NEXTera ENERGY
DUANE ARNOLD

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Report of Facility Changes, Tests and Experiments, Fire Plan Changes, and Commitment Changes

In accordance with the requirements of 10 CFR Section 50.59(d)(2), NextEra™ Energy Duane Arnold, LLC (hereafter, NextEra Energy Duane Arnold) hereby submits the subject report covering the time period from June 1, 2009 through May 1, 2011. There were no tests or experiments during this time period that require reporting. A summary of changes to the Duane Arnold Energy Center Fire Plan during this same time period is included in Enclosure 2. Enclosure 3 contains a description of one commitment change made during this period that requires reporting per the Nuclear Energy Institute's "Guidelines for Managing NRC Commitment Changes," dated July 1999.

This letter makes no new commitments. Should you have any questions regarding this matter, please contact Licensing Manager, Steve Catron at (319) 851-7234.

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Enclosures (3)

ENCLOSURE 1

DESCRIPTION OF CHANGES

This section contains brief descriptions of plant design changes and procedure changes completed during the period of June 1, 2009 through May 1, 2011, and a summary of the evaluations for the changes, pursuant to the requirements of 10 CFR Section 50.59(d)(2).

06-002 ECP 1748- 'A' Emergency Diesel Generator Governor Replacement

Description and Basis of Change

This modification involved replacement of the 'A' Emergency Diesel Generator (EDG) governor and automation of certain testing functions of the EDG. This change was performed to replace obsolete equipment, thereby improving the EDG reliability, and to minimize unavailability by automating certain functions during slow start testing.

Evaluation Summary

The EDG is not an initiator of any accident and therefore does not increase the likelihood of a previously evaluated accident or create the possibility of a different type of accident. Although additional equipment was installed under this change, the failure of any of this additional equipment did not result in more than a minimal increase in the likelihood of a malfunction of equipment important to safety because, in general, the equipment replaced manual operator actions with automatic actions and the new components are at least as reliable as the previous components, which were obsolete and would have been difficult to maintain in the future. This modification did not change the consequences of an EDG failure. Since the likelihood of a malfunction of the EDG was not increased by this change, there was no increase in the consequences of any accident. The change did not create the possibility of a malfunction with a different result because the only malfunction that could occur is a failure of the EDG which was already evaluated. No common cause failures were introduced by this change. No Design Basis Limit for a Fission Product Barrier (DBLFPB) was affected by this change because EDG reliability and availability to support front line safety systems that directly protect the DBLFPBs were not degraded by this change. The change did not involve any method of evaluation. Based upon this evaluation, prior NRC approval for this change was not required.

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Notice of Withdrawal of 5059Eval022686
Operation At 1912 MWth With Less Than 5% Excess Feedwater Flow Margin

Description of Change and Basis for Withdrawal

Evaluation 5059Eval022686 was previously reported in NextEra Energy Duane Arnold's "Report of Facility Changes, Tests and Experiments, Fire Plan Changes, and Commitment Changes," dated August 27, 2009. Evaluation 5059Eval022686 allowed operation of the plant up to 1912 MWth with less than 5% excess feedwater flow margin previously reserved for response to reactor steady state water level transients. Operation of the plant with less than 5% excess feedwater margin never occurred. Therefore, Evaluation 5059Eval022686 was not implemented and should be withdrawn.

5059Eval033361 Revision to Evaluation 5059Eval025710 – ECP-1761
Control Room Fire Damper Replacement

Description and Basis of Change

This modification replaced the existing fire dampers 1V-FD-033 and 1V-FD-306 each with an assembly containing a fire damper and a bubble tight smoke damper. Installation of these dampers addressed potential smoke intrusion into the Control Room due to a fire in either Essential Switchgear Room.

Evaluation Summary

This activity did not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated. The damper assembly is designed and constructed to prevent the spread of fire and combustion products between fire zones.

There was not a more than minimal increase in the likelihood of occurrence of a malfunction for the following reasons;

1) Inadvertent closure of the damper does not impact the overall operation of the Control Building ventilation system. The effect of an inadvertent closure of the damper on Essential Switchgear Room temperatures is bounded by already established compensatory actions for elevated Essential Switchgear Room temperatures,

2) Failure of the normal 120VAC power source or battery charger will not adversely impact damper operation. The backup battery will maintain damper control power for a minimum of 4 hours.

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Because installation of the dampers did not impact equipment used to mitigate the consequences of an accident, there was no change to the consequences of an accident or consequences of malfunction of an SSC important to safety as described in the UFSAR. All components utilized in this modification are similar in design and construction to components utilized throughout the plant. Inadvertent closure of the damper is bounded by already established compensatory actions for elevated Essential Switchgear Room temperatures.

This modification will not create the possibility for an accident of a different type than previously evaluated. The accident evaluated is an area fire resulting in the loss of the Control Room ventilation system. This modification adds additional components that are similar in design and construction to components utilized throughout the plant. Failure of any new component would not introduce a different type of accident than previously evaluated. Failure of any new component would not introduce a malfunction with a different result than previously evaluated.

This modification did not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. This modification did not involve a change to any method of evaluation. No activity requiring prior NRC approval was identified by this evaluation.

50.59 Evaluation #9131 EC156104 – Steam Dryer Tie Bar Replacement

Description and Basis of Change

During Refueling Outage 21, an examination of the Steam Dryer Tie Bar #4 revealed “crack-like” indications on the middle weld to the center baffle plate. The indication was shown to be 360º around. There are three tie bars (TB-3, TB-4 and TB-7) that connect the tops of the center steam dryer banks. Although no indications were noticed on TB-3 and TB-7 tie bars, Engineering Change EC156104 replaced all three tie bars.

Evaluation Summary

The new tie bars are of an improved design and material and have been shown in Westinghouse document “Duane Arnold Steam Dryer Tie Bar Replacement: Structural Analysis,” to be stronger and subject to lower stresses than the existing tie bars and will have no significant effect on the seismic response of the overall steam dryer. The method used by Westinghouse to evaluate the new tie bars was a comparative analysis with the existing design. The comparative

analysis is allowed for a steam dryer repair classified as Category A in EPRI document BWRVIP-181-A, "BWR Vessel and Internals Project." BWRVIP-181-A has been accepted by the NRC. This comparative analysis differs from the method used to evaluate the original tie bars described in the UFSAR.

50.59 Evaluation 9131 evaluates use of this methodology to determine if prior NRC approval is needed for this change in methodology. The design functions and method of performing the design functions of the steam dryer are not affected by this modification. As such, only 10 CFR 50.59 (c)(2) criterion (viii) must be addressed for this change.

The proposed change does not result in a departure from a method of evaluation described in the DAEC UFSAR used in establishing the design bases of the reactor vessel internals. This conclusion is based on the fact that an NRC approved alternative method is being used to evaluate the steam dryer tie bar repair. The comparative method is allowed per EPRI document BWRVIP-181-A, "BWR Vessel and Internals Project." BWRVIP-181-A has been accepted by the NRC in an SER dated January 11, 2010. This EPRI document is applicable to this type of repair to the DAEC steam dryer. Based on the above, this modification does not constitute a departure from a method of evaluation as described in the UFSAR and additional NRC approval was not required for this repair.

50. 59 Evaluation #9238 EC249768 Manual Control of 'B' Feedwater Regulating Valve

Description and Basis of Change

Due to a failure in the control system for CV1621, 'B' Feedwater Regulating Valve (BFRV), this valve went to the full open position. Manual control of the valve was taken, but the valve did not respond. EC249768 installed a temporary modification to take manual control of the valve using air regulators to control air to the top and bottom of the actuator. This temporary modification allowed control of the valve position so that BFRV could be isolated during repair and testing.

Evaluation Summary

UFSAR Section 15.1.1.1 discusses the Feedwater Maximum Demand Transient. When the BFRV failed, this was a similar, but less severe transient. The 'A' Feedwater Regulating Valve remained under automatic control. UFSAR Section 7.7.1 specifies that normal plant controls are provided for manual control. Therefore, taking local manual control of the BFRV did not result in more than a minimal increase in the frequency of occurrence of this transient.

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UFSAR Section 15.1.7.3 discusses Loss of Feedwater Flow. Taking local manual control of the BFRV did not result in more than a minimal increase in the frequency of occurrence of this transient since administrative controls were provided to ensure the valve was operated properly. The frequency of occurrence of a Loss of Feedwater Flow remained unchanged.

UFSAR Section 15.1.7.4 discusses a Trip of One Feedwater Pump. This temporary modification did not increase the probability of a Feedwater Pump trip.

The following UFSAR transients state that the Feedwater Control system reacts to the transient by increasing or reducing feedwater flow:

- UFSAR Section 15.1.1.3 - Inadvertent HPCI Actuation
- UFSAR Section 15.1.2.3.1 - Closure of One Main Steam Line Isolation Valve (1 MSIV) - High Power
- UFSAR Section 15.1.3.2 Trip Of One Recirculation Pump
- UFSAR Section 15.1.5.1 Startup Of An Idle Recirculation Pump
- UFSAR Section 15.1.5.3 Recirculation Flow Controller Failure - Slow Flow Runout

The Feedwater Control system is not an initiator for these transients. Since the 'A' Feedwater Regulating Valve remained in automatic control, taking local manual control of the BFRV did not result in more than a minimal increase in the frequency of occurrence of these transients.

The Anticipated Transients Without Scram (ATWS) analysis is discussed in UFSAR Section 15.3.1. This temporary modification did not prevent the required actions of this EOP. The Feedwater Control system is not an initiator for this event.

UFSAR Section 15.2.3 discusses a Recirculation Pump Seizure Accident. The Feedwater Control system is not an initiator for this accident.

UFSAR Section 7.7.1 specifies that normal plant controls are already provided for manual control. Therefore, providing another manual method of control of the BFRV did not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Transients, accidents, and malfunctions associated with the BFRV have no radiological consequences associated with them. The consequences of the transients and accidents associated with this temporary modification were bounded by the evaluation of feedwater system failures already existing in UFSAR Chapter 15. The consequences of malfunctions of the Feedwater

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Control System and Feedwater System remain the same with this temporary modification. Therefore, this temporary modification did not result in more than a minimal increase in the consequences of an accident or malfunction previously evaluated in the UFSAR.

No new accident initiators were introduced by this modification. Therefore, this change did not create a possibility for an accident of a different type than any previously evaluated in the UFSAR.

An accident or transient involving the installed manual controls would be the same for the additional manual control proposed for the BFRV. Therefore, this change did not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

The transients involving Feedwater Control System failure do not involve any challenges to fission product barriers. Therefore, this change did not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

This change did not involve any changes to methodologies. Based on the above, NRC approval was not required for this temporary modification.

50. 59 Evaluation #9254 Flow Accelerated Corrosion Acceptance Criteria

Description and Basis of Change

This 50.59 Evaluation addressed the use of ASME Section XI Code Case N-597 and the method used to determine the stresses in safety-related buried piping for Flow Accelerated Corrosion inspections.

Evaluation Summary

The Flow Accelerated Corrosion (FAC) monitoring program minimizes challenges to plant safety that could occur in the event of an unexpected pipe failure due to excessive pipe wall thinning. The FAC monitoring program provides a means for systematic inspection and evaluation of FAC.

The FAC monitoring program allows for engineering evaluations to be performed in accordance with the original code of construction with the conditions approved by the NRC in Regulatory Guide 1.147 for above ground piping systems. Buried piping systems were originally designed and constructed to code USAS B31.1.0.

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The methodology used to determine stresses in buried piping is not described in the Duane Arnold UFSAR.

Regulatory Guide 1.147 documents NRC approval of ASME N-597 Code Case. Code Case N-597-2 is listed in Regulatory Guide 1.147 as a "Conditionally Acceptable Section XI Code Case" and the NRC has specified additional requirements and limitations for the acceptability of this code case.

Since this 50.59 Evaluation addressed the use of ASME Section XI Code Case N-597 and the method used to determine the stresses in safety-related buried piping for Flow Accelerated Corrosion inspections, only 10 CFR 50.59 (c)(2) criterion (viii) must be addressed for this change.

When used, ASME Code Case N-597 amends the original code of construction for the piping evaluated. The use of ASME Code Case N-597 is approved by the NRC for the intended application. DAEC ensures compliance with the five NRC conditions of acceptability listed in Regulatory Guide 1.147 as follows:

- DAEC FAC Program ensures that the inspection requirements, the method of predicting the rate of wall thickness loss, and the value of predicted remaining wall thickness are in accordance with EPRI NSAR-202L-R2.
- DAEC FAC Program establishes inspection requirements that ensure that components affected by FAC will be repaired or replaced prior to the value of t_p reaching the allowable minimum wall thickness t_{min}.
- Minimum wall thickness evaluations are not performed for Class 1 piping. Moreover, each evaluation lists the requirement described above, to ensure that code case N597-2 will not be used for Class 1 piping.
- After measurements are made, the DAEC FAC Program determines the inspection frequency requirements that ensure the components affected by FAC are repaired or replaced prior to the value of t_p reaching the allowable minimum wall thickness t_{min}.
- Every evaluation performed using code case N-597-2 will clearly state that the minimum wall thickness results are only valid if the corrosion phenomenon is determined to be Flow Accelerated Corrosion.

As the methodology being used for evaluating pipe minimum wall thickness for the above ground piping has already been approved by the NRC for the intended application, as documented in Regulatory Guide 1.147, the proposed activity does not constitute a departure from a method of evaluation described in the UFSAR.

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50.59 Evaluation 9254 also addresses using the methodologies and guidance stated in the following documents for evaluating pipe minimum wall thickness due to flow accelerated corrosion for buried seismic piping:

1. Seismic Responses of Buried Pipes and Structural Components, American Society of Civil Engineers, copyright 1983.
2. Flexibility Analysis of Buried Pipe, ASME, dated June 1978.
3. More on Flexibility Analysis on Buried Pipe, ASME, dated August 1980.

When used, the above methodologies amend the original code of construction for the piping evaluated. These methodologies use equations that have an additional term to account for pipe and soil interaction during a seismic event and differ from the equations in the original code of construction. These methodologies produce results that are conservative with respect to just applying the code equations without accounting for pipe and soil interaction. It is therefore determined that the methodologies described do not constitute a departure from a method of evaluation described in the UFSAR.

50. 59 Evaluation #9484 Revision to Evaluation 5059Eval022769 (previously identified as 50.59 Evaluation 07- 001) - Revision Of Refueling Procedures (RFPs) 110, "Reactor Pressure Vessel Disassembly" and 210, "Reactor Pressure Vessel Reassembly"

Description and Basis of Change

Evaluation 5059Eval022769 was previously reported in NextEra Energy Duane Arnold's "Report of Facility Changes, Tests and Experiments, Fire Plan Changes, and Commitment Changes," dated August 27, 2009. Evaluation 5059Eval022769 evaluated changes to refueling procedures (RFP 110 and 210) to include the use of a new tensioning system for the detensioning and tensioning of the reactor vessel studs using a reduced pass process. In addition, the changes to RFP 210 included increasing the reactor vessel stud elongation acceptance tolerance range. These changes reduced the overall length of refueling outages and the radiation exposure received by the reactor head assembly and disassembly work crew.

The additional weight of the new tensioners affects the strongback for lifting the tensioners into place for the Reactor Pressure Vessel (RPV) de-tensioning and tensioning activities. A calculation demonstrated that the strongback was adequate "as is" for the heavier lift. The strongback was load tested and inspected per ANSI-N14.6-1978 prior to lifting the additional weight of the new tensioners. In addition, a calculation demonstrated that the dropping of the tensioners on the refuel floor is acceptable for the plant. The calculation

assumes a weight of 3000 lbs for each tensioner which bounds the new tensioner weight. The changes did not affect the design basis, function, or operation of the tensioner strongback.

50.59 Evaluation 9484 is a revision of Evaluation 5059Eval022769. This revision better explains the method of evaluation used by Dominion Engineering in their analysis of reactor vessel and stud stresses as a result of the reduced pass tensioning/detensioning method for the reactor vessel head. This revision also incorporates the revised DEI Report which concludes that no changes are required in the reduced pass tensioning procedure or tensioning equipment that was evaluated in Evaluation 5059Eval022769. Accordingly, this 50.59 Evaluation revision is not related to any physical or procedural changes from that originally evaluated.

Evaluation Summary

The reactor vessel flange/stud connections are part of the reactor pressure vessel (RPV) boundary. This activity did not adversely affect the vessel pressure boundary. While the tensioning and detensioning procedure change did permit the reactor vessel studs to have a larger preload stress than had been previously permitted, the closure flange and studs were demonstrated to meet applicable ASME Code stresses and fatigue usage limits, so there was no change to the design basis of the RPV. The additional weight of the new tensioners on the strongback for lifting the tensioners over the vessel head for tensioning and de-tensioning was evaluated and it was determined the strongback was adequate "as is" for the heavier lift. The strongback was satisfactorily load tested and inspected per ANSI-N14.6-1978 prior to lifting the additional weight of the new tensioners. The changes did not increase the frequency of occurrence of an accident previously evaluated in the UFSAR, and they did not increase the likelihood of occurrence of a malfunction of any SSC important to safety previously evaluated in the UFSAR. The RPV studs have no active role during any hypothesized accident other than to seal the reactor vessel head to the reactor vessel shell. The changes meet the original design and construction standards as applicable to the RPV System. Therefore, the RPV closure flange will remain functional during all accidents considered in the UFSAR. The consequences of an accident previously evaluated in the UFSAR were not increased, and the consequences of a malfunction of a SSC important to safety previously evaluated were not increased. Implementation of the reduced pass detensioning/tensioning process and new elongation acceptance criteria creates no new credible failure modes. Implementation of the new tensioners did not create new credible failure modes. The possibility for an accident of a different type than previously evaluated in the UFSAR was not created, and the possibility for a malfunction of an SSC important to safety with a different result than any

previously evaluated in the UFSAR was not created. The DBLFPB as described in the UFSAR was not exceeded or altered. This change did not result in a departure from the method of evaluation described in the UFSAR used in establishing the design bases or in the safety analysis for the reactor vessel flange, studs or strongback. No activity requiring prior NRC approval was identified.

50. 59 Evaluation #9526 ECP 1873 – Core Modification Package for Cycle 22

This evaluation covers the upgrade of methods used as part of the cycle-specific reload licensing calculation for the Core Operating Limits Report (COLR), specifically for the thermal-hydraulic stability analysis, as described in UFSAR 15.3.4.1. These new methods were used to calculate the power-to-flow map stability monitoring regions in the COLR, as part of the Core Modification Package (ECP 1873) for Cycle 22. These new methods had two significant changes from the previously used methodology:

- Removal of the 0.15 bias (adder) to the core decay ratios calculated by the ODYSY code, which was a previous NRC condition for acceptance (NEDC-32992P-A);
- Use of the Modified Shape Function to define the boundaries of the Exclusion and Buffer Zones on the power-to-flow map, instead of the previous Generic Shape Function approved by the NRC (NEDC-32992P-A).

The NRC approved the new methods in a safety evaluation report titled, “Final Safety Evaluation for Boiling Water Reactors Owners’ Group (BWROG) Licensing Topical Report (LTR) NEDE-33213P, Applications for Stability Licensing Calculations Including Option I-D and II Long Term Solutions,” dated February 23, 2009. The NRC stated thirteen conditions of the approval of the new methods.

Since this 50.59 Evaluation addressed the use of NEDE-33213P-A, only 10 CFR 50.59 (c)(2) criterion (viii) must be addressed for this change.

As documented in GEH report, “Alternate Exclusion Region Evaluation for Duane Arnold Cycle 22,” NextEra Energy Duane Arnold complies with the required NRC conditions for approval of NEDE-33213P-A. Based on the above, this change does not constitute a departure from a method of evaluation as described in the UFSAR and additional NRC approval is not required.

ENCLOSURE 2

Fire Plan Changes

The information contained in this section identifies and briefly describes changes to the DAEC Fire Plan during the time period beginning June 1, 2009 through May 1, 2011.

Revision 57

Fire Plan Surveillance Requirement FPSR 12.1.D.1.2 requires functional testing of CO_2 thermal detectors on a 1 year frequency. The corresponding Bases Section FB 12.1.D stated the testing was performed on a semi-annual basis. The Fire Plan Bases Section FB 12.1.D was changed to reflect the annual testing frequency consistent with Fire Plan Surveillance Requirement FPSR 12.1.D.1.2.

Use of the term “fire watch” in the Fire Plan was changed to “fire patrol” to distinguish between a “fire watch” for maintenance activities (e.g., hot work) and a “fire patrol” used as a compensatory measure for an impaired fire protection feature.

Revision 58

The frequency of Fire Plan Surveillance Requirement FPSR 12.1.B.2.1 was revised from one month to one week. FPSR 12.1.B.2.1 requires starting the diesel-driven fire pump and operating for at least 30 minutes.

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COMMITMENT CHANGES

The information contained in this section identifies and briefly describes the commitment change made during the period beginning June 1, 2009 through May 1, 2011. The changes described are being reported per the Nuclear Energy Institute's "Guidelines for Managing NRC Commitment Changes," dated July 1999.

CR01621097 NextEra Energy Duane Arnold committed to implement the recommendations of Generic Letter 89-13. Recommendation II of Generic Letter 89-13 requires conducting a testing program to verify the heat transfer capability of safety related heat exchangers cooled by service water. Recommendation II also requires a final minimum testing frequency of once every 5 years.

"B" High Pressure Coolant Injection (HPCI) room cooler testing was required by March 21, 2011. Cooling coil replacement for this cooler was planned for April 2011. The commitment change delayed performance of the required testing until the new cooling coil was installed. After installation, a 4 year testing period was established.

Justification for this change was based on the 2006 as-found condition of the cooling coils, the current flow through the cooling coils and the fact that there is a redundant "B" HPCI room cooler capable of removing the design heat load.