----CNOPLL-ST-HE--	OP FAILS TO MANUAL INITIATE INJECTION SYSTEM GIVEN AUTO START FAILURE (LLOCA-ST)- FPRA VERSION	4.6E-01	High HEP value - low Trec, potential for negative PSFs caused by the loss of control room indications due to fire impact.	Control Room	High	High Workload / Negative PSFs	14	2	10.5	1.5	1.8
DRHRSWDNOPLTINJ-HE--	OP FAILS TO ALIGN ALT LP INJECTION (TRAN, SLOCA, MLOCA, IORV) - FPRA Version	4.1E-01	High HEP value - low Trec, potential for negative PSFs caused by the loss of control room indications due to fire impact.	Control Room	High	High Workload / Negative PSFs	29	5	14	10	3.0

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
D-----CNOPML-WA-HE--	OP FAILS TO MANUAL INITIATE INJECTION SYSTEM GIVEN AUTO START FAILURE (MLOCA-WA)- FPRA VERSION	1.8E-01	High HEP value - low Trec, potential for negative PSFs caused by the loss of control room indications due to fire impact.	Control Room	High	High Workload / Negative PSFs	15	2	10.5	2.5	2.3
DCRD--CNOPFTRNIIHE--	OPERATOR FAILS TO MAXIMIZE CRD FLOW FOR IORV AND MLOCA-ST - FPRA VERSION	9.8E-02	High HEP value - low Trec, potential for negative PSFs caused by the loss of control room indications due to fire impact.	Control Room	High	High Workload / Negative PSFs	15	1	10.25	3.75	4.7
DPCONTNNOPHUISOLHE--	OPERATOR FAILS TO ISOLATE PATH GIVEN ISOLATE SIGNALS FAIL - FPRA Version	6.0E-02	For FPRA, the FPIE screening value is maintained for use give this is a post core damage action.		n/a	n/a	n/a	n/a	n/a	n/a	n/a
DRHR--CNOPSPRYM-HE--	OPERATOR FAILS TO INITIATE DRYWELL SPRAY (MEDIUM LOCA IORV) - FPRA Version	4.9E-02	Fairly high HEP value - low Trec, potential for negative PSFs caused by the loss of control room indications due to fire impact.	Control Room	High	High Workload / Negative PSFs	20	1	0.67	18.33	19.3
DRHR--CNOPSPRYS-HE--	OPERATOR FAILS TO INITIATE DRYWELL SPRAY (SMALL LOCA) - FPRA Version	4.8E-02	Fairly high HEP value - low Trec, potential for negative PSFs caused by the loss of control room indications due to fire impact.	Control Room	High	High Workload / Negative PSFs	30	1	10.5	18.5	19.5
DRHR--CNOPSPCLT-HE--	CONDITIONAL HEP FOR LATE SPC INITIATION GIVEN FAILURE IN EARLY TIME FRAME	4.6E-02	For FPRA, this conditional HEP is calculated using the individual FPRA HEP values.		n/a	n/a	n/a	n/a	n/a	n/a	n/a

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
DCBHV-NNOPDORFANHE--	OP FAILS TO PROP OPEN CB DOORS OR START APP R FANS - FPRA VERSION	2.7E-02	HEPf dominated by CBDTM values due to potential loss of cue instrumentation in fire and reliance on operator detection of high temperatures. This cue is judged as reasonable because of the long system window. The PSFs are "negative" by default given the Hot environment selected in the "Execution PSF" tab. Due to the sufficient amount of time available to complete the action in the event of a fire (Trec > 1 hour), the execution stress for this action has not changed for the FPRA.	Essential Switchgear Room	Moderate	Low Workload / Negative PSFs	1620	30	1510	80	3.7
D-----CNOPML-ST-HE--	OP FAILS TO MANUAL INITIATE INJECTION SYSTEM GIVEN AUTO START FAILURE (MLOCA-ST, IORV/SORV)-FPRA VE	2.4E-02	Fairly high HEP value - low Trec, potential for negative PSFs caused by the loss of control room indications due to fire impact.	Control Room	High	High Workload / Negative PSFs	45	2	21	22	12.0
DADS--ANOPS2-WA-HE--	OP FAILS TO INITIATE RPV EMERGENCY DEPRESS (SL-WATER) - FPRA VERSION	1.9E-02	Fairly high HEP value - low Trec, potential for negative PSFs caused by the loss of control room indications due to fire impact.	Control Room	High	High Workload / Negative PSFs	21	0.25	12.4	8.35	34.4

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
DCRD--CNOPFTRNI-HE--	OPERATOR FAILS TO MAXIMIZE CRD FLOW FOR TRANSIENTS AND SLOCA-ST - FPRA VERSION	1.8E-02	Fairly high HEP value - low Trec, potential for negative PSFs caused by the loss of control room indications due to fire impact.	Control Room	High	High Workload / Negative PSFs	30	1	10.5	18.5	19.5
D125DCENOPLDSHEDHE--	Operator fails to load shed batteries during SBO - FPRA Version	1.7E-02	Fairly high HEP value - For the FPIE version of this action, the assigned execution stress level is moderate. For FPRA, the execution stress level is not increased because it is assumed that most fires will be extinguished or contained within 65 minutes of the start of the fire (based on Appendix P of NUREG/CR-6850). The cue for this action occurs after the 65 minute point. As the given operator action is not necessary within the first 65 minutes, the fire can be assumed to be out and thus not continuing to cause delayed spurious activity and other late-scenario complicating disturbances, and that there is sufficient time available to diagnose and execute the action (SPAR-H ratio = 2.5).	Essential Switchgear Room	Moderate	Low Workload / Negative PSFs	120	20	70	30	2.5

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
DSYSTM-NOP-302-1HE--	OP FAILS TO PERFORM LOCAL STARTS PER AOP 302.1 - FPRA Version	1.5E-02	Fairly high HEP value - Trec < 30 min, potential for negative PSFs caused by the loss of control room indications due to fire impact.	Essential Switchgear Room	High	High Workload / Negative PSFs	55	20	10	25	2.3
D-----CNOPSL-WA-HE--	OP FAILS TO MANUAL INITIATE INJECTION SYSTEM GIVEN AUTO START FAILURE (SLOCA-WA) - FPRA VERSION	1.1E-02	Fairly high HEP value - Trec < 30 min, potential for negative PSFs caused by the loss of control room indications due to fire impact.	Control Room	High	High Workload / Negative PSFs	36	2	12.4	21.6	11.8
DTSC--ENOPALNTSCHE--	OPERATOR FAILS TO ALIGN TSC DIESEL - FPRA Version	1.1E-02	SPAR-H ratio close to 2, potential for negative PSFs caused by the loss of control room indications due to fire impact.	Essential Switchgear Room	High	High Workload / Negative PSFs	240	75	70	95	2.3
D-----CNOPTRSLSTHE--	OP FAILS TO MANUAL INITIATE INJECTION SYSTEM GIVEN AUTO START FAILURE (TRANSIENT, SLOCA-ST) - FPRA V	1.0E-02	The FPIE version of this action used high workload but optimal PSFs to arrive at a moderate execution stress level. For FPRA, the execution stress is increased to high due to the potential negative PSFs if control room indications fail from fire impact.	Control Room	High	High Workload / Negative PSFs	50	2	14.1	33.9	18.0

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
DHPCI-CNOP15----HE--	OP FAILS TO TRANSFER HPCI FROM CST TO TORUS IN SUFFICIENT TIME - FPRA Version	1.0E-02	The FPIE version of this action used low workload and optimal PSFs to arrive at a low execution stress level. For FPRA, the execution stress is increased to moderate due to the potential negative PSFs if control room indications fail from fire impact.	Control Room	Moderate	Low Workload / Negative PSFs	45	0.5	10	34.5	70.0
DFEED-CNOPSTRT--HE--	OPERATOR FAILS TO INITIATE FEEDWATER - FPRA Version	6.9E-03	Trec > 30 min; The FPIE version of this action used low workload and optimal PSFs to arrive at a low execution stress level. For FPRA, the execution stress is increased to moderate due to the potential negative PSFs if control room indications fail from fire impact.	Control Room	Moderate	Low Workload / Negative PSFs	45	2	10.25	32.75	17.4
DSYSTM-NOP-PCFLDHE--	Operators Fail to Implement Primary Containment Flooding - FPRA Version	5.6E-03	Given the excessive LOCA initiator, this HEP calculation assumes a "High" workload applies. The PSFs are "negative" by default given the Complexity of the execution selected in the "Execution PSF" tab. For FPRA, there is no change given the time available.	Control Room	High	High Workload / Negative PSFs	300	25	40	235	10.4

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
D125DCENOPALNCHGHE--	OPERATOR FAILS TO ALIGN STANDBY 125V DC BATTERY CHARGER - FPRA VERSION	4.8E-03	Due to the sufficient amount of time available to complete the action in the event of a fire, the execution stress for this action has not changed for the FPRA.	Essential Switchgear Room	Moderate	Low Workload / Negative PSFs	240	25	10	205	9.2
DADS--ANOP150PSIHE--	OP VIOLATES AOP 301 150 PSI CAUTION - FULL RPV ED PERFORMED IN SBO - FPRA VERSION	4.8E-03	The FPIE version of this action used low workload and optimal PSFs to arrive at a low execution stress level. For FPRA, the execution stress is increased to Moderate due to the potential negative PSFs if control room indications fail from fire impact.	Control Room	Moderate	Low Workload / Negative PSFs	282	2.5	228	51.5	21.6
DSYSTMNNOPRESTRTHE--	OP FAILS TO RESTART EQUIPMENT FOLLOWING LOOP - FPRA Version	3.6E-03	For FPRA, the execution stress is increased to moderate due to the potential negative PSFs if indications fail from fire impact.	Control Room	Moderate	Low Workload / Negative PSFs	55	2	10	43	22.5

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
DHPCI-CNOPOPENDRHE--	Operator Fails to Ventilate HPCI Room - FPRA Version	2.9E-03	The plant has lost all AC power and is running on battery power (SBO), the level of workload is reasonably assumed to be "High" for this HEP calculation. By default ("hot environment" and "emergency lighting" defined in tab "execution PSFs") the PSFs are "negative". For the FPIE version of this action, the assigned execution stress level is high. For FPRA, the execution stress level is increased by a factor of 2 for a total multiplier of 10.	Ex-MCR	High	(High Workload / Bad PSFs) in FPIE; x10 for FPRA	60	25	10	25	2.0
DCBHV-NNOPFTSHV-HE--	OP FAILS TO START STANDBY CB HVAC TRAIN - FPRA VERSION	2.8E-03	The FPIE version of this action used low workload and negative PSFs to arrive at a moderate execution stress level. Due to the sufficient amount of time available to complete the action in the event of a fire (SPAR-H ratio still > 3) and the fact that the action takes place in the control room, the execution stress for this action has not changed for the FPRA.	Control Room	Moderate	Low Workload / Negative PSFs	60	15	10	35	3.3

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
D250DCENOPCB4023HE--	OPERATOR FAILS TO ALIGN ALT 250VDC BATT CHGR W/I 4HOURS - FPRA Version	2.6E-03	Due to the sufficient amount of time available to complete the action in the event of a fire, the execution stress for this action has not changed for the FPRA. Based on Appendix P of NUREG/CR-6850, it is assumed that most fires will be extinguished or contained within 65 minutes of the start of the fire. If the given operator action is not necessary within the first 65 minutes, the fire can be assumed to be out and thus not continuing to cause delayed spurious activity and other late-scenario complicating disturbances, and that there is sufficient time available to diagnose and execute the action.	Essential Switchgear Room	Moderate	Low Workload / Negative PSFs	240	35	10	195	6.6
DCNDSTCNOPTINJ--HE--	OP FAILS TO INIT. COND. FOR ALT. INJ. (TRANSIENT EVENTS) - FPRA VERSION	2.4E-03	The FPIE version of this action used high workload but optimal PSFs to arrive at a moderate execution stress level. For FPRA, the execution stress is increased to high due to the potential negative PSFs if control room indications fail from fire impact.	Control Room	High	High Workload / Negative PSFs	56	0.5	10.5	45	91.0

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
DCRD--CNOPFTRNILHE--	OPERATOR FAILS TO INJECT THRU CRD (LATE TIMEFRAME) - FPRA Version	2.1E-03	The workload late in the accident sequence is considered to be low since the initial plant stabilization actions would be complete. For FPRA, no change required for workload given the extensive time available to complete the action. PSFs considered negative due to the potential for failed MCR indications.	Control Room	Moderate	Low Workload / Negative PSFs	390	1	258	131	132.0
DWELLWDNOPELLWTRHE--	OP Fails to Maximize Well Water to Maintain Condenser - FPRA Version	1.9E-03	Stress maintained at FPIE value of "low" due to the extensive time available.	Control Room and Pumphouse	Low	Low Workload / Optimal PSFs	300	20	10	270	14.5
DPHVACNNOPPHDORSHE--	OPERATOR FAILS TO OPEN PUMPHOUSE DOORS - FPRA Version	1.8E-03	For both FPIE and FPRA, the PSFs are "negative" by default given the "hot environment" selected in the "Execution PSF" tab. No change in workload considering that Trec is 60 min.	Pumphouse	Moderate	Low Workload / Negative PSFs	135	30	30	75	3.5
DADS--ANOPS1STRVHE--	OP FAILS TO INITIATE RPV EMERGENCY DEPRESS (ML-STEAM OR IORV) - FPRA VERSION	1.2E-03	The FPIE version of this action used high workload but optimal PSFs to arrive at a moderate execution stress level. For FPRA, the execution stress is increased to high due to the potential negative PSFs if control room indications fail from fire impact.	Control Room	High	High Workload / Negative PSFs	45	0.25	13.4	31.35	126.4

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
DADS--ANOPS2-ST-HE--	OP FAILS TO INITIATE RPV EMERGENCY DEPRESS (SL-STEAM) - FPRA VERSION	1.2E-03	The FPIE version of this action used high workload but optimal PSFs to arrive at a moderate execution stress level. For FPRA, the execution stress is increased to high due to the potential negative PSFs if control room indications fail from fire impact.	Control Room	High	High Workload / Negative PSFs	50	0.25	14.1	35.65	143.6
DADS--ANOPTRANS-HE--	OP FAILS TO INITIATE RPV EMERGENCY DEPRESS (TRANSIENTS) - FPRA VERSION	1.2E-03	The FPIE version of this action used high workload but optimal PSFs to arrive at a moderate execution stress level. For FPRA, the execution stress is increased to high due to the potential negative PSFs if control room indications fail from fire impact.	Control Room	High	High Workload / Negative PSFs	52	0.25	13.9	37.85	152.4
DCNDSTCNOP02----HE--	OPERATOR FAILS TO OPEN HOTWELL MAKEUP BYPASS LINE - FPRA VERSION	1.2E-03	The FPIE version of this action used high workload but optimal PSFs to arrive at a moderate execution stress level. For FPRA, the execution stress is increased to high due to the potential negative PSFs if control room indications fail from fire impact.	Ex-MCR	Moderate	Low Workload / Negative PSFs	180	20	90	70	4.5

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
DRCIC-CNOP2LPTRIHE--	Operator fails to bypass low RPV Pressure trip circuitry for RCIC - FPRA Version	1.1E-03	The PSFs are "negative" by default given the Emergency Lighting and the Complexity of the execution selected in the "Execution PSF" tab. For FPRA, there is no change in assigned workload given the time available for recovery.	Control Room	Moderate	Low Workload / Negative PSFs	240	2	70	168	85.0
DN2---ANOPGRP3BYHE--	OP FAILS TO BYPASS GROUP 3 ISOLATION SIGNAL AND REOPEN CV-4317A - FPRA Version	1.0E-03	FPIE: The action diagnosis occurs in the early stages of the scenario and ample time is available to perform the action successfully, but the action is likely taken around the time it is diagnosed, so that workload is considered to be high to account for the actions associated with the initial stabilization of the plant. The FPIE version of this action used high workload but optimal PSFs to arrive at a moderate execution stress level. For FPRA, the execution stress is increased to High due to the potential negative PSFs if indications fail from fire impact.	Control Room	High	High Workload / Negative PSFs	30	0.5	0	29.5	60.0

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
DADS--NOPAFLVL2HE--	Operator Fails to Depress Before RPV Fails Given Operator Failed in Level 1	7.0E-04	The FPIE version of this action used high workload but optimal PSFs to arrive at a moderate execution stress level. For FPRA, the execution stress is increased to high due to the potential negative PSFs if control room indications fail from fire impact.	Control Room	High	High Workload / Negative PSFs	120	0.25	14.1	105.65	423.6
DRHR--CNOPSPCELYHE--	OPERATOR FAILS TO INITIATE TORUS COOLING (EARLY TIMEFRAME) - FPRA Version	4.1E-04	Initiation of SPC is a very routine action for the operating crew. The FPIE version of this action assigned low execution stress. For FPRA, the execution stress is not increased because of the extensive time available for recovery for this action (over 3 hours).	Control Room	Low	Low Workload / Optimal PSFs	228	5	25	198	40.6
DSBGT-CNOP-VENT-HE--	OP FAILS TO VENT PRIMARY CONTAINMENT (EOP-2 Step PC/P-10) - FPRA Version	2.7E-04	The PSFs are "negative" by default given the Complexity of the execution selected in the "Execution PSF" tab. For FPRA, there is no change in the assigned workload given the time available.	Control Room	Moderate	Low Workload / Negative PSFs	1188	12	744	432	37.0

Fire PRA HEP Consistency Review Table

Basic Event ID	Description	HEPf	Comments	Execution Location	Stress Level	Workload/PSFs	Tsw	Tm	Tdelay	Trec	SPAR-H ratio
DCNDSRCNOPPRESCTHE--	OP FAILS TO PROPERLY CONTROL RPV PRESSURE WITH TBVs - Fire PRA Version	1.0E-04	The FPIE version of this action used low workload and optimal PSFs to arrive at a low execution stress level. For FPRA, the execution stress is kept the same given the time available.	Control Room	Low	Low Workload / Optimal PSFs	600	0.5	10	589.5	1180.0
DRHR--CNOPSPCNATHE--	OP FAILS TO ALIGN TORUS COOLING (NON-ATWS) - FPRA Version	1.9E-05	Initiation of SPC is a very routine action for the operating crew. The FPIE version of this action assigned a low execution stress level. For FPRA, the execution stress is not increased because of the extensive time available for recovery for this action (over 19 hours).	Control Room	Low	Low Workload / Optimal PSFs	1200	5	20	1175	236.0

DAEC RAI PRA 78

It is noted that the LAR identifies certain motor operated valves (MOVs) which are subject to spurious operation as described in Information Notice (IN) 92-18, "Potential For Loss Of Remote Shutdown Capability During A Control Room Fire." These valves include steam supply valves in the HPCI and RCIC systems, Core Spray, residual heat removal (RHR) system, as well as others. Describe whether any CIVs are included as a IN 92-18 MOV in the FPRA, and, if so, describe the assumptions and methods that are applied in modeling them. Describe whether all the IN 92-18 MOVs are included in the FPRA and how they are addressed for this application (i.e., are they treated as VFDRs). Also, it is noted that Appendix S of the LAR does not show modifications for IN 92-18 valves. Confirm or clarify if there are no modifications associated with the IN 92-18 valves.

RESPONSE:

The fire PRA does not credit manual operation of MOVs given fire damage. That is, if cable damage to a valve is postulated then the applicable failure mode is assumed and not recovered. This treatment extends to the modeled CIVs.

IN 92-18 MOVs were identified as VFDRs and are included in the fire PRA treatment. That is, if cable damage is postulated for a MOV then the MOV is assumed failed for alternate shutdown capability, as well. Report Number 0027-0042-000-004 Duane Arnold Energy Center Fire Risk Evaluation, Attachment – Fire Area CB1, identifies the applicable VFDRs and the fire PRA treatment for the variant and compliant cases.

There are no modifications associated with the IN 92-18 valves themselves. Failures of IN 92-18 valves contribute to high CCDP values. For the MCR, incipient detection is being installed to minimize the impact of a fire in a MCR panel that may cause IN 92-18 valves to fail.

DAEC RAI PRA 79

Appendix G of the FSR provides fire damage probability for single cable bundles and multiple cable bundles. According to Section 5.1.5.1, single cable bundle results are used for ventilated/open high voltage switchgear, MCC, and AC/DC distribution panels, while multiple cable bundle results are used for ventilated/open load centers and other type of electric panels. Provide the basis for the assignment of single or multiple cable bundle results to the specific cabinet types.

RESPONSE:

The guidance in NUREG/CR-6850 Appendix G was used to assign heat release rates to the various panel types. Section 5.1.5.1 of the Fire Scenario Report (0493080001.003) identifies how the guidance was applied to the fire PRA.

The following panel types were assigned the single cable bundle heat release rate distribution based on the recommendations in the discussions on the identified NUREG/CR-6850 Appendix G page:

- 4160V Switchgears based on page G-25
- 480V MCCs based on page G-25 – note that guidance is not provided for lower voltage MCCs. The fire PRA treated lower voltage MCCs consistent with 480V MCCs.
- Distribution panels based on page G-28

The following panel types were assigned the multiple cable bundle heat release rate distribution based on the recommendations in the discussions on the identified page:

- 480V Switchgears based on page G-29
- Inverters based on page G-29
- Control Panels based on page G-33 – the recommendations for large panels were used
- Relay Racks based on page G-34

Other Panel types without specific NUREG/CR-6850 guidance (e.g., battery chargers) were assigned the multiple cable bundle heat release rate distribution.

DAEC RAI PRA 80

Table V-1 of the LAR shows that the peer review had findings on SRs UNC-A1 and UNC-A2, which had been addressed. However, Table V-3 of the LAR does not include these. Clarify or provide the corresponding Table V-3 information for these SRs.

RESPONSE:

The Duane Arnold Energy Center Fire PRA Peer Review Report Using the ASME PRA Standard Requirements documents the peer review findings. Appendix C of the report provides the details of each finding and identifies the SR that the finding was originated from, as well as associated SRs. Table V-3 of the LAR (ML11221A280) identifies the findings and associates them with the fire PRA SR that the finding was originated from. The findings associated with SRs UNC-A1 and UNC-A2 were originated from other fire PRA SRs. The findings associated with SR UNC-A1 are 2-17, 5-15, 5-16, and 5-18. The findings associated with SR UNC-A2 are 1-5, 4-18, and 4-28. These findings are included in Table V-3.

DAEC RAI PRA 81

Explain why SR FSS-C3 was determined to be not applicable in the LAR Table V-1.

RESPONSE:

The postulated fire scenarios use the results of the Generic Fire Modeling Treatments report and Supplemental Generic Fire Model Treatments: Hot Gas Layer Tables. Per Section 6.1 of the Generic Fire Modeling Treatments, the results are based on a constant heat release rate and do not include burnout. Per Section 1.0 of The Supplemental Generic Fire Modeling Treatments: Hot Gas Layer Tables, the results may include fire growth but do not include burnout. Therefore, SR FSS-C3 was determined to be not applicable.