NEXTera ENERGY DUANE ARNOLD

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10 CFR 50.59(d)(2)

U. S. Nuclear Regulatory Commission
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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 May 1, 2015 through March 31, 2017. A summary of specific facility changes and procedure changes completed during this time period and a summary of the 10 CFR 50.59 evaluation of each is included in Enclosure 1. There were no tests or experiments during this time period that require reporting. Enclosure 2 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 or changes to existing commitments. Should you have any questions regarding this matter, please contact Michael Davis at 319-851-7032.

for

Dean Curtland
Site Director
NextEra Energy Duane Arnold, LLC

Enclosures

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

ENCLOSURE 1 to NG-17-0095

DESCRIPTION OF CHANGES

This section contains brief descriptions of plant design changes and procedure changes completed during the period of May 1, 2015 through March 31, 2017, and a summary of the evaluations for the changes, pursuant to the requirements of 10 CFR Section 50.59(d)(2).

50.59 Evaluation EC281703 - Chromalox Replacement Project

Description and Basis of Change

The Primary Containment Isolation System (PCIS) - Main Steam Line Steam Leak Detection - (MSL-SLD) temperature indicating alarm units (TIS4445, TIS4446, TIS4479, and TIS4480) are obsolete and at the end of the useful life. The existing units are currently Chromalox model 3421 Process Alarm Units (PAU). The dependability of the units has decreased with recent failures attributed to instrument drift and false trips.

To resolve the obsolescence and dependability issue, the Chromalox PAUs will be replaced. The two "A2" logic units (TIS4445, TIS4479) and the two "B2" (TIS4446, TIS4480) will be replaced with Yokogawa model DX1006N recorders. The two "A1" logic units (TIS4443, TIS4477) and the two "B1" (TIS4444, TIS4478) were replaced with Yokogawa DX1006N recorders under a previous modification package, EC 277929.

The Chromalox model 3421 PAU, and the Yokogawa model DX1006N recorder are digital instrumentation, therefore, this modification is considered a digital to digital modification. The acceptability of this digital instrumentation for the PCIS-MSL-SLD logic was previously evaluated via 10CFR50.59 Evaluation 99-011.

EC 277929 also installed Yokogawa DX1006N recorders into the PCIS-MSL-SLD logic. Installation of Yokogawa DX1006N recorders into the PCIS-MSL-SLD system was previously reviewed via the 10CFR50.59 screening process for EC 277929 as well as this EC 281703. Therefore, this 10CFR 50.59 evaluation will address the issue created by installing the same model equipment into both logic trains of the PCIS-MSL-SLD. Namely the Common-Cause Failure potential created by using Digital Equipment.

The Yokogawa model DX1006N recorder and Chromalox PAUs are individual digital components. For this application, they are standalone instruments and are not networked with any other digital components. Therefore this digital modification is not complex and can be considered simple.

A digital evaluation was performed in accordance with EPRI manual TR-102348 Rev 1 "EPRI Guidelines for Licensing Digital Upgrades".

Based on the completed Digital Evaluation and the following justifications, using Yokogawa DX1006N recorders on both logic trains of the PCIS-MSL-SLD is acceptable and the Common-Cause risk probability is low.

(1) The Yokogawa series DX1000N recorder is a highly reliable instrument. The instrument is used extensively at DAEC and throughout the industry. A review of the Operating Experience (OE) database shows that the vast majority of the issues with the recorder are display failures. The display function is not required for this application. Very few logic type issues have been identified for the recorder.
(2) The Yokogawa series DX1000N recorder has gone through extensive qualification testing. This testing includes seismic, EMI/RFI, and software V&V testing. All testing documents have been reviewed and are acceptable for this application.
(3) The MSL-SLD high temperature logic is considered a backup to the MSL-SLD High Flow logic. The Updated Final Safety Analysis Report (UFSAR) Chapter 15 uses the MSL-SLD High Flow logic in the accident analysis. Both the high temperature logic and the high flow logic generate isolation signals.
(4) Several alternate means of determining if there is a steam leak in the Steam Tunnel and/or Heater Bay exist, including the Main Steam Line-SLD high flow logic and the SLD logic for High Pressure Coolant Injection (HPCI), Reactor Core Isolations Cooling (RCIC) and Reactor Water Cleanup (RWCU).
(5) Each of the MSL-SLD instrumentation are standalone instruments and operate as individual components. They do not interface with any other digital components and are not networked.
(6) The recorders are password protected and employ cyber-security protection features to prevent unauthorized tampering with the components.
(7) Operations and Maintenance personnel are very familiar with the operation of these recorders. They have had extensive training on the Yokogawa recorders and thousands of hours of operational experience with the recorders.

<u>Evaluation Summary</u>

EC 281703 is replacing the instrumentation for the Main Steam Line - Steam Leak Detection High Temperature (Steam Tunnel and Heater Bay) input to the Primary Containment Isolation System logic. The MSL-SLD function is not an initiator of any accident described in the plant UFSAR. It is used to mitigate the effects of an accident or transient involving a Main Steam Line steam leak once the accident or transient has initiated. Therefore, the frequency of occurrence of an accident previously evaluated in the UFSAR will not change.

EC 281703 is replacing a Process Alarm Unit with a recorder. Both units use input processing, A/D converters, and output relays to process the signals from the temperature elements. There is not a more than minimal increase in the likelihood of occurrence of a malfunction for the following reasons:

(1) As described earlier, the replacement equipment will be qualified to meet seismic, environmental, EMI/RFI and software V&V requirements for this application. The new

equipment will also be installed in accordance with approved specifications. These qualifications will ensure the equipment will continue to operate during all required conditions.

(2) Even though this modification is performing a 1-for-1 replacement, equipment reliability is expected to increase. The existing Chromalox PAUs have had several issues that ultimately resulted in them being considered unreliable. Operating Experience with the Yokogawa recorders at NextEra Energy Duane Arnold and throughout the industry has been very favorable with few documented failures.

(3) The use of Yokogawa recorders in both trains of the PCIS-MSL-SLD temperature monitoring logic does introduce a new failure mode called Common-Cause failure, however as shown in the evaluation section, the risk of a Common-Cause failure occurring is very low. In addition, the effects caused by a Common-Cause failure are compensated for by using multiple trains (IE: PCIS-MSL-SLD High Temperature subsystems A1, A2, B1, and B2) and multiple systems (IE: PCIS-MSL-SLD High Flow system and HPCI/RCIC/RWCU-SLD system) to monitor for and isolate steam leaks.

The equipment specifications and time response of the new instrumentation is as good as or better than the existing equipment. The Technical Specifications Allowable Value of ≤ 205.1°F for the Steam Leak Detection – high temperature (Steam Tunnel and Turbine Building) logic will be maintained. This value was verified to be acceptable by engineering calculation. By verifying the replacement instrumentation meets the specifications and time response requirements for this application, the existing analysis for all transients/accidents involving Loss-Of-Coolant is validated. Therefore, there is no change to the consequences of an accident as described in the UFSAR.

The worst case accident applicable to this modification is a double ended guillotine break of a MSL outside containment. All other steam leaks outside containment are bounded by this accident. The MSL-SLD high temperature logic is for detecting small steam leaks which are bounded by the worst case accident. Therefore, the worst case radiological consequence for the accident analyzed in the UFSAR is unaffected.

The MSL-SLD high temperature logic is not the sole means of detecting small steam leaks outside of containment in the vicinity of the MSLs. Other systems that are capable of detecting steam leaks include the HPCI/RCIC/RWCU steam leak detection system and the Turbine Building radiation monitoring system. The diversity in systems capable of detecting small steam leaks in conjunction with the low probability of Common-Cause failure of the MSL-SLD high temperature logic minimizes the risk associated with the malfunction of the MSL-SLD high temperature instrumentation.

MSL-SLD High Temperature logic consist of two subsystems, "A" and "B", and each trip subsystem is made up of two trip channels, "A1", "A2", "B1", and "B2" and each channel contains two instruments (Steam Tunnel and Turbine Building) for a total of eight units. Each temperature monitor acts as an independent trip unit with no interconnections between other temperature monitors or other systems. PCIS-MSL-SLD is configured as a 1-of-2-twice logic. This means that a failure in a single temperature monitor will not prevent or generate a trip in a separate monitor (channel). Also as discussed previously, with the low probability of a Common-Cause failure and the fact that MSL-SLD temperature monitoring is considered a backup to the MSL flow monitoring instrumentation. The PCIS-MSL-SLD logic is considered single failure proof in that a single failure will not prevent operation of the PCIS in response to a Loss-Of-Coolant accident or transient

Based on the above, a malfunction of an SSC (specifically the MSL-SLD high temperature instrumentation) does not result in a more than minimal increase in radiological consequences.

EC281703 does not introduce the possibility of a new accident because PCIS-MSL-SLD is not an initiator of any accident and no new failure modes are identified that are not bounded by previously evaluated modes.

EC 281703 does not introduce the possibility for a malfunction of the PCIS-MSL-SLD with a different result because the activity does not introduce a failure mode that is not bounded by those described in the plant UFSAR for the PCIS system. The use of Yokogawa recorders in both subsystems of the PCIS-MSL-SLD does introduce a potential new failure mechanism (Common-Cause failure), however, as previously described these failures are bounded by the existing failure analysis for the PCIS-MSL-SLD logic and no new unanalyzed failures modes are created.

As previously described, the function of PCIS instrumentation (including MSL-SLD) is to initiate appropriate responses from the system to ensure that the fuel is adequately cooled in the event of a design basis accident or transient. The PCIS, and specifically MSL-SLD, acts to limit the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature increase through the initiation of an automatic isolation of the Main Steam Lines. This instrument is used to protect the nuclear system process barrier from over temperature conditions by minimizing the loss of coolant from the reactor vessel.

The PCIS-MSL-SLD calculation, CAL-E94-001 has been revised in accordance with DGC-E111 (Instrument Setpoint Guide). DGC-E111 meets the UFSAR section 7.1.3 specified Instrument Setpoint Methodology NEDC-31336 (General Electric Instrument Setpoint Methodology). As previously described, the existing Analytical Limit, Allowable Value, and Trip Setpoint used to establish the design bases for the MSL-SLD will not change.

ENCLOSURE 2 to NG-17-0095

DESCRIPTION OF COMMITMENT CHANGE

The information contained in this section identifies and briefly describes the commitment change made during the period beginning May 1, 2015 through April 31, 2017. The change described is being reported per the Nuclear Energy Institute's "Guidelines for Managing NRC Commitment Changes," dated July 1999.

Commitment Change **CA2003738 – Protective Coatings**

In our response to Generic Letter 98-04, NG-98-1901, dated November 11, 1998, NextEra Energy Duane Arnold stated that the qualification testing and evaluation of the Service Level I coatings used for new applications or repair/replacement activities were performed in accordance with ANSI Standard N101.2-1972. That commitment has been changed to allow qualification testing and evaluations of Service Level I coatings used for new applications or repair/replace activities inside containment to be performed in accordance with ANSI Standard N101.2 "Protective Coatings (Paints) for Light Water Nuclear Reactor Containments Facilities" or ASTM 3911-95 "Standard Test Method for Evaluating Coatings USED in Light Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions."

Regulatory Guide (RG) 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," Revision 1 was issued in July 2000. RG 1.54 Revision 1 states: *ANSI N101.4 and N101.2 were formally withdrawn in 1988; responsibility for updating, rewriting and issuing appropriate replacement standards was transferred to the American Society for Testing and Materials (ASTM), specifically, ASTM Committee D-33 on Protective Coating and Lining Work for Power Generation Facilities. However, RG 1.54 was not revised as new ASTM Standards were developed for the application and maintenance of nuclear power plant protective coatings.*

RG 1.54, Revision 1, further states: *ASTM D3911-95 provides guidance that is acceptable to the NRC Staff on procedures for evaluating protective coating systems test specimens under simulated DBA conditions. ASTM D3911-95 also provides guidance on conditions and test apparatus for temperature-pressure testing, conditions for radiation testing, and procedures for preparing, examining, and evaluating samples.* ASTM D3911-95 provides similar material, surface preparation, testing, and evaluation requirements with one notable difference. ASTM D3911-95 allows blisters provided they are intact and completely surrounded by sound coating. Blister size and density are evaluated using ASTM D714.

Therefore, it is acceptable to incorporate the option to use ASTM D3911-95 as an alternative for present and future Service Level 1 installations.