DUANE ARNOLD

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

Report of Facility Changes, Tests and Experiments, 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 April 1, 2017 through March 31, 2019. A summary of specific facility changes and procedure changes completed during this period and a summary of the 10 CFR 50.59 evaluation of each is included in the Enclosure. There were no tests or experiments during this time period that require reporting. There were no commitment changes made during this period that require reporting per the Nuclear Energy Institute's, "Guidelines for Managing NRC Commitment Changes," dated July 1999.

This letter makes no new commitments or changes to existing commitments. Should you have any questions regarding this matter, please contact Michael Davis at 319-851-7032.

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Site Director
NextEra Energy Duane Arnold, LLC

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, DAEC, USNRC
Senior Resident Inspector, DAEC, USNRC

ENCLOSURE to NG-10-0053

DESCRIPTION OF CHANGES

This section contains brief descriptions of plant design changes and procedure changes completed during the period of April 1, 2017 through March 31, 2019, and a summary of the evaluations for the changes, pursuant to the requirements of 10 CFR Section 50.59(d)(2).

50.59 Evaluation EC282458 – River Water Supply Cable Replacement

Description and Basis of Change

The scope of this Engineering Change (EP) is to replace and reroute nine (9) cables in Division II of the River Water System (RWS). These cables run together in the Turbine Building through embedded conduit and exit to the Intake Structure. The existing cables were either removed from the conduit raceways or were spared in place. This activity was undertaken because of cable degradation in the embedded conduit in Turbine Building Base Mats.

EC 282458 installed an interim design control configuration for the D River Water Supply (RWS) Pump using the control cable from the B RWS Pump. The cable used for this interim design configuration was not able to power the green indicating light in the control room and at remote shutdown panel (RSP) 1C388 for the "D Pump Not Running" indication. The interim control configuration, without the green indicating light and annunciator for the motor starter not in operating position or control switch in pull-to-lock (PTL), was evaluated and found to be an acceptable design configuration. The existing configuration was restored when the modification installation was completed and the equipment was returned to operation.

While the interim design configuration was installed for the D RWS pump, and during final installation of the control cables with the B RWS pump operable, the annunciator for the "motor starter not in operating position or control switch in PTL" annunciator was not available for the respective pump. This annunciator is specifically described in UFSAR Section 7.5.3 and is considered a required design function. The 50.59 screening performed for this interim design configuration determined that this was an adverse effect on an UFSAR required design function which required further evaluation under the 50.59 process.

Evaluation Summary

The proposed activity removed the annunciation of an operating bypass for the B and D RWS Pumps during installation of EC282458. This annunciator notifies the operator when the B or D RWS pump trips, its supply breaker is not racked in or its control switch is in PTL. The RWS system supplies make-up water to the RHRSW/ESW wet pits. RHRSW provides the heat sink for the RHR system when operating in the containment cooling and shutdown cooling modes. ESW supplies cooling water in support of the ECCS systems and the Diesel Generators. Loss of RWS is not an accident initiator.

The elimination of this annunciator does not affect the initiation of any accident. The loss of the B or D RWS pump could eventually lead to a loss of shutdown cooling. However, eliminating this annunciator will not increase the likelihood of a failure of the B or D RWS pump. If the B or D RWS pump does trip while operating without this annunciator available, the control room operator has alternate indications, including (1) loss of the D RWS Pump red operating light or change of indication for the B RWS pump red and green indicating lights, (2) RWS flow indication and (3) RHRSW/ESW Pit Low Level alarms, which would alert him of this condition. Therefore, the proposed change does not result in a more than minimal increase in the frequency of an accident previously evaluated in the UFSAR.

The proposed change to eliminate the B or D RWS pump trip annunciator does not increase the likelihood of a failure of the B or D RWS pump. Additionally, if the B or D RWS pump does trip while operating without this annunciator, the likelihood of a failure of the RWS system to provide adequate make-up flow to the RHRSW/ESW pits is not increased more than minimally because the control room operator has several alternate indications as discussed above. These alternate indications ensure that the operator would respond properly to any failure of the B or D RWS pump and the likelihood of failure of the RWS system would not increase. Therefore, the proposed change does not result in a more than minimal increase in the likelihood of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The RWS system supports the plant response to several accidents by providing make-up water to the RHRSW/ESW pits. This RWS design function is unaffected by the proposed change. The elimination of the B or D RWS pump trip annunciator does not change the plant response to any accident such that more fuel failures would occur which would increase the radiological consequences of the accident.

Additionally, if the B or D RWS pump does trip while operating without this annunciator, the likelihood of a failure of the RWS system affecting the plant response to an accident is not increased more than minimally because the control room operator has several alternate indications of this pump trip, as previously discussed, that ensure the operator would respond properly to any failure of the B or D RWS pump. Therefore, the proposed change cannot result in any increase in the radiological consequences of an accident previously evaluated in the UFSAR.

The important to safety SSC related to this proposed change is the RWS system, specifically the B or D RWS pump. The plant response to the failure of the B or D RWS pump will not change due to the elimination of the B or D RWS pump trip annunciator. The elimination of this annunciator will not cause an increase in the amount of fuel failures due to a loss of shutdown cooling or any other event caused by a loss of the B or D RWS pump. Additionally, if the B or D RWS pump does trip while operating without this annunciator, the likelihood of a failure of the RWS system affecting the plant response to an accident is not increased more than minimally because, as previously discussed, the control room operator has several alternate indications of this pump trip. Therefore, the proposed change cannot result in any increase in the radiological consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The elimination of the B or D RWS pump trip annunciator cannot cause an accident of a different type than those already evaluated in the UFSAR. This design function only provides notification of a B or D RWS pump failure and cannot cause an accident of any type. Therefore, the proposed change does not create the possibility of an accident of a different type.

The elimination of the B or D RWS pump trip annunciator cannot change the result of a failure of the B or D RWS pump or the RWS system. The result of a failure of the RWS system would be failure to provide adequate make-up flow to the RHRSW/ESW pits. This result does not change if there is no annunciation of the trip of the B or D RWS pump. Therefore, the proposed change does not create the possibility for a malfunction of an SSC important to safety with a different result than that already evaluated in the UFSAR.

The elimination of the B or D RWS pump trip annunciator has no effect on any Design Basis Limit for Fission Product Barrier (DBLFPB). All DBLFPB limits will remain the same and since the results of all accidents will remain the same this proposed change does not result in any DBLFPB being exceeded.

This proposed change does not affect or change any method of evaluation described in the UFSAR.

50.59 Evaluation **EC287931 – Implementation of Revised Calculations for Loss Of Coolant Accident (LOCA) Dose Consequences**

Description and Basis of Change

This evaluation addresses the implementation of the following calculations:

- CAL-R00-PUP-006 Revision 2, "NUREG-0737 Item II.B.2 – Mission Dose Assessment for AEP and AST,"
- CAL-R00-PUP-007 Revision 3, "DBA LOCA Radiological Consequences Doses with Alternate Source Term," and
- CAL-R00-PUP-009 Revision 2, "DBA LOCA – Radionuclide Loading on Safety Filters."

Previous revisions of those calculations are analyses of record for the dose consequences of the LOCA analyses. The LOCA dose consequences analysis is described in UFSAR section 15.2.1, based on the methodology and results documented in the three aforementioned calculations.

The proposed activity is the implementation of the three calculations into the UFSAR. There are no plant physical changes associated with this proposed activity.

The calculations have been updated to incorporate the following changes –

(1) Modifying the bounding case RADTRAD models to correct an error identified in the RADTRAD Power's natural deposition of aerosols model which stopped deposition prematurely and limited decontamination,

(2) Modifying the RADTRAD Brockmann-Bixler input for deposition in piping to include only horizontal piping using the DAEC bounding case RADTRAD models,

(3) The post-accident torus suppression pool temperature is allowed to exceed 212 degrees F with a 10% flash fraction for ESF leakage,

(4) The maximum drywell 2-minutes post-LOCA pressure is increased from 25.45 psia to 3.61 atm (53.1 psia),

(5) The MSIV leakage rate from the drywell is increased from 50% to 57% of the design leakage rate at 24 hours post-LOCA and beyond,

(6) Including an additional 2" of concrete shielding due to the addition of the Data Acquisition Center around and on top of the TSC,

(7) Modifying the dose resulting from the reactor building shine to the control room operators associated with plant ingress/egress from 0.874 rem to 0.67 rem to be consistent with the current schedule for control room operators, and

(8) Updating the RADTRAD code from version 3.02 to version 3.03 currently accepted by the NRC for LOCA dose consequences analyses.

The first change addresses an error identified in the RADTRAD code; the second is for a previous commitment made to the NRC in case this analysis was revised; changes three through five were due to unverified assumptions in the original analyses that have now been validated; changes six and seven were included to update to current plant conditions and practices; and change eight is made as that is the current version of the RADTRAD code.

The LOCA dose consequences analysis was approved by the NRC and was subsequently implemented via License Amendment 240. The NRC safety evaluation for that license amendment does not contain any conditions or limitations. Items 1-8 listed above were included as part of the three updated calculations after Amendment 240 was approved, so they require 50.59 review.

Evaluation Summary

Proposed activities 1-7 are input values changes that only apply to the dose consequences calculated for the LOCA analysis, and have no physical impact on the plant. Proposed activity 8 is a potential change in the method of evaluation and is addressed later in this evaluation. Therefore, the proposed activities have no impact on the frequency of occurrence of previously evaluated accidents in the UFSAR.

Proposed activities 1-7 are input values changes that only apply to the dose consequences calculated for the LOCA analysis. These changes do not have any effect on the malfunction of any SSC assumed in the calculations, nor any physical impact on the plant. Proposed activity 8 is a potential change in the method of evaluation and is addressed later in this evaluation. Therefore, the proposed activities have no impact on the likelihood of occurrence of an SSC malfunction important to safety previously evaluated in the UFSAR.

Proposed activities 1-7 are input values changes that apply to the dose consequences calculated for the LOCA analysis. These changes have a direct impact on the radiological consequences for the LOCA analysis. The first change addresses an error identified in the RADTRAD code, the second is for a previous commitment made to the NRC in case this analysis was revised, changes three through five were due to unverified assumptions in the original analyses that have now been validated, and changes six and seven were included to update to current plant conditions and practices.

Table 3.1 summarizes the impact of these changes for the LOCA dose consequences analysis. The table notes the current (as noted in the table in the UFSAR page 15.2-36) and revised dose values (from updated calculations) for each of the zones analyzed. The table then notes the regulatory limit, the increase in dose due to this activity, and 10% of the margin between the regulatory limit and current value. Finally, the table notes whether the dose increase is greater than 10% of the margin between the regulatory limit and current value, and whether the revised value exceeds the regulatory limit. All values noted in the table are in 'rem' units.

Table 3.1 – LOCA (TEDE; units in rem)

	CR	**TSC**	**EAB**	**LPZ**
Current	4.152	4.397	0.250	0.601
Revised	4.094	3.976	0.247	0.626
Regulatory Limit	5.00*^	5.00*^	25.00^	25.00^
Increase	-0.058	-0.421	-0.003	0.025
10% of Margin	0.085	0.060	2.475	2.440
More than minimal?	NO	NO	NO	NO
Exceeds limit?	NO	NO	NO	NO

* R.G. 1.183 limits; ^10CFR50.67 limits

The definition of "minimal increase" found in NEI 96-07 Revision 1 is as follows: "An increase in consequences from a proposed activity is defined to be no more than minimal if the increase (1) is less than or equal to 10 percent of the difference between the current calculated dose value and the regulatory guideline value (10 CFR 100 or GDC 19, as applicable), and (2) the increased dose does not exceed the current SRP guideline value for the particular design basis event. The current calculated dose values are those documented in the most up-to-date analyses of record."

The only increase case is LPZ and is within the 10% increase in consequences. Given that definition and the results presented in Table 3.1, it is concluded that the proposed activity does not result in more than a minimal increase in the radiological consequences of the LOCA analysis. Furthermore, the proposed activity has no physical impact on the plant, thus it has no impact on any other accident or the radiological consequences of any other accident.

Proposed activity 8 is a potential change in the method of evaluation and is addressed later in this evaluation.

Therefore, the proposed activities do not result in more than a minimal increase in the radiological consequence of an accident previously evaluated in the UFSAR.

Proposed activities 1-7 are input values changes that only apply to the dose consequences calculated for the LOCA analysis. These changes have no impact on the dose consequences of any other event, including SSC malfunctions. No new equipment or systems are introduced, so no new malfunctions are created. Proposed activity 8 is a potential change in the method of evaluation and is addressed later in this evaluation. Therefore, the proposed activities have no impact on the radiological consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Proposed activities 1-7 are input values changes that only apply to the dose consequences calculated for the LOCA analysis. As these changes do not include new plant equipment or components, no new or different accidents are created. Proposed activity 8 is a potential change in the method of evaluation and is addressed later in this evaluation. Therefore, the proposed activities do not create a possibility for accidents of a different type than any previously evaluated in the UFSAR.

Proposed activities 1-7 are input values changes that only apply to the dose consequences calculated for the LOCA analysis. These changes have no physical impact on the plant or effect on any malfunction of an SSC important to safety. Proposed activity 8 is a potential change in the method of evaluation and is addressed later in this evaluation. Therefore, the proposed activities do not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

Proposed activities 1-7 are input values changes that only apply to the dose consequences calculated for the LOCA analysis. These changes have no physical impact on the plant.

Regardless of the input values used for this analysis, core operation will remain compliant with required Technical Specification power distribution limits (CPR, LHGR, and MAPLHGR) and all other design basis limits for fission product barrier (DBLFPBs). Since the proposed activities have no physical impact on the plant, there is no impact to the reactor coolant system or containment. Proposed activity 8 is a potential change in the method of evaluation and is addressed later in this evaluation. Therefore, the proposed activities have no impact on current DBLFPBs or exceeding those values described in the UFSAR during operation.

Proposed activities 1-7 (all except for the RADTRAD version change) are input values changes that only apply to the dose consequences calculated for the LOCA analysis. From NEI 96-07 Revision 1 page 18, "If a methodology permits the licensee to establish the value of an input parameter on the basis of plant-specific considerations, then that value is an input to the methodology, not part of the methodology." Therefore, these proposed activities do not impact the method of evaluation.

Proposed activity 8 (use of RADTRAD Version 3.03) is a potential change in the method of evaluation. NEI 96-07 Revision 1 page 14 provides the definition as follows: "Departure from a method of evaluation described in the FSAR (as updated) means (i) changing any of the elements of the method described in the FSAR (as updated) unless the results of the analysis are conservative or essentially the same; or (ii) changing from a method described in the FSAR to another method unless that method has been approved by NRC for the intended application."

The RADTRAD code version change is in compliance with the statement "unless that method has been approved by the NRC for the intended application," thus it is not a departure from a method of evaluation described in the FSAR (as updated). The RADTRAD Version 3.03 code has been approved by the NRC for the intended application, which is LOCA dose consequences analysis at a similar BWR plant. The following two letters document NRC approval of applications that used RADTRAD Version 3.03 for LOCA dose consequences analyses:

- Letter from Richard V. Guzman (NRC) to Keith J. Polson (Nine Mile Point), "Nine Mile Point Nuclear Station, Unit No. 2 – Issuance of Amendment re: Implementation of Alternate Radiological Source Term (TAC No. MD5758)," May 29, 2008.
- Letter from Christopher Gratton (NRC) to Michael J. Pacilio (Exelon Nuclear), "LaSalle County Station, Units 1 and 2 – Issuance of Amendment re: Application of Alternative Source Term (TAC Nos. ME0068 and ME0069)," September 6, 2010.

Therefore, it is concluded that the proposed activity is not a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses.