NEXTera ENERGY DUANE ARNOLD

April 12, 2016

NG-16-0080
10 CFR 50.90

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

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

Response to Request for Additional Information, License Amendment Request (TSCR-153) to Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits

References: 1) License Amendment Request (TSCR-153) to Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits, NG-15-0048, dated August 6, 2015 (ML15246A410)

2) Electronic Communication, Request for Additional Information – Duane Arnold Energy Center – LAR to Reduce the Reactor Steam Dome Pressure specified in the Technical Specifications 2.1.1, "Reactor Core Safety Limits" – MF6618, dated March 14, 2016

In Reference 1, NextEra Energy Duane Arnold, LLC submitted a License Amendment Request (LAR) for the Duane Arnold Energy Center (DAEC) pursuant to 10 CFR 50.90. Subsequently, the Nuclear Regulatory Commission (NRC) Staff requested, via Reference 2, additional information regarding that application be placed on the docket within 30 days.

Specifically, the NRC noted that the 685 psig value proposed as the lower bound for reactor steam dome pressure was slightly outside the pressure range (lower limit 700 psia or 685.3 psig) for the critical power correlations used for the DAEC fuel types.

Attachment 1 to this letter contains the full text of the NRC request, and the response to the requested information. This information has resulted in the 685 psig value specified in the original proposed TS change (Reference 1) being

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revised by DAEC to 686 psig in the new proposed TS pages and TS bases, which are included as Attachment 2. This minor change has no impact on the statements and conclusions documented in the original LAR.

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

This letter does not change any existing 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 12, 2016.

T. A. Vehec
Vice President, Duane Arnold Energy Center
NextEra Energy Duane Arnold, LLC

Attachments: As stated

cc: NRC Regional Administrator
NRC Resident Inspector
NRC Project Manager
A. Leek (State of Iowa)

ATTACHMENT 1

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-153)
TO REDUCE THE REACTOR STEAM DOME PRESSURE SPECIFIED IN THE REACTOR CORE SAFETY LIMITS

Response to NRC Request for Additional Information (RAI) for LAR to Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits

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Introduction

In Reference 1, NextEra Energy Duane Arnold, LLC submitted a License Amendment Request to the Technical Specifications (TS) for Duane Arnold Energy Center (DAEC). The proposed change would revise TS 2.1.1 to reduce the reactor steam dome pressure specified in the reactor core safety limits. In Reference 2, the NRC accepted the LAR for review.

In Reference 3, the NRC transmitted a Request for Additional information on this LAR. The responses to that RAI are provided below in Section 1.

The impact of this RAI response on the previously submitted Technical Specification changes is discussed in Section 2. References are noted in Section 3.

1. Response to NRC RAI

The NRC RAI (Reference 2) consists of one question. The question is provided below, followed by the response.

RAI-1

In page 4 of the application, dated August 6, 2015, the license stated,

Duane Arnold proposes to utilize the fact that the GE14 and GNF2 fuel that comprise the DAEC core, utilize critical power correlations that have an approved pressure range from 700 to 1400 psia.

Converting 700 psia to psig, the lower bound pressure for the GEXL17 and GEXL14 correlations is approximately 685.3 psig. As such, the 685 psig value specified in the proposed TS change is slightly outside the pressure range in which the GEXL17 and GEXL14 correlations are valid for GNF2 and GE14 fuel. Please provide further justification for the proposed 685 psig value or propose a revised pressure value for this TS change that is supported by the GEXL17 and GEXL14 correlations (e.g., 700 psia).

Response:

A value of 686 psig will be used instead of the previously proposed value of 685 psig. This new value is within the pressure range in which the GEXL17 and GEXL14 correlations are valid for GNF2 and GE14 fuel.

2. Discussion on Proposed Technical Specification Changes

The original LAR (Reference 1) used a lower limit of 685 psig in the proposed TS pages and the TS bases. The value has been updated to 686 psig in the new proposed TS pages and TS bases. This minor change has no impact on the statements and conclusions documented in the original LAR.

3. References

1. Duane Arnold Energy Center, *License Amendment Request (TSCR-153) to Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits*, NG-15-0048, August 6, 2015. (ADAMS Accession No.ML15246A410)

2. NRC email to Duane Arnold Energy Center, *LIC-109 Acceptance Review – Duane Arnold Energy Center (DAEC) LAR (TSCR-153) to Reduce the Reactor Steam Dome Pressure Specified in the Reactor Core Safety Limits – MF6618*, September 22, 2015.

3. NRC email to Duane Arnold Energy Center, *Request for Additional Information – Duane Arnold Energy Center – LAR to Reduce the Reactor Steam Dome Pressure specified in the Technical Specifications 2.1.1, "Reactor Core Safety Limits" – MF6618*, March 14, 2016.

ATTACHMENT 2

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-153)
TO REDUCE THE REACTOR STEAM DOME PRESSURE SPECIFIED IN THE REACTOR CORE SAFETY LIMITS

Revised Technical Specification and Technical Specification Bases Pages

3 Pages to Follow

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 Fuel Cladding Integrity – With the reactor steam dome pressure < ~~785~~ psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 21.7% RTP. 686

2.1.1.2 MCPR – With the reactor steam dome pressure ≥ ~~785~~ psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

2.1.1.3 Reactor Vessel Water Level – Reactor vessel water level shall be greater than 15 inches above the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1335 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Fully insert all insertable rods.

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 43% RTP. Thus, a THERMAL POWER limit of 21.7% RTP for reactor pressure < ~~785~~ 686 psig is conservative.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which transition boiling is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid transition boiling, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.

For SLO, the SLMCPR is greater to account for the increased uncertainties.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Main Steam Line Isolation

1.a. Reactor Vessel Water Level — Low Low Low

Low Reactor Pressure Vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level — Low Low Low Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level — Low Low Low Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSLs on Reactor Vessel Water Level - Low Low Low supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four level switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level — Low Low Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level — Low Low Low Allowable Value is chosen to be the same as the ECCS Reactor Vessel Water Level - Low Low Low Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential Loss of Coolant Accident (LOCA) to prevent offsite doses from exceeding 10 CFR 50.67 limits.

This Function isolates the Group 1 valves.

1.b. Main Steam Line Pressure — Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure — Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below ~~785~~ 686 psig, which results in a scram due to

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