NEXTera ENERGY
DUANE ARNOLD

April 27, 2018

NG-18-0046
10 CFR 50.55a

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, Fifth Inservice Inspection Interval Program Plan, Relief Request RR-05

References: 1) Letter, Curtland (NextEra) to U.S. NRC, "Fifth Inservice Inspection Interval Program Plan, Relief Request No. RR-05," dated November 16, 2017 (ML17325B215)
2) Electronic Communication, Final RAI for Duane Arnold (DAEC) Re: 5th – 10 year ISI (RR-05), dated March 28, 2018.

In the Reference 1 letter, NextEra Energy Duane Arnold, LLC (hereafter NextEra Energy Duane Arnold) submitted Relief Request RR-05 in support of our Fifth Inservice Inspection Interval Program Plan pursuant to 10 CFR 50.55a. The NRC Staff requested, via Reference 2, additional information regarding Relief Request RR-05.

The Enclosure to this letter contains the requested information.

This letter does not contain any new or revised commitments.

If you have any questions or require additional information, please contact J. Michael Davis at 319-851-7032.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on April 27, 2018

Mike Strope for Dean Curtland

Dean Curtland
Site Director
NextEra Energy Duane Arnold, LLC

Enclosure

cc: NRC Regional Administrator
NRC Resident Inspector
NRC Project Manager

Enclosure to NG-18-0046

Response to Request for Additional Information, Fifth Inservice Inspection Interval Program Plan, Relief Request RR-05

15 pages follow

1.0 INTRODUCTION

In the Reference 1 letter, NextEra Energy Duane Arnold, LLC (hereafter NextEra Energy Duane Arnold) submitted Relief Request RR-05 in support of our Fifth Inservice Inspection Interval Program Plan pursuant to 10 CFR 50.55a. The NRC Staff requested, via Reference 2, additional information regarding Relief Request RR-05. The requested information is given below.

2.0 REQUEST FOR ADDITIONAL INFORMATION

By letter dated November 16, 2017 (ADAMS Accession No. ML17325B215), NextEra Energy Duane Arnold, LLC (the licensee) submitted a relief request (RR), as part of the Fifth-10 year Inservice Inspection Interval Program Plan for Duane Arnold Energy Center (DAEC). RR-05 is being requested in order for DAEC to continue the implementation of the previously approved Risk-Informed Inservice Inspection Program (RI-ISI), as an alternative to the current requirements of Class 1 and 2 examination Categories B-F, B-J, C-F-1, and C-F-2 as specified in Table IWB 2500-1, and Table IWC 2500-1 of the 2007 Edition with 2008 Addenda of ASME Code Section XI.

The U.S. Nuclear Regulatory Commission (NRC) staff is reviewing your proposed request and has identified an area where additional information is needed to complete its review. The following requests for additional information (RAIs) require licensee responses:

APLA-RAI-01

In DAEC RR-05 submittal, page 5, "Risk Impact Analysis," the licensee states, in part:

> The revised program continues to be less than the EPRI criterion with a total change in plant risk of 4.34E-09 in regards to CDF and 4.3E-09 in regards to LERF.

<u>NRC Staff Request:</u>

a. Regarding the statement above, since LERF is typically at least an order of magnitude lower than CDF, explain why the total changes in CDF and LERF are essentially equal.

b. Explain why the 03RWCU System parameter (exhibited in the Risk Impact Results matrix), on page 5, has a change in LERF exceeding that of CDF. The NRC staff is requesting clarification in the data set since CDF values would typically be greater than LERF values.

DAEC Response:

In accordance with EPRI Topical Report TR-112657, Rev. B-A, the change in core damage frequency (CDF) due to application of the risk-informed inservice inspection (RI-ISI) process was estimated based on the equation presented below.

- $\Delta R_{CDF} = CCDP*RF*[(POD_S*N_S) - (POD_R*N_R)]$

CCDP	Conditional Core Damage Probability
RF	Rupture Frequency
POD_S	Probability of Detection associated with the ASME Section XI Code Program
POD_R	Probability of Detection associated with the EPRI TR-112657 Program
N_S	Number of Inspection Locations in the ASME Section XI Code Program
N_R	Number of Inspection Locations in the EPRI TR-112657 Program

This equation was applied to groups of welds within a system that are in the same Risk Category with the same consequence rank (i.e., same CCDP), and are susceptible to the same degradation mechanisms (i.e., same RF and PODs). The results for each of these groups within a given system were then added together to get the cumulative value for that system.

The change in large early release frequency (LERF) (ΔR_{LERF}) due to application of the RI-ISI process was estimated by substituting the conditional large early release probability (CLERP) for CCDP in the above equation.

Most of the values in the equation are predetermined by the EPRI RI-ISI methodology. However, for High Consequence piping segments, the equation uses the highest plant-specific values of CCDP and CLERP for the piping in the scope of the application. At most plants, the highest CCDP is typically around a factor of 10 greater than the highest CLERP. When these numbers are inserted into the equation, they result in a corresponding change in CDF greater than the change in LERF. However, at Duane Arnold the highest CCDP and CLERP are both 3.00E-03. This is because this value corresponds to piping in the reactor water cleanup (RWCU) system located outside containment where a break is just as likely to result in large early release as it is to core damage. When equal values for the maximum CCDP and CLERP are inserted into the equation, they result in a change in risk that is nearly the same for LERF as it is for CDF. Therefore, the reason the changes in CDF and LERF are so close is because the maximum CCDP and CLERP at Duane Arnold are equal.

This is the same factor that leads to the change in LERF being slightly higher than the change in CDF for the RWCU system. Since the CCDP and CLERP are the same for most of the Risk Categories, the calculation results in equal changes in CDF and LERF for these Risk Categories. However, for Risk Category 5 in the RWCU system,

predetermined values of CCDP and CLERP are used instead of the plant-specific maximum values. This results in the following changes in CDF and LERF for Risk Category 5:

ΔR_{CDF} = CCDP*RF*[(POD_S*N_S) – (POD_R*N_R)]
ΔR_{CDF} = 1.00E-04*2.00E-07*[(0.3*2) – (0.9*1)] = -6.00E-12

ΔR_{LERF} = CLERP*RF*[(POD_S*N_S) – (POD_R*N_R)]
ΔR_{LERF} = 1.00E-05*2.00E-07*[(0.3*2) – (0.9*1)] = -6.00E-13

For all the other Risk Categories in the RWCU system, the ΔR_{CDF} and ΔR_{LERF} are the same and have a combined positive value of 7.50E-11. When adding the negative values for ΔR_{CDF} and ΔR_{LERF} in Risk Category 5 to the values for the remaining Risk Categories in RWCU, the smaller decrease in ΔR_{LERF} results in a larger overall positive value (7.44E-11) than the overall positive value for ΔR_{CDF} (6.90E-11).

Therefore, the calculation results support the change in LERF being slightly higher than the change in CDF for the RWCU system.

APLA-RAI-02

According to Regulatory Issue Summary 2007-06, the NRC would expect licensees to fully address all scope elements consistent with Revision 2 of Regulatory Guide (RG) 1.200 by the end of 2009 (i.e., one year implementation period). Revision 2 of RG 1.200 endorses, with exceptions and clarifications, the combined American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard (ASME/ANS RA-Sa-2009).

In Attachment 2, "DAEC PRA Model Quality Summary," page 2 (of the submittal), the licensee states, in part:

> The focused scope peer review conducted in March 2011 ...utilized version RA-Sa-2009 of the ASME Standard as endorsed and clarified by the NRC in Regulatory Guide 1.200, Revision 2.

<u>NRC Staff Request:</u>

The NRC staff is requesting the licensee to confirm that all technical elements in Part 2 of the Standard, including internal flooding, were included in this March 2011 review. If not, please identify any gaps between the peer review and the requirements in RG 1.200, Revision 2, particularly focusing on those technical elements for which the most current peer review remains from 2007 against Rev. 1 of RG 1.200.

DAEC Response:

The 2011 DAEC PRA Focused Peer Review is a review of the Technical Elements of the internal events--including internal flood--at-power PRA with a focus on DAEC's disposition of findings and suggestions and PRA upgrades as a result of the 2007 peer review.

Table 3-1 of the 2011 focused peer review report provides a summary of results for both the 2007 full-scope peer review and the 2011 focused-scope peer review. In the table, all supporting requirement identifiers--including those for internal flood--from the RG 1.200, Revision 1 era standard are listed and the assessment result (met, not met, etc.) and assessment basis for each one is provided. The corresponding surveillance requirement from the RG 1.200, Revision 2 era standard is provided on the same row and the assessment result and basis for assessment for those evaluated in 2011 are provided.

APLA-RAI-03

The NRC staff identified during its review the need for the licensee to further address several previously identified gaps that were incorporated into the DAEC PRA. Specifically, more information is requested re: DAEC Focused Peer Review Findings (Table C-1) which include:

APLA-RAI-03(a) Findings & Observations (F&O) DA-D4-01A, which the licensee states, in part:

> To address this associated finding, the reasonableness of the prior and posterior distributions was reviewed; it was concluded no model changes were required.

This implies that the mean of the likelihood (3/544 = 0.0055) was significantly higher (a factor of 60) than that for the prior (9E-5). The NRC staff determined this may suggest a Bayesian approach may not be appropriate if plant-specific behavior is significantly different (worse) than generic.

NRC Staff Request:

Did these values change subsequently such that the focused scope review had different values where the prior and likelihood were more aligned? Provide the results of the final Bayesian analysis and the conclusion by the peer review team that this Supporting Requirement (SR) is now MET.

DAEC Response:

Supplement 1 to the DAEC PRA Data Analysis notebook provides discussion on the reasonableness of posteriors documented in Appendix C of the notebook. The supplement was prepared with the intent of closing F&O DA-D4-01A.

NextEra Energy Duane Arnold's resolution to F&O DA-D4-01A was assessed during an independent closure review of open F&Os for DAEC's Internal Events and Fire PRAs. The review, conducted in July 2017, was performed in accordance with the process documented in Appendix X to NEI 05-04, as well as the requirements published in the ASME/ANS PRA Standard (RA-Sa-2009) and Regulatory Guide 1.200, Revision 2. The review team concluded that NextEra Energy Duane Arnold's resolution is adequate and complete.

APLA-RAI-03(b) F&O IE-B3-01A, which the licensee states, in part:

> Several findings and suggestions under HLR-A and HLR-B have been dispositioned / resolved, but the subsuming (IE-B3) and screening (IE-C4 (C6)) of initiating events does not meet the standard.

In particular, the Finding cited, (1) RBCCW is subsumed by TT, but not failed given TT; (2) GSW is subsumed by TC, but not failed given TC; (3) 1A1/1A2 bus failures and partial loss of feedwater are binned to TT, but their impact not modeled given TT; and (4) 1A3/1A4 bus failures are subsumed with TT, but unavailability of emergency power given failure of the bus. Additionally, the licensee stated that the CDF is 3.5E-09/yr.

NRC Staff Request:

Provide the NRC staff with information regarding these deficiencies due to subsuming, specifically are they rectified? Additionally, the information provided in the disposition cites conservatisms with respect to piping leg breaks not yet incorporated into the model. It is not clear to the NRC staff if any sequences initiated by or subsequently involving these breaks could be affected by the application such that failure to incorporate these into the PRA could underestimate the change in CDF or LERF. If so, provide information (e.g., a sensitivity evaluation) that includes these into the applicable risk metrics and determine any effect.

DAEC Response:

NextEra Energy Duane Arnold's resolution to F&O IE-B3-01A was assessed during the 2017 independent closure review of open F&Os for DAEC's Internal Events and Fire PRAs. The review team concluded that NextEra Energy Duane Arnold's resolution is adequate, but the F&O remains open pending update of associated documentation.

Deficiencies identified in F&O IE-B3-01A were rectified in the Revision 7 PRA by incorporating initiating event fault trees for River Water Supply, General Service Water, non-essential 4KV power distribution buses 1A1 and 1A2, and essential 4KV power distribution buses 1A3 and 1A4. Acceptability of not including initiating event fault trees for reference and variable leg breaks is based on results of a sensitivity case described in Table C-1 where the associated CDF is estimated to be only 3.5E-09 per year. (Loss of the Feedwater, HPCI, and RCIC high pressure systems is assumed in the sensitivity case.) Not including an initiating event fault tree for Reactor Building Closed Cooling Water (RBCCW) is likewise acceptable since associated impacts on cooled equipment

are already accounted for by incorporation of the General Service Water (GSW) initiating event fault tree into the PRA; the RBCCW system is cooled by the GSW system and it therefore loses its ability to cool downstream components when the running GSW pumps trip.

For the Risk-informed ISI application, values of conditional core damage probability and large/early release probability are calculated given the occurrence Class 1 and Class 2 pipe segment breaks. In general, pipe segments are evaluated using fluid release initiating events. In some cases, an initiating event such as turbine trip or loss of feedwater is used with additional equipment failures to represent impact of flood. None of the cases however, use or interact with the initiating event fault trees incorporated into the Revision 7 PRA to address F&O IE-B3-01A. Not including reference and variable leg break initiating event fault trees in the PRA and not including an RBCCW initiating event fault tree in the PRA do not impact the Risk-informed ISI application because the pipe segments for these systems are not in scope.

APLA-RAI-03(c) F&O HR-A1-01A and F&O HR-A2-01A, which the licensee states, in part:

> The approach used in the DAEC PRA was different than currently prescribed in the standard, but is considered capable of accurately identifying pre-initiators. As such this variance from the standard has no impact on this application.

NRC Staff Request:

Provide the NRC staff with information regarding why the use of this "different approach", as stated above, accurately identifies pre-initiators such that it is equivalent to the approach required by the Standard (ASME/ANS RA-Sa-2009).

DAEC Response:

NextEra Energy Duane Arnold's resolution to F&Os HR-A1-01A and F&O HR-A2-01A was assessed during the 2017 independent closure review of open F&Os for DAEC's Internal Events and Fire PRAs. The review team concluded that NextEra Energy Duane Arnold's resolution is adequate, but the F&O remains open pending update of associated documentation.

The basis of and process used for identifying pre-initiator human error probability (HEP) events for inclusion in the PRA are described in Section 2.3.3, Section 4.3, and Appendix J of DAEC's PRA Human Reliability Analysis Notebook. Approaches for finding both restoration type and calibration type HEPs are systematic as required by high level requirement HLR-HR-A, but have a focus on 'systems analysis' rather than on 'procedures and practices' as specified in supporting requirements HR-A1 and HR-A2. As stated in the notebook, "Review of procedures alone would result in overlooking pre-initiator HEPs that would otherwise be identified by understanding and reviewing system configuration and operational aspects."

Table 1 compares the actual steps used for identifying restoration error HEPs to steps that would be used if a 'procedure and/or practices' approach had been used. Likewise, Table 2 compares the actual steps used for identifying calibration error HEPs to steps that would be used if a 'procedure and/or practices' approach had been used. The right-hand column of each table states how DAEC's process meets the intent of high level requirement HLR-HR-A and supporting requirements HR-A1 and HR-A2.

Table 1

Equivalency of DAEC Process for Identifying Restoration Error HEPs

SR HR-A1	DAEC PRA	Equivalency
Identify maintenance and testing procedures and/or practices performed on SSCs that are considered in the PRA.	Identify system trains explicitly modeled in the PRA. These are potential candidates for pre-initiator HEPs.	The result is identical. Both screen out procedures for systems not modeled in PRA.
Review the relevant maintenance and testing procedures and/or practices and identify those which require realignment of equipment.	Use P&IDs, System Descriptions and/or Operating Instructions to assist in the pre-initiator HEP identification process.	System description used instead of reviewed all procedures.
Identify specific test, inspection, and maintenance activities with potential to leave systems outside their normal operational or standby status.	System trains that are in standby at-power, as opposed to normally running at-power, are potential candidates for pre-initiator HEPs. For multi-train systems in which one train may be assumed to be normally running and the other(s) assumed to be in standby, all trains are assumed to be in standby for the pre-initiator HEP identification process.	Same result achieved. One through review of system trains, the other through review of procedures.
Screen out procedures that are performed on SSCs with compelling indication such as an annunciator or monitor recognizable by operator before, during or immediately after restoration to service	System trains with one or more manual valves in the pathway NOT equipped with position indication in the Control Room.	Instead of screening procedures performed on SSCs, the SSCs were screened. The result is equivalent.
Screen out procedures that are performed on SSCs that can automatically be re-aligned on demand	System trains with Controlled valves (e.g., MOVs, AOVs, SOVs) in the pathway that do NOT automatically re-align	Instead of screening procedures performed on SSCs, the SSCs were screened. The result is

SR HR-A1	DAEC PRA	Equivalency
	upon system demand, and are NOT equipped with position indication in the Control Room.	equivalent.
Not required by the standard	Identify and review DAEC plant experience related to potential restoration errors to identify additional pre-initiator actions worthy of inclusion in the modeled but not identified by the previous step.	Review of plant experience exceeds the requirements of the Standard.

Table 2

Equivalency of DAEC Process for Identifying Miscalibration HEPs

SR HR-A2	DAEC PRA	Equivalency
Identify calibration procedures and/or practices performed on SSCs that are considered in the PRA.	Identify system trains explicitly modeled in the PRA. These are potential candidates for pre-initiator HEPs.	The result is identical. Both screen out procedures for systems not modeled in PRA.
Review the relevant calibration procedures and/or practices and identify those involving automatic initiation of standby safety equipment.	Use P&IDs, System Descriptions and/or Operating Instructions to assist in the pre-initiator HEP identification process.	System description used instead of reviewed all procedures.
Identify specific calibration activities with potential to adversely impact automatic initiation of standby safety equipment.	System trains that are in standby at-power, as opposed to normally running at-power, are potential candidates for pre-initiator HEPs. For multi-train systems in which one train may be assumed to be normally running and the other(s) assumed to be in standby, all trains are assumed to be in standby for the pre-initiator HEP identification process.	Same result achieved. One through review of system trains, the other through review of procedures.
IDENTIFY, through a review of procedures and practices, those calibration activities that if performed incorrectly have an adverse impact on the automatic	Identify I&C components the miscalibration of which will impact redundant system trains or redundant system components. Miscalibrations	Both screen in miscalibration impacting PRA.

SR HR-A2	DAEC PRA	Equivalency
initiation of standby safety equipment.	that impact a single component may be screened from further consideration and assumed to be inherent in the component failure rate.	
Not required by the standard	Identify and review DAEC plant experience related to miscalibration errors to identify additional pre-initiator actions worthy of inclusion in the modeled by not identified by the previous step.	Review of plant experience exceeds the requirements of the Standard.

APLA-RAI-03(d) F&O HR-C1-01A, which the licensee states, in part:

> In other cases, the [Human Failure Events] HFE is at the train level, but no corresponding system level dependent HFE is included.

NRC Staff Request:

The F&O disposition cites addition of train-level pre-initiating HFEs to the model. However, it is not clear to the NRC staff if this deficiency regarding "corresponding system-level dependent HFEs" has been corrected as well. If not, provide the NRC staff with information (e.g., performance of a sensitivity evaluation) incorporating these dependencies and assess any effect on the risk metrics.

DAEC Response:

NextEra Energy Duane Arnold's resolution to F&O HR-C1-01A was assessed during the 2017 independent closure review of open F&Os for DAEC's Internal Events and Fire PRAs. The review team concluded that NextEra Energy Duane Arnold's resolution is adequate, but the F&O remains open pending update of associated documentation.

The basis of and process used for selecting pre-initiator human error probability (HEP) events for inclusion in the PRA is described in Section 2.3.3, Section 4.3, and Appendix J of DAEC's PRA Human Reliability Analysis Notebook. Pre-initiator dependencies that can be induced by human interaction were investigated from three perspectives:

1. Unique features of Duane Arnold that may deserve special attention based on a review of plant PRA systems
2. Duane Arnold and other BWR experience
3. Human interactions in the pre-initiator evaluation in previous PRAs (based on the analyst's experience)

Key types of dependencies that are considered most important in the modeling process are:

- Dependencies that would affect multiple trains such as errors that would defeat multiple trains of DC power and inhibit recovery or fail multiple components through common cause maintenance or testing errors
- Dependencies that can cause an initiating event and inhibit effective response
- Latent dependencies that defeat multiple channels of instrumentation, and therefore lend to the inability of effective post-initiator operator action

System design is a consideration for which pre-initiator HEP events are selected for use at the train level or system level. For example, the RHR Service Water system contains four pumps, two in each of two divisions. The system has manual valves, which if left in the closed position following performance of maintenance, would prevent the use of both pumps in the respective division. Pre-initiator HEP events are therefore selected for use at both the pump leg level and the division level. The RHR system on the other hand, has valves that are either re-aligned automatically on demand, or are locked-open with position indication in the control room. There is no basis for including a division level pre-initiator HEP event, only events for individual pump legs.

Application of pre-initiator HEP events at the system level is more common for calibration errors than for system restoration errors. Example include terms for failure of all ECCS initiation RPV level instruments, failure of all ECCS initiation drywell pressure instruments, and failure of in LPCI Loop Selection instruments which detect the broken loop.

Based on the quality of DAECs documented process for selecting pre-initiator human error probability (HEP) events for inclusion in the PRA, the apparent deficiency regarding "corresponding system-level dependent HFEs" is judged to be adequately addressed in the PRA Human Reliability Analysis Notebook. Sensitivity analyses deemed not necessary.

APLA-RAI-04

The NRC staff identified during its review of the licensee's source material regarding the RI-ISI consequence evaluation (Table C-2), that several items exhibited in the matrix required additional information, including:

APLA-RAI-04(a) For Item No. 49, the licensee states, in part:

> Given that Level 2 results are often the determining parameter for assignment of consequence category in the RI-ISI evaluation, this item is judged to have little influence on its results.

NRC Staff Request:

Provide the NRC staff with information regarding the stated term "little influence" and justify why any such influence would not affect the risk metrics appropriate for this application in any detrimental way.

DAEC Response:
The statement of estimated RI-ISI impact provided for this item is insufficient since any deficiency in the Level 2 PRA would indeed potentially impact RI-ISI results. The stated concern of Item 49 relates to supporting requirement LE-A4 of the ASME PRA Standard, which requires that dependencies between the Level 1 and Level 2 PRA are properly treated. The concern however, is not based on any open peer review finding, but is rather an observation made by a PRA update team member as a potential future enhancement to the Level 2 PRA. Unfortunately, no examples were provided in the observation text which would provide guidance in its resolution.

SR LE-A4 was assessed as met in the 2007 full-scope peer review. Following is the text of the peer review team's assessment.

> 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.

Section 3.6.1 of the DAEC Level 2 PRA Notebook lists assumptions used in the Level 2 PRA analysis. An assumption regarding the Level 1 to Level 2 interface states:

> Generally, the treatment of functional recovery is explicitly treated in the Level 1 analysis. The Level 2 model considers recovery of functions that primarily affect the ability of the operator to maintain RPV coolant inventory (e.g., high pressure and low pressure injection systems). These recoveries are assigned very high failure probabilities reflective of the time available and the severe accident progression (i.e., in effect, these recoveries are place holders in the analysis to illustrate recovery potential).

The Level 1 to Level 2 interface is further described in Section 4.2.2.2.

> A vital task to the accurate quantitative assessment of containment capability is ensuring that the interface and dependencies between the Level 1 PRA evaluation and the containment evaluation are precisely defined. This is assured by employing two approaches: (1) a unique containment evaluation for each type of core damage accident end state; and (2) transfer of accident sequence logic and systems fault tree logic into the Level 2 CETs. Such a coupled evaluation allows the following types of information to be accurately transferred from the Level

1 study to the containment evaluation and accounted for explicitly in the Level 2 assessment:

- Front line and support system availability
- Reactor coolant system parameters
- Time available for mitigative actions and functional recoveries
- Reactor power level, and
- Containment status

Since uncertainty associated with recovery of RPV injection systems following onset of core damage is treated conservatively--through use of high failure probabilities--in the Level 2 PRA, Item 49 of Table C-2 is judged to have no adverse impact on the RI-ISI application.

APLA-RAI-04(b) For Item No. 58, there is no information provided in either the Status or Estimated RI-ISI Impact columns of the matrix.

NRC Staff Request:

Provide this information for Item No. 58, or justify to the NRC staff why there is no information.

DAEC Response:
The status of Item 58 was inadvertently omitted from Table C-2. Item 58 is to modify logic under the RCIC test line failure gate to maintain consistency with the change made in Item #57. The desired fault tree logic was confirmed to be incorporated in to the Revision 7 PRA. Since the change has been incorporated into the PRA model version used for the RI-ISI application, completing the Estimated RI-ISI Impact column for this item in Table C-2 is not necessary.

APLA-RAI-04(c) For Item No. 59, the licensee states, in part:

> The subject gate [RCIC-07-01] is a very low contributor to RCIC system failure. Therefore this item has essentially no influence on the RI-ISI application.

NRC Staff Request:

Provide the NRC staff with information regarding any effect from the retention of gate RCIC-07- 01 that could lead to an underestimate of any of the applicable risk metrics for this application.

DAEC Response:
Water hammer in steam supply drain lines is included as a potential failure mode for both the RCIC and HPCI systems in the Revision 7 PRA. The risk significance of this failure mode is very small since one of two normally open isolation valves must fail to

remain open to establish conditions for water hammer. A probability of 0.01 is set--per engineering judgment--as the conditional probability that steam supply drain pot overfill causes water hammer. Control room operators are also provided an opportunity to respond to water buildup in the RCIC or HPCI drain due to availability of control room annunciators. The failure probability of operators to respond to the respective RCIC or HPCI annunciator is conservatively estimated to be 0.01. When these probability values are multiplied by a factor of ten for both RCIC and HPCI drain lines, CDF increases from its baseline value of 2.979E-06 per year to 3.004E-06 per year. The difference is only 2.5E-08 per year. This sensitivity case demonstrates that the risk significance of the water hammer failure mode is very low compared to other failure modes for RCIC and HPCI. As such, including this failure mode for RCIC and HPCI in the PRA is unlikely to lead to the underestimation of any applicable risk metric for the RI-ISI application.

APLA-RAI-04(d) For Item No. 63, the licensee states, in part:

> Modeling of water hammer is very low contributor to HPCI and RCIC system failure.

NRC Staff Request:

Provide the NRC staff with information regarding the basis for the statement above (e.g., what are the failure likelihoods for HPCI and RCIC currently modeled and how do they compare with the likely value for water hammer).

DAEC Response:

Water hammer in steam supply drain lines is included as a potential failure mode for both the RCIC and HPCI systems in the Revision 7 PRA. The risk significance of this failure mode is very small since one of two normally open isolation valves must fail to remain open to establish conditions for water hammer. A probability of 0.01 is set--per engineering judgment--as the conditional probability that steam supply drain pot overfill causes water hammer. Control room operators are also provided an opportunity to respond to water buildup in the RCIC or HPCI drain due to availability of control room annunciators. The failure probability of operators to respond to the respective RCIC or HPCI annunciator is conservatively estimated to be 0.01. When these probability values are multiplied by a factor of ten for both RCIC and HPCI drain lines, CDF increases from its baseline value of 2.979E-06 per year to 3.004E-06 per year. The difference is only 2.5E-08 per year. This sensitivity case demonstrates that the risk significance of the water hammer failure mode is very low compared to other failure modes for RCIC and HPCI. As such, including this failure mode for RCIC and HPCI in the PRA is unlikely to lead to the underestimation of any applicable risk metric for the RI-ISI application.

APLA-RAI-04(e) For Item No. 64, the licensee states, in part:

> The current treatment of human actions to manage water level following plant trips, including ATWS, is believed to be overly conservative and

would benefit from refinement. The current model is conservative for the RI-ISI application with respect to this time.

NRC Staff Request:

Provide the NRC staff with information regarding any potential dependencies among these separate events, such that the current combined failure probability may be under-estimated if these dependencies are not modeled. If so, provide a quantitative evaluation (e.g., a sensitivity evaluation) addressing any potential dependencies.

DAEC Response:
The subject human actions are represented by the following basic events.

DFEED-CNOPVEL-TTHE-- Operator Fails to Control Feedwater Following Scram

DFEED-CNOPVEL-TTHE-- Operator Fails to Initiate Feedwater given Feedwater Pump Trip

The first action is to take control of feedwater immediately following a plant scram to prevent trip of the feedwater pumps on high RPV water level. The second action is to restart feedwater pumps following a plant scram given they have tripped. Since these human actions always appear together in cutsets, it is proposed in Item 64 that they be combined into a single basic event as a means of simplifying quantification of the model.

The events are included in the HRA dependency analysis performed as part of the model update process. Basic event DFEED-CNOPVEL-TTHE-- has a failure probability of 1.68E-03 while basic event DFEED-CNOPSTRT--HE-- has a failure probability of 2.07E-03. The HRA combination basic event for these two human actions alone is assigned a failure probability of 8.72E-05. This is a factor of approximately 25 greater than the multiplied independent failure probabilities.

By inspection of cutsets generated for the RI-ISI application, sensitivity of potential inadequacy of the dependency calculation for human actions DFEED-CNOPVEL-TTHE-- and DFEED-CNOPVEL-TTHE-- is found to be low.

The human actions for managing RPV level after a scram and subsequent restarting of feedwater pumps in the event they trip are almost always found in cutsets together with other human actions. The HRA combination event in these cutsets is generally set to a floor value that is several orders of magnitude greater than the product of the independent human action probabilities. Increasing the dependency between the two feedwater related actions is not expected to cause the value of these combination events to increase above their floor values; therefore, calculation of CCDP and CLERP for the RI-ISI application would not be impacted.

APLA-RAI-05

The NRC staff's review of the licensee's source material regarding plant modifications assessed as potentially having a PRA impact (Table C-3), that EC No. 156110, "Installation of Well Water Isolation Valve V44-0509," requires additional information:

For EC No. 156110, the licensee states, in part:

> This is acceptable based on low safety significance of well water system isolation valves.

NRC Staff Request:

Provide the NRC staff with information regarding if the use of the above stated term "low safety significance" includes "low risk importance" such that the exclusion remains justified for this application.

DAEC Response:
Use of the term "low safety significance" in the PRA disposition of EC Number 156110 in Table C-3 can be replaced with "low risk importance" for better understanding. DAEC's Well Water system is credited in the PRA as a cooling water supply for CRD room coolers, as a backup cooling water supply for control building chillers, and as an alternate supply of makeup water to the circulating water pit. The risk importance of individual valves in the system is low due to the relatively high level of redundancy provided by its four pumps.

3.0 REFERENCES

1. Letter, Curtland (NextEra) to U.S. NRC, "Fifth Inservice Inspection Interval Program Plan, Relief Request No. RR-05," dated November 16, 2017 (ML17325B215).
2. Electronic Communication, Final RAI for Duane Arnold (DAEC) Re: 5th – 10 year ISI (RR-05), dated March 28, 2018.