NEXTera ENERGY DUANE ARNOLD

Attachment 6 Contains Proprietary Information

Withhold Attachment 6 from Public Disclosure in Accordance with 10 CFR 2.390

May 18, 2016

NG-16-0102
10 CFR 50.90

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

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

License Amendment Request (TSCR-161) to Revise Technical Specifications 2.1.1.2 Safety Limit Minimum Critical Power Ratio, and to Remove an Outdated Historical Footnote from Table 3.3.5.1-1

In accordance with the provisions of Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), NextEra Energy Duane Arnold, LLC (hereafter, NextEra Energy Duane Arnold) is submitting a request for an amendment to the Technical Specifications (TS) for Duane Arnold Energy Center (DAEC).

The proposed amendment revises TS Section 2.1.1, Reactor Core SLs, item 2.1.1.2, to change the Safety Limit Minimum Critical Power Ratio (SLMCPR) for two recirculation loop operation from ≥ 1.10 to ≥ 1.08, and to change the SLMCPR for single recirculation loop operation from ≥ 1.12 to ≥ 1.11. It also removes an outdated historical footnote from Table 3.3.5.1-1.

Attachment 1 provides an evaluation of the proposed changes to the SLMCPR. Attachment 2 provides marked-up pages of the existing TS to show the proposed changes to 2.1.1.2 and 3.3.5.1-1. Attachment 3 provides revised (clean) TS pages. Attachment 4 provides the affidavit from the vendor (GNF) performing the SLMCPR analysis. Attachment 5 provides the non-proprietary vendor discussion of the proposed SLMCPR change. Attachment 6 provides the proprietary version of Attachment 5. There are no new Regulatory Commitments or revisions to existing Regulatory Commitments.

As noted, the proposed amendment also removes an outdated historical footnote from Table 3.3.5.1-1. The footnote text itself states that it is specific only to an earlier refueling outage (RFO 23 in 2012) at Duane Arnold Energy Center, as such no further evaluation is provided for this change.

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Attachment 6 transmitted herewith contains Proprietary Information.
When separated from Attachment 6, this document is decontrolled.

NextEra Energy Duane Arnold, LLC, 3277 DAEC Road, Palo, IA 52324

Approval is requested by September 1, 2016, to support restart from Refueling Outage (RFO) 25, with the amendment being implemented within 60 days of its receipt.

In accordance with 10 CFR 50.91(b)(1), "Notice for Public Comment; State Consultation," a copy of this application and its attachments is being provided to the designated State of Iowa official.

The DAEC Onsite Review Group has reviewed the proposed license amendment request. 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 May 18, 2016,

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

Attachments: As stated

cc: Regional Administrator, USNRC, Region III,
Project Manager, USNRC, Duane Arnold Energy Center
Resident Inspector, USNRC, Duane Arnold Energy Center
A. Leek (State of Iowa)

ATTACHMENT 1 TO NG-16-0102

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-161)

For Revision of Technical Specifications 2.1.1.2 Safety Limit Minimum Critical Power Ratio, and to Remove an Outdated Historical Footnote from Table 3.3.5.1-1

EVALUATION OF PROPOSED CHANGES

Technical Specifications 2.1.1.2 Safety Limit Minimum Critical Power Ratio

1.0 SUMMARY DESCRIPTION

2.0 DETAILED DESCRIPTION

3.0 TECHNICAL EVALUATION

4.0 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

4.2 PRECEDENTS

4.3 SIGNIFICANT HAZARDS CONSIDERATION

4.4 CONCLUSIONS

5.0 ENVIRONMENTAL CONSIDERATIONS

6.0 REFERENCES

5 pages follow

1.0 SUMMARY DESCRIPTION

NextEra Energy Duane Arnold, LLC (NextEra) hereby requests an amendment to the Duane Arnold Energy Center (DAEC) Technical Specifications (TS). The requested amendment modifies TS 2.1.1, "Reactor Core SLs," to revise the Safety Limit Minimum Critical Power Ratio (SLMCPR) due to the cycle-specific analysis performed by Global Nuclear Fuel (GNF) for DAEC Cycle 26.

A marked-up copy of the proposed changes to the TS is provided in Attachment 2 and a clean copy in Attachment 3. The vendor affidavit in support of the proprietary version of the report is provided in Attachment 4. A non-proprietary version of the SLMCPR analysis report is provided in Attachment 5. A proprietary version of the SLMCPR analysis report is provided in Attachment 6.

2.0 DETAILED DESCRIPTION

The proposed change involves revising the SLMCPRs contained in TS 2.1.1.2 for two recirculation loop operation and single recirculation loop operation, based on the Cycle 26 core loading design changes. The SLMCPR value for two recirculation loop operation is being changed from ≥ 1.10 to ≥ 1.08. The SLMCPR value for single recirculation loop operation is being changed from ≥ 1.12 to ≥ 1.11.

The proposed amendment also removes an outdated historical footnote from Table 3.3.5.1-1. The footnote text itself states that it is specific only to an earlier refueling outage (RFO 23 in 2012) at Duane Arnold Energy Center, as such no further evaluation is provided for this change.

Marked-up TS pages 2.0-1 and 3.3-41 showing the requested changes are provided in Attachment 2.

3.0 TECHNICAL EVALUATION

The proposed change will revise the SLMCPRs contained in TS 2.1.1 for two recirculation loop operation and single recirculation loop operation to reflect the changes in the cycle-specific analysis performed by GNF for DAEC Cycle 26.

The purpose of the SLMCPR is to assure that the specified acceptable fuel design limit for fuel rod overheating is not violated during normal operation or design-basis anticipated operational occurrences (transients). Since the parameters that result in fuel rod overheating 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 cladding damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rod cladding, the critical power at which boiling transition is calculated to occur has been adopted as a convenient and conservative limit. However, 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 SLMCPR is defined as the critical power ratio in the limiting fuel assembly (with margin) such that, if the limit is not violated, 99.9% of the fuel rods will not be susceptible to boiling transition during normal operation or the most limiting postulated design basis transient event.

The new SLMCPRs are calculated using NRC-approved methodology described in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (Reference 1). A listing of the associated NRC-approved methodologies for calculating the SLMCPRs is provided in Section 3.0 ("Methodology") of Attachments 5 and 6.

The SLMCPRs include cycle specific parameters and, in general, are dominated by two key parameters: 1) flatness of the core bundle-by-bundle MCPR distribution, and 2) flatness of the bundle pin-by-pin power/R-Factor distribution. Information supporting the cycle specific SLMCPRs is included in Attachments 5 and 6. The attachments summarize the methodology, inputs, and results for the change in the SLMCPRs. The DAEC Cycle 26 core will be a full core of GNF2 fuel assemblies as described in Table 2 of Attachments 5 and 6.

No plant hardware or operational changes are required with this proposed change.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix A, General Design Criteria (GDC) 10, "Reactor Design," states that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operation occurrences. The proposed change in the SLMCPR values in TS 2.1.1 complies with the requirements of GDC 10 and will continue to assure that fuel clad integrity is maintained.

10 CFR 50.36, "Technical Specifications," paragraph (c)(1), requires that power reactor facility TS include safety limits for process variables that protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity. The SLMCPR analysis establishes SLMCPR values that will ensure that during normal operation and during abnormal operational transients, at least 99.9% of all fuel rods in the core do not experience transition boiling if the limit is not violated. Thus, the SLMCPR is required to be contained in TS.

4.2 Precedents

The NRC has approved similar SLMCPR changes for a number of plants, see References 2 through 5.

4.3 Significant Hazards Consideration

NextEra Energy Duane Arnold has evaluated the proposed changes to the TS using the criteria in 10 CFR 50.92 and has determined that the proposed changes do not involve a significant hazards consideration.

Description of Amendment Request: The proposed amendment would revise TS 2.1.1, "Reactor Core SLs," in accordance with the enclosed SLMCPR analysis report and also removes an outdated historical footnote from Table 3.3.5.1-1.

Basis for proposed no significant hazards determination: As required by 10 CFR 50.91(a), the NextEra Energy Duane Arnold analysis of the issue of no significant hazards consideration is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The SLMCPR ensures that 99.9% of the fuel rods in the core will not be susceptible to boiling transition during normal operation or the most limiting postulated design-basis transient event. The new SLMCPR values preserve the existing margin to the onset of transition boiling; therefore, the probability of fuel damage is not increased as a result of this proposed change.

   The determination of the new SLMCPRs has been performed using NRC-approved methods of evaluation. These plant-specific calculations are performed each operating cycle. The new SLMCPR values do not change the method of operating the plant; therefore, they have no effect on the probability of an accident initiating event or transient.

   The proposed change does not involve any plant modifications or operational changes that could affect system reliability or performance or that could affect the probability of operator error. The proposed change does not affect any postulated accident precursors, does not affect any accident mitigating systems, and does not introduce any new accident initiation mechanisms.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The SLMCPR is a TS numerical value, calculated to ensure that during normal operation and during abnormal operational transients, at least 99.9% of all fuel rods in the core do not experience transition boiling if the limit is not violated. The new SLMCPRs are calculated using NRC-approved methodology discussed in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel." The proposed change does not involve any new modes of operation, any changes to setpoints, or any plant modifications. The new SLMCPRs have been shown to be acceptable for DAEC Cycle 26 operation. The core operating limits will continue to be developed using NRC-approved methods. The proposed SLMCPRs or methods for establishing the core operating limits do not result in the creation of any new precursors to an accident. The proposed change does not involve any new or different methods for operating the facility. No new initiating events or transients result from the proposed change.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

The new SLMCPRs have been calculated using NRC-approved methods of evaluation with plant and cycle-specific input values for the fuel and core design for the upcoming cycle of operation. The SLMCPR values ensure that 99.9% of the fuel rods in the core will not be susceptible to boiling transition during normal operation or the most limiting postulated design-basis transient event. The MCPR operating limit is set appropriately above the safety limit value to ensure adequate margin when the cycle-specific transients are evaluated. Accordingly, the margin of safety is maintained with the revised values.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, NextEra concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92, and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATIONS

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," Revision 22.
2. Letter from Richard B. Ennis (U.S. Nuclear Regulatory Commission) to B. C. Hanson (Exelon Nuclear), "Peach Bottom Atomic Power Station, Unit 2 – Issuance of Amendment RE: Safety Limit Minimum Critical Power Ratio Change (CAC NO. MF5383)," dated February 8, 2016.
3. Letter from Robert Martin (U.S. Nuclear Regulatory Commission) to C. R. Pierce (Southern Nuclear Operating Company, Inc.), "Edwin I. Hatch Nuclear Plant, Unit No. 1, Issuance of Amendment Regarding Minimum Critical Power Ratio (CAC NO. MF6681)," dated January 29, 2016.

4. Letter from Alan B. Wang (U.S. Nuclear Regulatory Commission) to Entergy Operations, Inc., "Grand Gulf Nuclear Station, Unit 1 – Issuance of Amendment Regarding Technical Specification 2.1.1.2 of Technical Specification Section 2.1.1, Reactor Core SLs (TAC NO. MF5304)," dated August 18, 2015.
5. Letter from B. K. Singal (U.S. Nuclear Regulatory Commission) to M. E. Reddemann (Energy Northwest), "Columbia Generating Station – Issuance of Amendment RE: Technical Specification Change to Safety Limit Minimum Critical Power Ratio (TAC NO. MF5327)," dated May 11, 2015.

ATTACHMENT 2 TO NG-16-0102

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-161)
For Revision of Technical Specifications 2.1.1.2
Safety Limit Minimum Critical Power Ratio,
and to Remove an Outdated Historical Footnote from Table 3.3.5.1-1

PROPOSED TECHNICAL SPECIFICATIONS CHANGES
(MARKUP COPY)

2 pages 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.

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~~ 1.08 for two recirculation loop operation or ≥ ~~1.12~~ 1.11 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.

Table 3.3.5.1-1 (page 1 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level – Low Low Low	1,2,3, 4[a], 5[a]	4[b]	B	SR 3.3.5.1.1 SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	≥ 38.3 inches
b. Drywell Pressure - High	1,2,3	4[b]	B	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	≤ 2.19 psig
c. Reactor Steam Dome Pressure – Low (Injection Permissive)	1,2,3	4	C	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	≥ 363.3 psig and ≤ 485.1 psig
	4[a], 5[a]	4	B	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	≥ 363.3 psig and ≤ 485.1 psig
d. Core Spray Pump Discharge Flow – Low (Bypass)	1,2,3, 4[a], 5[a]	1 per pump~~(e)~~	E	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	≥ 256.6 gpm and ≤ 2382.1 gpm
e. Core Spray Pump Start Time Delay Relay	1,2,3, 4[a], 5[a]	1 per pump	C	SR 3.3.5.1.8 SR 3.3.5.1.9	≥ 2.6 seconds and ≤ 6.8 seconds
f. 4.16 kV Emergency Bus Sequential Loading Relay	1,2,3, 4[a], 5[a]	1 per pump	F	SR 3.3.5.1.5 SR 3.3.5.1.6 SR 3.3.5.1.9	≤ 3500 V
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level- Low Low Low	1,2,3, 4[a], 5[a]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	≥ 38.3 inches
b. Drywell Pressure - High	1,2,3	4	B	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	≤ 2.19 psig

(continued)

(a) When associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2, ECCS-Shutdown.

(b) Also required to initiate the associated Diesel Generator (DG).

~~(e) During Refuel Outage (RFO) 23, the MODE 4 and 5 requirement for Function 1.d is revised to be zero (0) required channels per pump.~~

ATTACHMENT 3 TO NG-16-0102

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-161)
For Revision of Technical Specifications 2.1.1.2
Safety Limit Minimum Critical Power Ratio,
and to Remove an Outdated Historical Footnote from Table 3.3.5.1-1

REVISED TECHNICAL SPECIFICATIONS PAGES

2 pages 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.

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

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.11 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

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Table 3.3.5.1-1 (page 1 of 5)
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FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level – Low Low Low	1,2,3, 4[(a)], 5[(a)]	4[(b)]	B	SR 3.3.5.1.1 SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	≥ 38.3 inches
b. Drywell Pressure - High	1,2,3	4[(b)]	B	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	≤ 2.19 psig
c. Reactor Steam Dome Pressure – Low (Injection Permissive)	1,2,3	4	C	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	≥ 363.3 psig and ≤ 485.1 psig
	4[(a)], 5[(a)]	4	B	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	≥ 363.3 psig and ≤ 485.1 psig
d. Core Spray Pump Discharge Flow – Low (Bypass)	1,2,3, 4[(a)], 5[(a)]	1 per pump	E	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	≥ 256.6 gpm and ≤ 2382.1 gpm
e. Core Spray Pump Start Time Delay Relay	1,2,3, 4[(a)], 5[(a)]	1 per pump	C	SR 3.3.5.1.8 SR 3.3.5.1.9	≥ 2.6 seconds and ≤ 6.8 seconds
f. 4.16 kV Emergency Bus Sequential Loading Relay	1,2,3, 4[(a)], 5[(a)]	1 per pump	F	SR 3.3.5.1.5 SR 3.3.5.1.6 SR 3.3.5.1.9	≤ 3500 V
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level- Low Low Low	1,2,3, 4[(a)], 5[(a)]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	≥ 38.3 inches
b. Drywell Pressure - High	1,2,3	4	B	SR 3.3.5.1.3 SR 3.3.5.1.8 SR 3.3.5.1.9	≤ 2.19 psig

(continued)

(a) When associated ECCS subsystem(s) are required to be OPERABLE per LCO 3.5.2, ECCS-Shutdown.

(b) Also required to initiate the associated Diesel Generator (DG).