UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 25, 2016

Mr. Thomas A. Vehec
Vice President
NextEra Energy
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

SUBJECT: DUANE ARNOLD ENERGY CENTER - ISSUANCE OF AMENDMENT TO REVISE AND RELOCATE PRESSURE AND TEMPERATURE LIMIT CURVES TO A PRESSURE AND TEMPERATURE LIMITS REPORT (CAC NO. MF6617)

Dear Mr. Vehec:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 294 to Renewed Facility Operating License No. DPR-49 for the Duane Arnold Energy Center (DAEC). The amendment consists of changes to the technical specifications (TSs) in response to your application dated July 30, 2015, as supplemented by letters dated December 18, 2015, February 19, March 11, and March 30, 2016.

The amendment modifies the DAEC TSs to revise Section 3.4.9, "[Reactor Coolant System (RCS)] Pressure and Temperature (P/T) Limits," to replace the existing reactor vessel heatup and cooldown rate limits and the P/T limit curves with references to the P/T Limits Report (PTLR). In addition, the definition of the PTLR is added to TS Section 1.1, "Definitions," and a section addressing administrative requirements for the PTLR is added to TS Section 5.6, "Reporting Requirements." The amendment also implements new P/T limits for DAEC that are valid through 54 effective full power years of operation.

A copy of the Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Mahesh L. Chawla, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures:
1. Amendment No. 294 to
   License No. DPR-49
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NEXTERA ENERGY DUANE ARNOLD, LLC

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 294
License No. DPR-49

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by NextEra Energy Duane Arnold, LLC dated July 30, 2015, as supplemented by letters dated December 18, 2015, February 19, March 11, and March 30, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 294, are hereby incorporated in the license. NextEra Energy Duane Arnold, LLC, shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Kimberly Green for

David J. Wrona, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License No. DPR-49 and Technical Specifications

Date of Issuance: July 25, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 294

DUANE ARNOLD ENERGY CENTER

RENEWED FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following page of Renewed Facility Operating License DPR-49 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
3	3

Replace the following pages of Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
1.1-5	1.1-5
3.4-20	3.4-20
3.4-21	3.4-21
3.4-22	3.4-22
3.4-23	3.4-23
3.4-24	3.4-24
5.0-21	5.0-21
--	5.0-21a

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I; Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

NextEra Energy Duane Arnold, LLC is authorized to operate the Duane Arnold Energy Center at steady state reactor core power levels not in excess of 1912 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 294, are hereby incorporated in the license. NextEra Energy Duane Arnold, LLC shall operate the facility in accordance with the Technical Specifications.

(a) For Surveillance Requirements (SRs) whose acceptance criteria are modified, either directly or indirectly, by the increase in authorized maximum power level in 2.C.(1) above, in accordance with Amendment No. 243 to Facility Operating License DPR-49, those SRs are not required to be performed until their next scheduled performance, which is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment No. 243.

(b) Deleted.

(3) Fire Protection Program

NextEra Energy Duane Arnold, LLC shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated August 5, 2011 (and supplements dated October 14, 2011, April 23, 2012, May 23, 2012, July 9, 2012, October 15, 2012, January 11, 2013, February 12, 2013, March 6, 2013, May 1, 2013, May 29, 2013, two supplements dated July 2, 2013, and supplements dated August 5, 2013 and August 28, 2013) and as approved in the safety evaluation report dated September 10, 2013. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

1.1 Definitions (continued)

MINIMUM CRITICAL POWER RATIO (MCPR)	film boiling occur intermittently with neither type being completely stable.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE — OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1912 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------------NOTE------------- Required Action A.2 shall be completed if this Condition is entered. -------------------------------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ------------NOTE------------- Required Action C.2 shall be completed if this Condition is entered. ---------------------------------	C.1 Initiate action to restore parameter(s) to within limits.	Immediately
	AND	
Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.2 Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	--------------------------NOTE------------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. --	
	Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in the PTLR.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	------------------------------NOTE------------------------------ Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. --- Verify the difference between the bottom head coolant temperature and the Reactor Pressure Vessel (RPV) coolant temperature is within the limits specified in the PTLR.	 Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.4	------------------------------NOTE------------------------------ Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. --- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is within the limits specified in the PTLR.	 Once within 15 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.5	----------------------------NOTE------------------------ Only required to be performed when tensioning the reactor vessel head bolting studs. -- Verify temperatures at the reactor vessel head flange and the shell adjacent to the head flange are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.6	----------------------------NOTE------------------------ Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. -- Verify temperatures at the reactor vessel head flange and the shell adjacent to the head flange are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.7	----------------------------NOTE------------------------ Not required to be performed until 12 hours after RCS temperature ≤ 100°F in MODE 4. -- Verify temperatures at the reactor vessel head flange and the shell adjacent to the head flange are within the limits specified in the PTLR.	In accordance with the Surveillance Frequency Control Program

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5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method(s) of monitoring, describe the degree to which the alternate method(s) are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   i) Limiting Conditions for Operation Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits"

   ii) Surveillance Requirements Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits"

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

5.6 Reporting Requirements

5.6.7 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

i) SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," Revision 1, dated June 2013.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 294 TO FACILITY OPERATING LICENSE NO. DPR-49

NEXTERA ENERGY DUANE ARNOLD, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By letter dated July 30, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15253A310), as supplemented by letters dated December 18, 2015, February 19, March 11, and March 30, 2016 (ADAMS Accession Nos. ML15357A051, ML16055A145, ML16076A214, and ML16097A599, respectively), NextEra Energy Duane Arnold, LLC (NextEra), the licensee for Duane Arnold Energy Center (DAEC), submitted a license amendment request (LAR) to revise the DAEC technical specifications (TSs). The proposed amendment would revise DAEC TS Section 3.4.9, "[Reactor Coolant System (RCS)] Pressure and Temperature (P/T) Limits," to replace the existing reactor vessel heatup and cooldown rate limits, and the P/T limit curves, with references to the P/T Limits Report (PTLR). In addition, the definition of the PTLR would be added to TS Section 1.1, "Definitions," and a section addressing administrative requirements for the PTLR would be added to TS Section 5.6, "Reporting Requirements."

The proposed PTLR would include revised 54 effective full-power years (EFPY) P/T curves, neutron fluence, and adjusted reference temperature (ART) values. The revised P/T limits adopt the U.S. Nuclear Regulatory Commission (NRC or Commission)-approved methodology addressed in the PTLR, and would replace the current TS P/T limits for DAEC.

The supplemental letters dated December 18, 2015, February 19, March 11, and March 30, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 8, 2015 (80 FR 76328).

2.0 REGULATORY EVALUATION

2.1 System Description

All components of the RCS are designed to withstand the effects of cyclic loads resulting from system P/T changes. These loads are introduced by heatup and cooldown operations, power transients, and reactor trips. The reactor pressure vessel (RPV) contains the reactor core and all associated support and alignment devices. The RPV acts as part of the reactor coolant pressure boundary (RCPB), which is the second barrier to the release of fission products to the environment.

In accordance with Appendix G to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, the TSs limit the P/T changes during RCS heatup and cooldown to within the design assumptions and the stress limits for cyclic operation. These limits are defined by P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic testing. Each curve defines an acceptable region for normal operation. The curves are used for operational guidance during heatup and cooldown maneuvering, when P/T indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

2.2 Regulations and Guidance

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the RCPB in nuclear power plants. The NRC staff evaluates the acceptability of a licensee's proposal to relocate the P/T limit curves to the PTLR based on the following NRC regulations and guidance:

- 10 CFR 50.36, "Technical specifications," which establishes the regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36(c), TSs are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TSs. For this review, the NRC staff interprets the requirements in 10 CFR 50.36 using the accumulation of generically approved guidance that is provided in the improved standard technical specifications (ISTS).

- 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation." These requirements mandate that all light-water nuclear power reactors must meet the fracture toughness requirements for the RCPB set forth in 10 CFR Part 50, Appendix G, in order to protect the integrity of the RCPB. The provisions of 10 CFR Part 50, Appendix G, require that the P/T limits for an operating light-water nuclear power reactor be at least as conservative as those that would be generated if the methods of Appendix G, "Fracture Toughness Criteria for Protection Against Failure," to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) were used to generate the P/T limits. The provisions of 10 CFR Part 50, Appendix G, also require that applicable surveillance data from RPV material surveillance programs developed in accordance with 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," be incorporated into the calculations of plant-specific P/T limits. Appendix G to 10 CFR Part 50 also requires that the P/T limits for operating reactors be generated using a method that accounts for the effects of neutron irradiation on the material properties of the RPV beltline materials. Finally, Table 1 of 10 CFR Part 50, Appendix G, provides the NRC's criteria for meeting the P/T limit requirements of the ASME Code, Section XI, Appendix G, as well as the minimum temperature requirements for the RPV during normal heatup, cooldown, and pressure test operations.

- Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits" (ADAMS Accession No. ML031110004), as supplemented by Technical Specifications Task Force (TSTF) Standard TS Change Traveler TSTF-419, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR" (ADAMS Accession No. ML012690234). GL 96-03 delineates the requirements for both the methodology and the PTLR including, but not limited to, the requirements of Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements." GL 96-03 addresses the technical information necessary for a licensee's implementation of a PTLR, and establishes the information which must be included in: (1) an acceptable PTLR methodology (with the P/T limit methodology as its subset), and (2) the PTLR itself. The proposed LAR invokes the methodology documented in NRC-approved Boiling Water Reactor Owners' Group (BWROG) Licensing Topical Report (TR) BWROG-TP-11-022-A, Rev. 1 (SIR-05-044, Rev. 1-A), "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," dated August 2013 (ADAMS Accession No. ML13277A557). The NRC staff's review focused on both the implementation of the DAEC PTLR, and the appropriate application of the approved methodology, BWROG-TP-11-022-A, to generate the proposed DAEC P/T limits.

- Attachment 1 of GL 96-03 requires that the licensee evaluate seven criteria to demonstrate the acceptability of its PTLR, as follows:

  (1) The PTLR methodology describes the transport calculation methods including computer codes and formula used to calculate neutron fluence values.
  (2) The PTLR methodology describes the surveillance program.
  (3) The PTLR methodology describes how the low temperature overpressure protection (LTOP) system limits are calculated applying system/thermal hydraulics and fracture mechanics.
  (4) The PTLR methodology describes the method for calculating the ART values using RG 1.99, Revision 2.
  (5) The PTLR methodology describes the application of fracture mechanics in the construction of P-T limits based on ASME B&PV Code, Section XI, Appendix G, and NUREG-0800, Section 5.3.2.
  (6) The PTLR methodology describes how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P-T limits for boltup temperature and hydrotest temperature.
  (7) The PTLR methodology describes how the data from multiple surveillance capsules are used in the ART calculation.

- Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," (ADAMS Accession No. ML003740284). RG 1.99, Rev. 2, contains methodologies for calculating the ART of the nil-ductility transition due to neutron irradiation. The ART is dependent upon a chemistry factor (CF) which in turn is dependent upon the amount of copper and nickel in the material. The CF, copper, and nickel contents may be adjusted based on the test results from 10 CFR Part 50, Appendix H surveillance programs.

- GL 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)" (ADAMS Accession No. ML031200626), which requested that licensees submit their plant-specific RPV data to the NRC staff for review.

- GL 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," (ADAMS Accession No. ML031070449), which requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

- RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (ADAMS Accession No. ML010890301). Neutron fluence calculations for use in ART and P/T limit curve analyses are acceptable if they are performed with approved methodologies or with methods which are shown to conform to the guidance in RG 1.190. RG 1.190 describes methods and assumptions acceptable to the NRC for determining the pressure vessel neutron fluence with respect to the General Design Criteria (GDC) contained in Appendix A of 10 CFR Part 50. In consideration of the guidance set forth in RG 1.190, GDC 14, 30, and 31 are applicable. GDC 14, "Reactor Coolant Pressure Boundary," requires the design, fabrication, erection, and testing of the RCPB so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires, in part, that components comprising the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical. GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," pertains to the design of the RCPB, stating:

  > The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) materials properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

- Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," (ADAMS Accession No. ML14149A165). RIS 2014-11 clarified that the beltline definition in 10 CFR Part 50, Appendix G, is applicable to all reactor vessel ferritic materials with projected neutron fluence values greater than 1 x 10^{17} n/cm^2 with energy (E) greater than 1 megaelectron volts (MeV), and that this fluence threshold remains applicable for the design life as well as throughout the licensed operating period.

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: [Lightwater Reactor (LWR)] Edition," Section 5.3.2, "Pressure-Temperature Limits, Upper Shelf Energy, and Pressurized Thermal Shock," Revision 2, (ADAMS Accession No. ML070380185). NUREG-0800, Section 5.3.2, provides an acceptable method for determining the P/T limits for ferritic materials in the beltline of the RPV based on the ASME Code, Section XI, Appendix G, methodology.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

The licensee proposed the following changes in its LAR:

(1) Modify the Definitions section of the TSs to include a definition of a PTLR to which the figures, values, and parameters for P/T limits will be relocated on a unit-specific basis in accordance with the methodology approved by the NRC in BWROG-TP-11-022-A that maintains the acceptance limits and the limits of the safety analysis. As noted in the definition, plant operation within these limits is addressed by individual specifications. The PTLR provides the explanations, figures, values, and parameters of the P/T limits for the applicable effective operational period.

(2) Revise Limiting Conditions for Operation and Surveillance Requirements Section 3.4.9 to replace the P/T system limits with a reference to the PTLR.

(3) Add Specification 5.6.7, "Reactor Coolant System Pressure and Temperature Limits Report," to the reporting requirements of TS Section 5.0.

The licensee stated that relocation of the P/T limit curves to the PTLR adopts the methodology provided in BWROG-TP-11-022-A. The licensee also stated that the proposed TS changes are consistent with the guidance provided in GL 96-03, which allows the licensee to relocate their P/T curves and associated numerical limits (such as heatup and cooldown rates) from the plant TSs to PTLRs, which are licensee-controlled documents. In order for the licensee to implement the PTLR, the analytical methods used to develop the P/T limits must be consistent with those previously reviewed and approved by the NRC and must be referenced in the Administrative Controls section of the plant TSs.

The proposed PTLR for DAEC contains new P/T limit curves that are valid for peak inner diameter neutron fluence values of 4.26×10^{18} n/cm^2 and 7.49×10^{18} n/cm^2 with E > 1 MeV, corresponding to 32 and 54 EFPY of core operation, respectively. The licensee stated that the P/T curves were developed in accordance with the methodology and template in BWROG-TP-11-022-A, as documented in the PTLR provided in Enclosures 5 and 6 of the licensee's July 30, 2015, submittal. The purpose of the BWROG methodology is to provide boiling water reactors (BWRs) with an NRC-approved report that can be referenced in plant TSs to establish BWR fracture mechanics methods for generating P/T curves and limits, and other associated numerical limits, thereby allowing BWR plants to adopt the PTLR option. The BWROG methodology does not include development or licensing of vessel neutron fluence methods.

The PTLR states that the RPV neutron fluence values utilized in the development of the DAEC P/T limit curves were calculated in accordance with RG 1.190 methods. Specifically, the neutron fluences used in the development of the ART values were calculated using the NRC-approved Radiation Analysis Modeling Application (RAMA) methodology.

3.2 NRC Staff Evaluation

According to GL 96-03, there are three separate licensee actions needed in order to relocate P/T curves and setpoints to a licensee-controlled document. The licensee must (1) have a methodology approved by the NRC to reference in its TSs; (2) develop a report such as a PTLR or a similar document to contain the figures, values, parameters, and any explanation necessary; and (3) modify the applicable sections of the TSs accordingly.

Relocation of the P/T curves does not eliminate the requirement to operate in accordance with the limits specified in Appendix G to 10 CFR Part 50. The requirement to operate within the limits in the PTLR is specified in and controlled by the TSs. Only the figures, values, and parameters associated with the P/T limits are to be relocated to the PTLR. In order for the curves to be relocated to a PTLR, a methodology for their development must be approved in advance by the NRC and use the guidance of GL 96-03. Revised TS 5.6.7 references the approved NRC methodology that must be used to determine the P/T limits in the PTLR. Thus, any changes to the methodology must be approved by the NRC. Furthermore, revised TS 5.6.7 requires the PTLR to be submitted to the NRC upon issuance for each reactor vessel fluence period, and when any changes are made to it.

3.2.1 PTLR Acceptability

The NRC staff examined the proposed PTLR and determined that it was developed from the template PTLR found in Appendix B of BWROG-TP-11-022-A. Furthermore, the NRC staff determined that the seven criteria specified in Attachment 1 of GL 96-03 were satisfied, as documented in Section 1.3 and Table 1-1 of BWROG-TP-11-022-A and summarized below:

(1) PTLR Criterion 1 specifies that the PTLR should provide the values of neutron fluence that are used in the ART calculation for the RPV beltline materials. The NRC staff confirmed that the 32 and 54 EFPY neutron fluence values for the RPV beltline materials are provided in Tables 7 and 8 of the DAEC PTLR. Therefore, the NRC staff determined that PTLR Criterion 1 is satisfied.

(2) PTLR Criterion 2 specifies that the PTLR should provide the surveillance capsule withdrawal schedule, or reference by title and number the documents in which the schedule is located. The PTLR must also reference the surveillance capsule reports by title and number if ARTs are calculated using surveillance data. The NRC staff determined that the surveillance capsule withdrawal schedule is correctly described in Appendix A of the DAEC PTLR and is based on the licensee's participation in the NRC staff-approved BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP). This meets the requirements of 10 CFR Part 50, Appendix H. Therefore, PTLR Criterion 2 is satisfied.

(3) PTLR Criterion 3 specifies that the PTLR should provide the LTOP system setpoint curves or parameters if LTOP system limits are relocated to the PTLR. The NRC staff noted that LTOP systems are not used for BWRs. Therefore, the NRC staff determined that this criterion is not applicable to DAEC.

(4) PTLR Criterion 4 specifies that the PTLR should identify the limiting ART values and limiting RPV beltline materials. The NRC staff confirmed that 32 and 54 EFPY ART values for all RPV beltline materials, including the limiting RPV beltline materials, are provided in Tables 7 and 8 of the DAEC PTLR. Therefore, the NRC staff determined that PTLR Criterion 4 is satisfied.

(5) PTLR Criterion 5 specifies that the PTLR should provide the P/T limit curves for heatup, cooldown, criticality, and pressure testing conditions. The NRC staff confirmed that P/T limit curves for these conditions are provided in Figures 1, 2, and 3 of the DAEC PTLR for 32 EFPY and Figures 4, 5 and 6 of the DAEC PTLR for 54 EFPY. The corresponding tabulated P/T limit values for these conditions are provided in Tables 1, 2, and 3 of the DAEC PTLR for 32 EFPY and Tables 4, 5 and 6 of the DAEC PTLR for 54 EFPY. Therefore, the NRC staff determined that PTLR Criterion 5 is satisfied.

(6) PTLR Criterion 6 specifies that the PTLR should identify the minimum temperatures on the P/T limit curves, such as the minimum boltup temperature and the hydrotest temperature. The NRC staff confirmed that the applicable minimum temperature criteria, including the minimum boltup temperature and the minimum temperature for criticality, are identified in the P/T limit curves provided in Figures 1, 2, and 3 of the DAEC PTLR for 32 EFPY and Figures 4, 5 and 6 of the DAEC PTLR for 54 EFPY. Therefore, the NRC staff determined that PTLR Criterion 6 is satisfied.

(7) PTLR Criterion 7 specifies that the PTLR should provide supplemental data and calculations of the CF in the PTLR if the RPV surveillance data are used in the ART calculation. PTLR Criterion 7 also specifies that the PTLR should evaluate the RPV surveillance data to determine if they meet the credibility criteria of RG 1.99, Revision 2, and provide the results of the credibility assessment. The NRC staff noted that Tables 7 and 8 of the DAEC PTLR list ART and CF calculations that are based on RPV surveillance data from the BWRVIP ISP. Therefore, the NRC staff issued a request for additional information (RAI), wherein the staff requested that the licensee provide the supporting ISP data and information from the BWRVIP ISP Data Source Book for determining these CFs. The licensee provided the applicable information from the BWRVIP ISP Data Source Book containing the ISP data and calculations for determining the CFs for the DAEC ISP materials. The NRC staff reviewed these calculations and verified that they are consistent with the provisions of RG 1.99, Revision 2. Therefore, the NRC staff determined that PTLR Criterion 7 is satisfied.

Based on the above supplemental information and the fact that the seven GL 96-03 criteria are satisfied, the NRC staff finds that the DAEC PTLR is acceptable to be referenced in TS 3.4.9.

3.2.2 Consideration of Surveillance Program Data

Appendix H to 10 CFR 50 requires that plants have surveillance programs and use the test results from those programs in the development of 10 CFR Part 50, Appendix G P/T limits. Furthermore, BWROG-TP-11-022-A requires the use of data from the BWRVIP ISP, as documented in the NRC's safety evaluation of that report.

DAEC is part of the BWRVIP ISP that combines all the surveillance programs for United States BWRs into a single integrated program. Under the ISP, the limiting plate and weld (target materials) in the participating BWR plants are represented by similar materials under irradiation in other BWR plants.

The representative ISP plate and weld materials for DAEC are contained in the Duane Arnold and Supplemental Surveillance Program Capsule F surveillance capsules. The RAI response of February 19, 2016, states that the CF for the representative surveillance plate material heat (B0673-1) is based on credible surveillance data that bounds the RG 1.99, Revision 2, CF. The representative plate material is the same heat number as the target plate in the DAEC RPV. Therefore, the fitted CF is used for the limiting beltline plate, consistent with the guidance in RG 1.99, Rev 2. The representative weld material heat (DA1 SMAW) is not the same as the target weld material heat (432Z0471). Since the material heats for the limiting weld material and representative surveillance capsule weld material do not match, the CF was calculated using the tables in RG 1.99, Rev. 2. Per Appendix A of the DAEC PTLR, DAEC continues to be a host plant under the ISP, and one more surveillance capsule is schedule to be removed and tested under the ISP in approximately 2027. The NRC staff reviewed the CF information included in the licensee's February 19, 2016, letter and enclosures, and, based on the analysis and evaluation described above, found the revised CFs and margin terms to be acceptable for the DAEC surveillance program materials.

3.2.3 Beltline Materials Adjusted Reference Temperature

Limiting material ART values are used in the development of P/T limits. These values are to be calculated following RG 1.99, Revision 2.

In Section 5.0 of the DAEC PTLR, the licensee provided detailed ART calculations for all beltline and extended beltline materials (i.e., neutron fluence greater than 1 x 10^{17} n/cm^2, E > 1 MeV) in Tables 7 and 8 of the PTLR for 32 and 54 EFPY, respectively. Corresponding parameters at the three-quarter of the RPV wall thickness (3/4T) were not provided in the attachments. Instead, the licensee applied the maximum tensile stress for both heatup and cooldown at the one-quarter RPV thickness (1/4T) location. This approach is conservative as the 1/4T material toughness is lower than that in the 3/4T locations.

The NRC staff performed independent calculations for the DAEC limiting beltline material ART values at 32 and 54 EFPY using the methods of RG 1.99, Revision 2. In its February 19, 2016, RAI response, the licensee provided data from the BWRVIP ISP that provided the information necessary to complete their independent ART assessment. The NRC staff independently calculated the ART values using the methods of RG 1.99, Rev. 2. The resulting 1/4T ART values were consistent with the licensee's calculated values.

3.2.4 P/T Limit Confirmatory Calculations

The DAEC PTLR contains new P/T curves for Hydrostatic Pressure and Leak Test (Curve A), Normal Operation Core Not Critical (Curve B), and Normal Operation - Core Critical (Curve C) at 32 EFPY and 54 EFPY. The composite P/T curves are extended below 0 pounds per square inch gauge (psig) to -14.7 psig. The proposed modifications to the TSs clarify vacuum fill operations for the RCS, which can result in system pressures below 0 psig.

The BWROG methodology requires that separate P/T curves be generated for the upper vessel region (including the feedwater nozzle), the beltline region, and the bottom head. The NRC staff performed confirmatory calculations for each of these regions to ensure that the proposed P/T limits for DAEC are at least as conservative as those determined using the methodology of the ASME Code, Section XI, Appendix G, for Curves A, B, and C conditions.

Using the information provided in the February 19, 2016, supplement, the NRC staff confirmed the P/T curves for DAEC using the methodology of the ASME Code, Section XI, Appendix G, BWROG-TP-11-022-A, and the ART values reported in the PTLR for the beltline materials, including the instrumentation (N16) and recirculation inlet (N2) nozzles. The N16 nozzle is a ferritic forged nozzle design. It is welded to the RPV using a full penetration weld and does not experience any significant cycling. The N2 nozzle is the limiting beltline nozzle based on the examination of thermal transients.

In the July 30, 2015, submittal, the NRC staff noted a difference between the TS P/T curves for 32 EFPY and the proposed PTLR P/T curves for 32 EFPY. The NRC issued an RAI requesting that the licensee provide an explanation regarding the differences in the curves. In the February 19, 2016, RAI response, the licensee stated that the differences between the TS P/T curves and the proposed PTLR P/T curves were the result of incorporating various updates, including an updated CF for surveillance plate, recent fluence analysis, and a 25 degree Fahrenheit per hour heat up rate for the hydrostatic test. The licensee also stated the following:

> Although both curves periods of applicability were updated and submitted, Duane Arnold will transition to the 54 EFPY P-T curves upon approval of the PTLR as DAEC is nearing 32 EFPY. The 32 EFPY P-T limit curves in the PTLR will not be used.

The NRC staff finds the proposal of not using the proposed PTLR 32 EFPY P/T limit curves and instead implementing the 54 EFPY P/T limit curves prior to the restart from refueling outage 25 to be acceptable because this is a conservative approach.

The NRC staff performed detailed calculations for Curves A, B, and C for DAEC, and independently confirmed the licensee's P/T curves. The NRC staff also verified that the proposed P/T limits meet the minimum temperature requirements of 10 CFR 50, Appendix G, which contains additional requirements determined from the most limiting material in closure head flange and vessel flange regions.

Based on its confirmatory calculations, the NRC staff determined that the DAEC proposed P/T limits for Curves A, B, and C meet the minimum temperature requirements of 10 CFR Part 50, Appendix G, are as conservative or more conservative than P/T limits generated using the methods of the ASME Code, Section XI, Appendix G, and implement the methodologies of NRC-approved TR BWROG-TP-11-022-A. Thus, the NRC staff finds the licensee's revised P/T limits for 54 EFPY acceptable for DAEC.

3.2.5 Neutron Fluence Methodology

The licensee stated that neutron fluence calculations supporting the proposed, updated P/T limits were performed in accordance with RG 1.190 using the RAMA fluence methodology. The RAMA methodology has been generically approved for use at BWRs, subject to the condition that plant geometry-specific validation must be performed.

The NRC staff reviewed the neutron fluence methodology, consistent with NRC-approved RAMA methodology. The proposed PTLR references the RAMA methodology in TransWare Report No. DAE-FLM-001-R-004, Rev. 0, "Duane Arnold Energy Center Fluence Assessment Report – End of Cycle 24," dated April 3, 2015. Additionally, the PTLR references the DAEC license amendment dated August 25, 2003 (ADAMS Accession No. ML032310536), which used the General Electric fluence method from the General Electric Licensing TR NEDC-32983P-A. Given these two references in the proposed PTLR, the NRC staff requested the licensee supplement the application to clarify how the RAMA fluence methodology was applied in the proposed PTLR.

The licensee supplemented the application by letter dated February 19, 2016. The licensee clarified in the supplement that the GE fluence methodology is not used in the proposed DAEC PTLR. Instead, the fluence used in the development of irradiation embrittlement projections or ART projections for the reactor vessel materials was based on the TransWare Report No. DAE-FLM-001-R-004, Rev. 0. Based on the information provided in the supplement, the NRC staff determined that the neutron fluence methodology is based on the NRC-approved RAMA methodology for DAEC, and is, therefore, acceptable for the proposed PTLR.

3.2.6 Compliance with TSTF-419

Technical Specification Change Traveler TSTF-419, as approved by the NRC staff in a March 21, 2002, letter (ADAMS Accession No. ML020800488), and modified by the changes proposed in an NRC letter dated August 4, 2011 (ADAMS Accession No. ML110660285), requires that references to NRC-approved TRs for the PTLR methodology must include the revision number and date of the TR. The proposed markup of TS 5.6.7 in the licensee's LAR references Revision 1 of TR BWROG-TP-11-022-A dated June 2013. Therefore, the requirements of TSTF-419 are satisfied.

If the licensee revises the PTLR to use a different revision of a TR, it must submit a new LAR.

3.3 <u>Technical Evaluation Conclusion</u>

The NRC staff concludes that the proposed changes to TS Sections 1.0, 3.4.9, and 5.0 for DAEC to delete references to the P/T curves and to include references to the PTLR, are acceptable and satisfy the licensee actions required by GL 96-03.

In addition, the NRC staff finds that the licensee has properly developed the PTLR for DAEC to control the development of future P/T limits under licensee controlled programs. The proposed DAEC PTLR meets the seven criteria of GL 96-03 and is approved for implementation as part of the DAEC licensing basis. As long as the P/T limit methodology remains the same, this implementation of the PTLR allows the licensee to revise the P/T limits for DAEC through 54 EFPY under the 10 CFR 50.59 process.

Based on its review of the licensee's material information, the NRC staff finds that the licensee's CFs and ART values are correct and acceptable for use. Based on its confirmatory calculations of the licensee's proposed P/T limits for 54 EFPY for DAEC, the NRC staff finds that all of the licensee's proposed P/T limits meet the requirements of 10 CFR Part 50, Appendix G. Additionally, the NRC staff finds that the P/T limits included in the submittal were calculated per the guidance in the ASME Code, BWROG-TP-11-022-A. Therefore, the NRC staff finds the revised P/T limits for 54 EFPY acceptable. The NRC staff finds that the requirements of TSTF-419 are satisfied because the proposed LAR references the NRC-approved TR BWROG-TP-11-022-A for the PTLR methodology.

On the basis of the considerations discussed above, the NRC staff verified that the licensee's application satisfies the licensee required actions of GL 96-03. The NRC staff concludes that the licensee provided an acceptable means of establishing and maintaining the detailed values of the P/T limit curves for DAEC in a PTLR through 54 EFPY. The revised P/T limits adopt the NRC-approved methodology addressed in the PTLR. Furthermore, because plant operation continues to be limited in accordance with the requirements of Appendix G to 10 CFR Part 50, and the P/T limits in the TSs will be established using a methodology approved by the NRC, these changes will not adversely impact plant safety. Additionally, based on the above assessment, the NRC staff finds that the proposed change meets the 10 CFR 50.36 requirements. Therefore, the NRC staff concludes that the proposed PTLR for DAEC is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Iowa State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, which was published in the *Federal Register* on December 8, 2015 (80 FR 76328), and there has been no public comment on that finding. The amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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.

Principal Contributor: C. Fairbanks
M. Hardgrove

Date of issuance: July 25, 2016

A copy of the Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Mahesh L. Chawla, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures:
1. Amendment No. 294 to
   License No. DPR-49
2. Safety Evaluation

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ADAMS Accession No.: ML16180A086 ***via memo**

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