NEXTera ENERGY
DUANE ARNOLD

Attachment 5 Contains Proprietary Information

Withhold Attachment 5 from Public Disclosure in Accordance with 10 CFR 2.390

July 30, 2015

NG-15-0235
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-144) to Revise and Relocate Pressure and Temperature Limit Curves to a Pressure and Temperature Limits Report

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 1.1, "Definitions," Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits," and Section 5.6, "Reporting Requirements," by replacing the existing reactor vessel heatup and cooldown rate limits and the pressure and temperature (P-T) limit curves with references to a Pressure and Temperature Limits Report (PTLR).

Attachment 1 provides an evaluation of the proposed changes. Attachment 2 provides marked-up pages of the existing TS to show the proposed changes. Attachment 3 provides revised (clean) TS pages. Attachment 4 provides the marked-up TS Bases pages for information only. Attachment 5 provides the proprietary DAEC PTLR. Attachment 6 provides the non-proprietary version of Attachment 5. There are no new Regulatory Commitments or revisions to existing Regulatory Commitments.

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.

Attachment 5 transmitted herewith contains Proprietary Information.
When separated from Attachment 5, this document is decontrolled.

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 July 30, 2015.

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-15-0235

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-144) TO REVISE AND RELOCATE PRESSURE AND TEMPERATURE LIMITS CURVES TO A PRESSURE AND TEMPERATURE LIMITS REPORT

EVALUATION OF PROPOSED CHANGES

1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, NextEra Energy Duane Arnold, LLC (NextEra Energy Duane Arnold) hereby requests an amendment to Duane Arnold Energy Center (DAEC) Technical Specifications (TS). The requested amendment would modify the TS by replacing the reactor coolant system (RCS) pressure and temperature (P-T) limit curves with references to the Pressure and Temperature Limits Report (PTLR).

Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," (Reference 7.1) provides guidance for preparing a license amendment request to modify the TS to relocate the P-T limit curves contained in plant TS to a PTLR. GL 96-03 Attachment 1 requirements for relocating P-T limit curves to a PTLR are (1) have a methodology approved by the NRC to reference in its TS; (2) develop a report such as a PTLR to contain the figures, values, parameters, and any explanation necessary; and (3) modify the applicable sections of the TS accordingly. The NRC concluded in Reference 7.2 that Licensing Topical Report (LTR) BWROG-TP-11-022, Revision 1 satisfies the criteria in Attachment 1 to GL 96-03 and provides adequate methodology for BWR licensees to calculate P-T Limit curves. This conclusion was reached because the limited modifications in LTR BWROG-TP-11-022, Revision 1 (as compared with LTR SIR-05-044-A) were identified and evaluated and determined to be acceptable, while the rest of LTR BWROG-TP-11-022, Revision 1 contains only editorial changes and remains acceptable based on the February 6, 2007 Safety Evaluation attached to Structural Integrity Associates Licensing Topical Report (LTR) SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," (Reference 7.3). Additionally, the TS changes in this license amendment request are consistent with the guidance in Technical Specification Task Force (TSTF) Traveler TSTF-419-A, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," (Reference 7.4) and the guidance in the August 4, 2011 NRC letter (Reference 7.5) that requires the full methodology citation in TS Section 5.6, "Reporting Requirements."

2.0 PROPOSED CHANGES

The proposed changes include:

- TS Section 1.1, "Definitions" - A new definition, "Pressure and Temperature Limits Report," is added. The wording for this definition is consistent with that in Reference 7.4.

- TS Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits" - The P-T limit curves and the associated TS wording have been deleted and replaced with references to the PTLR.

- TS Section 5.6, "Reporting Requirements" - Section 5.6.1, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," is added. The format and content of Section 5.6.1 is consistent with Reference 7.4 and the guidance in Reference 7.5, which requires the full topical report citation to be included in the TS. This new Section: (1) identifies the individual TS that address RCS pressure and temperature limits; (2) identifies the NRC-approved Topical Report, including revision number and date for a complete citation; and (3) requires the PTLR to be provided to the NRC for each reactor vessel fluence period and for any revision or supplement.

A marked-up copy of the proposed changes to the TS is provided in Attachment 2. Attachment 3 provides revised (clean) TS pages. Proposed revisions to the TS Bases are also included for

information only in Attachment 4. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program upon receipt of the NRC approved License Amendment. Attachment 5 provides the proprietary PTLR, which includes P-T curves developed for all plant conditions at 54 effective full power years (EFPY). Attachment 6 provides the non-proprietary version of Attachment 5. TS Section 3.4.9 currently provides curves valid to 32 EFPY. The 2001 Edition of the ASME Boiler and Pressure Vessel Code including 2003 Addenda was used in this evaluation.

3.0 TECHNICAL ANALYSIS

10 CFR 50, Appendix G requires licensees to establish limits on the pressure and temperature of the reactor coolant pressure boundary (RCPB) in order to protect against brittle failure. These limits are defined by P-T curves for normal operations (including heatup and cooldown operations of the RCS, normal operation of the RCS with the reactor being in a critical condition and anticipated operational occurrences) and during pressure testing conditions (i.e., inservice leak rate testing and / or hydrostatic testing conditions). Historically, utilities have submitted License Amendment Requests (LARs) to update their P-T curves. Processing LARs has caused both the NRC and licensees to expend resources that could otherwise be devoted to other activities. The SIA LTR provides a generically approved method for utilities to generate P-T curves.

GL 96-03 allows plants to relocate their P-T curves and the associated numerical limits (such as heatup / cooldown rates) from the plant TS to a PTLR - a licensee-controlled document. As stated in the generic letter, during development of the improved Standard Technical Specifications (STS), a change was proposed to relocate the P-T limits contained in the plant TS to a PTLR. As one of the improvements to the STS, the NRC staff agreed with the industry that the P-T curves could be relocated outside the plant TS to a PTLR so that licensees could maintain these limits efficiently.

TSTF-419-A and the associated LTRs provide the ability for BWR licensees to relocate their P-T curves and the associated numerical values (such as heatup / cooldown rates) from the facility TS to a PTLR, a licensee-controlled document, using the guidelines in GL 96-03. The transmittal letter for the NRC Safety Evaluation Report (SER), dated February 6, 2007 that is contained in Reference 7.3 states, "The NRC staff has found that SIR-05-044 is acceptable for referencing in licensing applications for General Electric-designed boiling water reactors to the extent specified and under the limitations delineated in the TR [Technical Report] and in the enclosed final SE."

The proposed DAEC PTLR is based on the methodology and template provided in SIR-05-044-A. The purpose of the DAEC PTLR is to present operating limits related to Reactor Coolant System (RCS) pressure versus temperature limits during heatup, cooldown, and hydrostatic / class 1 leak testing. The curves, which have been prepared using NRC approved methodology, will allow system pressurization at lower temperatures thus saving critical path time and provide improved work environment conditions for the inspectors during leak testing inspections.

To apply the PTLR option, the method used to develop the P-T curves and associated limits must be NRC approved. Also, the associated LTR is required to be referenced in the specification for the PTLR program in the plant TSs. The SIA LTR provides one of the current NRC-approved BWROG fracture mechanics methodologies for generating P-T curves / limits.

As discussed in the following sections, the new P-T curves apply at the fluence levels associated with the twenty-year renewed operating license period. A full set of P-T curves was developed for all plant conditions at 54 EFPY, including curves for the following conditions:

- hydrostatic pressure testing (Curve A),
- plant operation - core not critical (Curve B), and
- plant operation - core critical (Curve C).

3.1 DEVELOPMENT OF THE P-T CURVES IN ACCORDANCE WITH THE SIA METHODOLOGY

One of the prerequisites for the PTLR option is that the method used to develop the P-T curves and associated limits are NRC approved, and that the associated LTR for such approval be referenced in the specification for the PTLR program in the plant TSs. The SIA LTR provides one of the current NRC-approved BWROG fracture mechanics methodologies for generating P-T curves / limits and allows BWR plants to adopt the PTLR option in accordance with TSTF-419-A and GL 96-03.

As discussed in the NRC's SER approving the SIA LTR, the licensing topical report has three sections and two appendices, the content of which is summarized below.

- Section 1.0 describes the background and purpose for the LTR.
- Section 2.0 of the SIA LTR provides the fracture mechanics methodology and its basis for developing P-T limits. Attachment 1 of GL 96-03 provides seven technical criteria that contents of a methodology should conform to, to develop P-T limits and to be acceptable by the NRC staff.
- Section 3.0 of the SIA LTR provides a step-by-step procedure for calculating P-T limit curves. This section indicates that typically three reactor pressure vessel (RPV) regions are evaluated with respect to P-T limits: (1) the beltline region; (2) the bottom head region; and (3) the non-beltline region.
- Appendix A of the LTR provides guidance for evaluating surveillance data.
- Appendix B provides a template for development of an acceptable PTLR.

The NRC staff evaluation of the contents of the BWROG SIA methodology against the seven criteria of GL 96-03 is provided in Section 3.1 of the SER.

3.2 ADJUSTED REFERENCE TEMPERATURE (ART) AND FLUENCE

Radiation embrittlement of RPV materials causes a decrease in the fracture toughness. Regulatory Guide (RG) 1.99 describes general procedures to calculate the effects of neutron irradiation embrittlement on alloy steels used in RPVs. The fluence value of 1.0×10^{17} n/cm^2 (E > 1 MeV) is considered to be a lower bound value below which there are insignificant material effects due to irradiation based on Section III.A of 10 CFR 50 Appendix H. The local fracture toughness, at the postulated flaw location (1/4 wall thickness or 1/4t), is determined considering initial RT_{NDT}, local fluence, margins, and chemical composition. The ART values reflect the results from the most recent 108° surveillance capsule (Reference 7.6). The fluences used in the development of the ART values were calculated using the NRC approved RAMA methodology (References 7.7 and 7.8). The ART is used to determine the fracture toughness described per the ASME B&PV Code, Section XI, Appendix G evaluations. As the chemistry factor used for the determination of the ART for the PTLR used EPRI proprietary data, this has

necessitated submittal of both a proprietary and a non-proprietary PTLR. Tables 7 and 8 of the PTLR contain the inputs and materials considered for the ART calculation.

3.3 TEMPERATURE AND PRESSURE INSTRUMENTS UNCERTAINTY AND THE PRESSURE HEAD FOR COLUMN OF WATER IN THE RPV

The instrument uncertainty assumed in the analysis for pressure is 0 psig. The instrument uncertainty assumed in the analysis for temperature is 0°F. The instrument uncertainty is assumed to be zero since the temperature and pressure monitoring are procedurally controlled and margin is placed on these limits for monitoring vessel temperature and pressure conditions. Procedural controls will continue to include sufficient margin with the introduction of the PTLR. The pressure head to account for the column of water in the RPV is 28.7 psig.

The composite P-T curves are extended below 0 psig to -14.7 psig which bounds the maximum expected vacuum pressure as well as external applied pressures the reactor vessel may experience.

3.4 LOWEST OPERATING TEMPERATURE

To comply with the Safety Evaluation Report (SER) for the curve development methodology with respect to the NRC condition for the lowest service temperature (LST) (Reference 7.9), the minimum temperature is set to 74°F, which is equal to the RT_{NDT},max + 60°F. This value is consistent with the minimum temperature limits and minimum bolt-up temperature specified in currently approved Technical Specifications (Reference 7.10). The value was also confirmed through a review of the piping design specifications to ensure the LSTs for non-RPV RCPB components are bounded by the bolt-up temperature of 74°F.

3.5 REACTOR VESSEL VACUUM CONSIDERATION

The P-T limit curves remain applicable for small values of negative gauge pressure and may be extended to 0 psia (-14.7 psig), i.e., the permissible temperature at 0 psig applies through -14.7 psig. The RPV can withstand significant external pressures, and the RPV cylinder, bottom head and top head locations have adequate structural margin for values of negative gauge pressure in excess of -14.7 psig, which greatly exceeds any vacuum that could be pulled on the RPV.

3.6 UPPER SHELF ENERGY (USE) ASSESSMENT

In support of the P-T limit curves, an assessment was performed to determine the impact of extending the P-T limit curves to 54 EFPY. The assessment demonstrated that end of life USE values for the DAEC beltline materials remain bounded by the BWRVIP-74-A (Reference 7.11) Equivalent Margin Analysis (EMA) evaluation and are expected to remain within the limits of RG 1.99 and satisfy the margin requirements of 10 CFR 50 Appendix G for 54 EFPY of operation.

4.0 REGULATORY SAFETY ANALYSIS

4.1 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

NextEra Energy Duane Arnold has evaluated the proposed changes to the Technical Specifications (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 requested amendment would modify the TS by replacing the reactor coolant system (RCS) pressure and temperature (P-T) limit curves with references to the Pressure and Temperature Limits Report (PTLR). The requested amendment would also adopt Licensing Topical Report (LTR) BWROG-TP-11-022, Revision 1, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," which has received NRC approval. The new P-T curves have been developed for all plant conditions at 54 effective full power years (EFPY).

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 proposed changes modify the TS by replacing the reactor coolant system (RCS) pressure and temperature (P-T) limit curves with references to the Pressure and Temperature Limits Report (PTLR). The requested amendment would also adopt Licensing Topical Report (LTR) BWROG-TP-11-022, Revision 1, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," for the preparation of new DAEC P-T curves developed for all plant conditions at 54 effective full power years (EFPY). 10 CFR 50 Appendix G establishes requirements to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. Implementing the NRC-approved methodology for calculating P-T curves and relocating those P-T curves from the TS to the PTLR provide an equivalent level of assurance that RCPB integrity will be maintained as specified in 10 CFR 50 Appendix G.

The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not require any physical change to any plant SSCs nor do they require any change in systems or plant operations. The proposed changes are consistent with the safety analysis assumptions and resultant consequences.

Therefore, the proposed changes do 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 proposed changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes.

The proposed changes do not introduce any new accident precursors, nor do they impose any new or different requirements or eliminate any existing requirements. RCPB integrity will continue to be maintained in accordance with 10 CFR 50 Appendix G; therefore, the assumed accident performance of plant structures, systems and components will not be affected. The proposed changes do not alter assumptions made in the safety analysis.

Therefore, the proposed changes do 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

Margin of safety is related to confidence in the ability of the fission product barriers (fuel cladding, reactor coolant system, and primary containment) to perform their design functions during and following postulated accidents. The proposed changes do not affect the function of the RCPB or its response during plant transients. By calculating the P-T curves using NRC-approved methodology, adequate margins of safety relating to RCPB integrity are maintained. The proposed changes do not alter the manner in which the safety limits are determined. There are no changes to setpoints at which protective actions are initiated. The operability requirements for equipment assumed to operate for accident mitigation are not affected.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

4.2 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The NRC established requirements in 10 CFR 50 Appendix G, "Fracture Toughness Requirements," in order to protect the integrity of the RCPB in nuclear power plants. Appendix G requires that the pressure-temperature limits for the reactor vessel must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. Appendix G also requires that the pressure-temperature limits be met for all plant conditions.

10 CFR 50.36, "Technical Specifications," provides the regulatory requirements for the content required in the TS. Historically, the P-T curves have been contained in the TS, which necessitates the submittal of license amendment requests to update the P-T curves. This caused both the NRC and licensees to expend resources that could otherwise be devoted to other activities. Reference 2 allows plants to relocate P-T curves from their plant TS to a PTLR. One of the prerequisites for having the PTLR option is that the P-T curves be derived using methodologies approved by the NRC. DAEC P-T curves have been developed for all plant conditions using Reference 1 that has been approved by the NRC.

DAEC UFSAR Section 3.1, "Conformance to AEC General Design Criteria for Nuclear Power Plants," provides an evaluation of the design basis of DAEC against Appendix A of 10 CFR 50 effective May 21, 1971 and subsequently amended on July 7, 1971. Five AEC General Design Criteria (GDC) are applicable to the proposed changes. The first applicable AEC GDC is Criterion 14, "Reactor Coolant Pressure Boundary," which states, "The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture." The second applicable AEC GDC is Criterion 15, "Reactor Coolant System Design," which states, "The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences." The third applicable AEC GDC is Criterion 30, "Quality of Reactor Coolant Pressure Boundary," which states, "Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage." The fourth applicable AEC GDC is Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," which states, "The reactor coolant pressure boundary shall be designed with sufficient margin to ensure 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 the 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) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws." The fifth applicable AEC GDC is Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," which states, "Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity and (2) an appropriate material surveillance program for the reactor pressure vessel."

NextEra Energy Duane Arnold has evaluated the proposed changes against the applicable regulatory requirements and acceptance criteria. Implementing the proposed changes provides an equivalent level of assurance that RCPB integrity will be maintained as specified in 10 CFR 50 Appendix G, TS, and AEC GDC. Based on this, there is reasonable assurance that the health and safety of the public, following approval of this TS change is unaffected.

5.0 ENVIRONMENTAL CONSIDERATION

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment of an operating license for a facility requires no environmental assessment, if the operation of the facility in accordance with the proposed amendment does not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (3) result in a significant increase in individual or cumulative occupational radiation exposure. NextEra Energy Duane Arnold has reviewed this license amendment request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment. The basis for this determination is as follows.

Basis

This change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

As demonstrated in the 10 CFR 50.92 evaluation, the proposed amendment does not involve a significant hazards consideration.

The proposed amendment does not result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The proposed amendment does not change or modify the design or operation of any plant systems, structures, or components. The proposed amendment does not affect the amount or types of gaseous, liquid, or solid waste generated onsite. The proposed amendment does not directly or indirectly affect effluent discharges.

The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure. The proposed amendment does not change or modify the design or operation of any plant systems, structures, or components. The proposed amendment does not directly or indirectly affect the radiological source terms.

6.0 PRECEDENT

This License Amendment Request is similar to a License Amendment Request approved by letter dated January 26, 2011 (ML110050298), "Pilgrim Nuclear Station - Issuance of Amendment Regarding Revised Pressure and Temperature (P-T) Limit Curves and Relocation of P-T Curves to the Pressure and Temperature Limits Report (TAC NO. ME3253)," and another License Amendment Request approved by letter dated February 27, 2013 (ML13025A155), "Monticello Nuclear Generating Plant – Issuance of Amendment to Revise and Relocate Pressure Temperature Curves to a Pressure Temperature Limits Report (TAC NO. ME7930)."

7.0 REFERENCES

7.1 Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996

7.2 Letter from S. Bahadur (NRC) to F. Schiffley (BWROG), "Final Safety Evaluation for Boiling Water Reactor Owners' Group Topical Report BWROG-TP-11-022, Revision 1, November 2011, 'Pressure-Temperature Limits Report Methodology for Boiling Water Reactors,' (TAC No. ME7649)," dated May 16, 2013 (ML13107A062)

7.3 Structural Integrity Associates Licensing Topical Report (LTR) SIR-05-044-A, Revision 0, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," dated April 2007 (ML072340283)

7.4 TSTF-419-A, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR"

7.5 Letter from J. Jolicoeur (NRC) to Technical Specifications Task Force, "Implementation of Travelers TSTF-363, Revision 0, 'Revise Topical Report References in ITS 5.6.5, COLR [Core Operating Limits Report],' TSTF-408, Revision 1, 'Relocation of LTOP [Low-Temperature Overpressure Protection] Enable Temperature and PORV [Power-Operated Relief Valve] Lift Setting to the PTLR [Pressure-Temperature Limits Report],' and TSTF-419, Revision 0, 'Revise PTLR Definition and References in ISTS [Improved Standard Technical Specification] 5.6.6, RCS [Reactor Coolant System] PTLR,'" dated August 4, 2011 (ML110660285)

7.6 BWRVIP- 279NP, BWR Vessel and Internals Project, Testing and Evaluation of the Duane Arnold 108° Capsule, EPRI, Palo Alto, CA: 2014. 3001003134

7.7 Letter from William H. Bateman (USNRC) to Bill Eaton (BWRVIP), "Safety Evaluation of Proprietary EPRI Reports BWRVIP-114, -115, -117, and -121 and TWE-PSE-001-R-001," dated May 13, 2005

7.8 Letter from Matthew A. Mitchell (USNRC) to Rick Libra (BWRVIP), "Safety Evaluation of Proprietary EPRI Report BWR Vessel and Internals Project, Evaluation of Susquehanna Unit 2 Top Guide and Core Shroud Material Samples Using RAMA Fluence Methodology (BWRVIP-145)," dated February 7, 2008

7.9 Licensing Topical Report (LTR) BWROG-TP-11-022-A (SIR-05-044), Revision 1, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," June 2013 (ML13277A557)

7.10 Letter to Mr. Mark A. Peifer, Site Vice President Duane Arnold Energy Center, from Darl S. Hood, Nuclear Regulatory Commission, dated August 25, 2003, subject: DUANE ARNOLD ENERGY CENTER - ISSUANCE OF AMENDMENT REGARDING PRESSURE AND TEMPERATURE LIMIT CURVES (TAC NO. MB8750) (ML032310536)

7.11 BWRVIP-74-A: "BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal," EPRI, Palo Alto, CA: 2003. 1008872. EPRI PROPRIETARY INFORMATION

ATTACHMENT 2 TO NG-15-0235

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-144)
TO REVISE AND RELOCATE PRESSURE AND TEMPERATURE LIMITS CURVES TO A PRESSURE AND TEMPERATURE LIMITS REPORT

PROPOSED TECHNICAL SPECIFICATIONS CHANGES
(MARKUP COPY)

7 pages follow

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).
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.

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.

(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 ~~limits~~.

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. ---------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1 Initiate action to restore parameter(s) to within limits. AND C.2 Determine RCS is acceptable for operation.	Immediately Prior to entering MODE 2 or 3.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1 Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---------- ~~Verify:~~ ~~a. RCS pressure and RCS temperature are within the applicable limits in Figure 3.4.9-1.~~ ~~b. RCS heatup and cooldown rates are ≤ 20°F in any 1 hour period during inservice leak and hydrostatic testing (Curve A).~~ ~~c. RCS heatup and cooldown rates are ≤ 100°F in any 1 hour period during non-nuclear heating (Curve B) and nuclear heating (Curve C).~~	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 ~~Figure 3.4.9-1.~~ 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 ~~≤ 145°F~~ 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 ~~≤ 50°F~~ 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 ~~≥ 74°F~~. 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 ~~≥ 74°F~~. 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 ~~≥ 74°F~~. within the limits specified in the PTLR	In accordance with the Surveillance Frequency Control Program

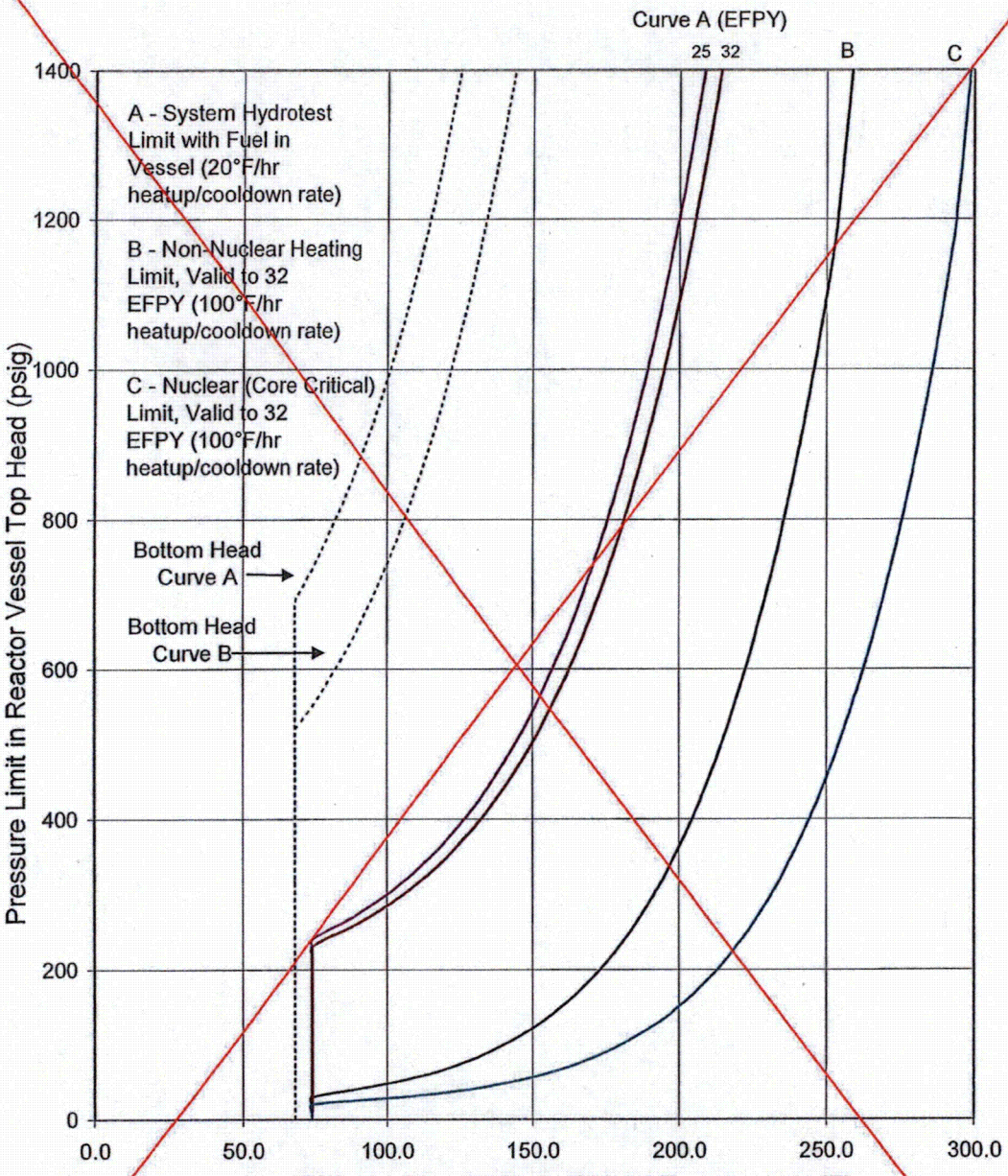

Figure 3.4.9-1 (page 1 of 1)

Pressure Versus Minimum Temperature Valid to Thirty-two Full Power Years, per Appendix G of 10CFR50

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:
   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.

ATTACHMENT 3 TO NG-15-0235

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-144)
TO REVISE AND RELOCATE PRESSURE AND TEMPERATURE LIMITS CURVES TO A PRESSURE AND TEMPERATURE LIMITS REPORT

REVISED TECHNICAL SPECIFICATIONS PAGES

8 pages follow

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.	30 minutes
	AND	
	A.2 Determine RCS is acceptable for continued operation.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 4.	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.

ATTACHMENT 4 TO NG-15-0235

NEXTERA ENERGY DUANE ARNOLD, LLC
DUANE ARNOLD ENERGY CENTER

LICENSE AMENDMENT REQUEST (TSCR-144)
TO REVISE AND RELOCATE PRESSURE AND TEMPERATURE LIMITS CURVES TO A PRESSURE AND TEMPERATURE LIMITS REPORT

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES
(FOR INFORMATION ONLY)

10 pages follow

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (Reference 7) → ~~Figure 3.4.9-1~~ contains P/T limit curves for heatup, cooldown, and inservice leakage and hydrostatic testing, and data for the maximum rate of change of reactor coolant temperature. The heatup curve provides limits for both heatup and criticality.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the Reactor Coolant Pressure Boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 5.

(continued)

BASES

BACKGROUND (continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limits include the Reference 1 requirement that they be at least 40°F above the heatup curve or the cooldown curve and not lower than the minimum permissible temperature for the inservice leakage and hydrostatic testing.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. ~~Reference 7 approved the curves and limits specified in this section.~~ Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T Limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

The elements of this LCO are:

a. RCS pressure ~~and~~ temperatures ~~are within the limits of the applicable curves of Figure 3.4.9-1~~ and heatup or cooldown rates are ≤ ~~100°F/hr during RCS heatup and cooldown, and ≤ 20°F/hr during pressure testing (e.g., hydrostatic testing). Note: The P/T limits and corresponding heatup/cooldown rates of either Curve A or B may be applied while achieving or recovering from test conditions. Curve A applies during pressure testing and when the limits of Curve B cannot be maintained~~;

b. The temperature difference between the reactor vessel bottom head coolant and the Reactor Pressure Vessel (RPV) coolant is ≤ ~~145°F~~ during recirculation pump startup;

within the limits specified in the PTLR

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is ≤ ~~50°F~~ during recirculation pump startup;

d. RCS pressure and temperature are within the criticality limits specified in ~~Figure 3.4.9-1~~ prior to achieving criticality; and

the PTLR,

e. The temperatures at the reactor vessel head flange and the shell adjacent to the head flange are ≥ ~~74°F~~ when tensioning the reactor vessel head bolting studs.

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

The rate of change of temperature limits control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and inservice leakage and hydrostatic testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

(continued)

BASES

LCO (continued)

Violation of the limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCS components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating pressure temperature regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The potential for violating a P/T limit exists at all times. For example, P/T limit violations could result from ambient temperature conditions that result in the reactor vessel metal temperature being less than the minimum allowed temperature for boltup. Therefore, this LCO is applicable even when fuel is not loaded in the core.

ACTIONS

A.1 and A.2

in the PTLR

Operation outside the P/T limits while in MODE 1, 2, or 3 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

Condition A is modified by a Note requiring Required Action A.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

There are no changes on this page; it is included for completeness only.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress, or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With the reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

Besides restoring the P/T limit parameters to within limits, an evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 212°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

Condition C is modified by a Note requiring Required Action C.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

Verification that operation is within limits is required periodically when RCS pressure and temperature conditions are undergoing planned changes. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, this Frequency permits a reasonable time for assessment and correction of minor deviations. ~~The limits of Figure 3.4.9-1 are met when operation is to the right of the applicable limit curve.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

~~NOTE: There are no changes on this page, it is included for completeness only.~~

SR 3.4.9.1 (continued)

Surveillance for heatup, cooldown, or inservice leakage and hydrostatic testing may be initiated and discontinued when the criteria given in the relevant plant procedure for starting and ending the activity are satisfied. During heatups and cooldowns, the temperatures at the reactor vessel shell adjacent to the shell flange, the reactor vessel bottom drain, recirculation loops A and B, and the reactor vessel bottom head shall be monitored. During inservice hydrostatic or leak testing, the reactor vessel metal temperatures at the outside surface of the bottom head in the vicinity of the control rod drive housing and reactor vessel shell adjacent to the shell flange shall be monitored.

This SR has been modified with a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leakage and hydrostatic testing.

SR 3.4.9.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical. ~~The limits of Figure 3.4.9-1 are met when operation is to the right of the applicable limit curve.~~

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

SR 3.4.9.3 and SR 3.4.9.4

Differential temperatures within the applicable limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation pump (Ref. 8) are satisfied.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.3 and SR 3.4.9.4 (continued)

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

For SR 3.4.9.3, an acceptable means of measuring Reactor Pressure Vessel (RPV) coolant temperature is by using the saturation temperature corresponding to reactor steam dome pressure.

Acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.4 include but are not limited to comparing the temperatures of the operating recirculation loop and the idle loop. The idle loop and RPV coolant temperature using saturation temperature corresponding to reactor steam dome pressure, or the idle loop and the bottom head coolant temperature with flow through the bottom head drain.

are acceptable means

SR 3.4.9.3 and SR 3.4.9.4 have been modified by a Note that requires the Surveillance to be met only in MODES 1, 2, 3, and 4 during a recirculation pump startup, since this is when the stresses occur. In MODE 5, the overall stress on limiting components is lower. Therefore, ΔT limits are not required.

SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7

Limits on temperature at the reactor vessel head flange and the shell adjacent to the head flange are generally bounded by the other P/T limits during system heatup and cooldown. However, operations approaching MODE 4 from MODE 5 and in MODE 4 with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

SR 3.4.9.5 requires that temperatures at the reactor vessel head flange and the shell adjacent to the head flange must be verified to be above the limits within the Surveillance Frequency before and while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When in MODE 4 with

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7 (continued)

RCS temperature ≤ 80°F, more frequent checks of the temperatures at the reactor vessel head flange and the shell adjacent to the head flange are required by SR 3.4.9.6 because of the reduced margin to the limits. When in MODE 4 with RCS temperature ≤ 100°F, monitoring of the temperatures at the reactor vessel head flange and the shell adjacent to the head flange are required periodically by SR 3.4.9.7 to ensure the temperatures are within the specified limits.

The Surveillance Frequencies are controlled under the Surveillance Frequency Control Program. The Frequency for SR 3.4.9.5 and SR 3.4.9.6 reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The Frequency for SR 3.4.9.7 is reasonable based on the rate of temperature change possible at these temperatures.

SR 3.4.9.5 is modified by a Note that requires the Surveillance to be performed only when tensioning the reactor vessel head bolting studs. However, per SR 3.0.4, the Surveillance needs to be met prior to tensioning, i.e., verified within the Surveillance Frequency prior to the start of tensioning. SR 3.4.9.6 is modified by a Note that requires the Surveillance to be initiated 30 minutes after RCS temperatures ≤ 80°F in Mode 4. SR 3.4.9.7 is modified by a Note that requires the Surveillance to be initiated 12 hours after RCS temperature
≤ 100°F in Mode 4. The Notes contained in these SRs are necessary to specify when the reactor vessel flange and head flange temperatures are required to be verified to be within the limits specified.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix G, December 1995

2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

3. ASTM E 185-82, July 1982.

4. 10 CFR 50, Appendix H.

5. Regulatory Guide 1.99, Revision 2, May 1988.

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

7. ~~D. Hood (NRC) to M. Peifer (NMC), TS Amendment No. 253 to Facility Operating License No. DPR-49, dated August 25, 2003.~~

8. UFSAR, Section 15.1.5.1.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)