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

July 10, 2020

Mr. Don Moul
Vice President, Nuclear Division and
Chief Nuclear Officer
NextEra Energy Duane Arnold, LLC
Mail Stop: NT3/JW
15430 Endeavor Drive
Jupiter, FL 33478

SUBJECT: DUANE ARNOLD ENERGY CENTER – ISSUANCE OF AMENDMENT NO. 311 RE: PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS (EPID L-2019-LLA-0130)

Dear Mr. Moul:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 311 to Renewed Facility Operating License No. DPR-49, for the Duane Arnold Energy Center. The amendment consists of changes to the license based on your application dated June 20, 2019, as supplemented by letters dated September 12, 2019, and November 4, 2019.

The amendment revises Renewed Facility Operating License No. DPR-49 and the associated technical specifications to Permanently Defueled Technical Specifications consistent with the permanent cessation of reactor operation and permanent defueling of the reactor.

A copy of our related safety evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Scott P. Wall, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures:
1. Amendment No. 311 to Renewed License No. DPR-49
2. Safety Evaluation

cc: Listserv

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. 311
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 June 20, 2019, as supplemented by letters dated September 12, 2019, and November 4, 2019, 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 Renewed Facility Operating License No. DPR-49 is hereby amended to read as follows:

- The title "RENEWED FACILITY OPERATING LICENSE" is to read "RENEWED FACILITY LICENSE"

- Paragraphs 1.B, 1.C, 1.D, 1.E, 1.G, 1.H, 1.I, 2.A, 2.B.(1), 2.B.(2), 2.B.(3), 2.B.(5), 2.C., 2.C.(1), 2.C.(2), 2.C.(2)(a), 2.C.(4), 2.C.(11), 2.C.(12), 2.C.(13), and 2.D are to read as follows:

1.B. Deleted;

1.C. The facility will be maintained in conformity with the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

1.D. There is reasonable assurance: (i) that the activities authorized by this renewed license can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

1.E. NextEra Energy Duane Arnold, LLC is technically qualified and NextEra Energy Duane Arnold, LLC, Central Iowa Power Cooperative and Corn Belt Power Cooperative are financially qualified to engage in the activities authorized by this renewed license in accordance with the rules and regulations of the Commission;

1.G. The issuance of this renewed license will not be inimical to the common defense and security or to the health and safety of the public;

1.H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of renewed Facility License No. DPR-49 is in accordance with 10 CFR Part 50, Appendix D, of the Commission's regulations and all applicable requirements of said Appendix D have been satisfied;

1.I. Deleted.

2. Renewed Facility License No. DPR-49 is hereby issued to NextEra Energy Duane Arnold, LLC, Central Iowa Power Cooperative (CIPCO) and Corn Belt Power Cooperative (Corn Belt) to read as follows:

2.A. This renewed license applies to the Duane Arnold Energy Center, a permanently defueled boiling water reactor and associated equipment (the facility), owned by NextEra Energy Duane Arnold, LLC, Central Iowa Power Cooperative and Corn Belt Power

Cooperative and operated by NextEra Energy Duane Arnold, LLC. The facility is located on NextEra Energy Duane Arnold, LLC's, Central Iowa Power Cooperative's and Corn Belt Power Cooperative's site near Palo in Linn County, Iowa. This site consists of approximately 500 acres adjacent to the Cedar River and is described in the "Final Safety Analysis Report" as supplemented and amended (Amendments 1 through 14) and the Environmental Report as supplemented and amended (Supplements 1 through 5).

2.B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

2.B.(1) NextEra Energy Duane Arnold, LLC, pursuant to Section104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess and use the facility as required for nuclear fuel storage; and CIPCO and Corn Belt to possess the facility at the designated location in Linn County, Iowa, in accordance with the procedures and limitations set forth in this license;

2.B.(2) NextEra Energy Duane Arnold, LLC, pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Updated Final Safety Analysis Report, as supplemented and amended as of June 1992 and as supplemented by letters dated March 26, 1993, and November 17, 2000.

2.B.(3) NextEra Energy Duane Arnold, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source or sealed sources for radiation monitoring equipment calibration, and to possess any byproduct, source and special nuclear material as sealed neutron sources previously used for reactor startup or reactor instrumentation; and fission detectors;

2.B.(5) NextEra Energy Duane Arnold, LLC, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear materials that were produced by the operation of the facility.

2.C. This renewed 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:

2.C.(1) Deleted

2.C.(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 311, are hereby incorporated in the license. NextEra Energy Duane Arnold, LLC shall maintain the facility in accordance with the Permanently Defueled Technical Specifications.

2.C.(2)(a) Deleted

2.C.(4) Deleted

2.C.(11) Deleted

2.C.(12) Deleted

2.C.(13) Deleted

2.D. The license is effective as of the date of issuance and is effective until the Commission notifies the licensee in writing that the license is terminated.

3. This license amendment is effective following the docketing of certifications required by 10 CFR 50.82(a)(1)(i) and (ii) that the Duane Arnold Energy Center has been permanently shut down and defueled. The amendment shall be implemented within 30 days of the effective date of the amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert F. Kuntz

Digitally signed by Robert F. Kuntz
Date: 2020.07.10 09:50:13 -04'00'

Nancy L. Salgado, Chief
Plant Licensing Branch III
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 10, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 311

DUANE ARNOLD ENERGY CENTER

RENEWED FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following pages of the Renewed Facility Operating License No. DPR-49; Appendix A, Technical Specifications; and Appendix B, Additional Conditions, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating area of change.

Renewed Facility Operating License No. DPR-49

REMOVE	INSERT
-1- through -5-	-1- through -5-
-7-	-7-

Appendix A, Technical Specifications

REMOVE	INSERT
TOC i through iii	TOC i
1.1-1 through 1.1-10	1.1-1
1.2-1 through 1.2-3	1.2-1
1.3-1 through 1.3-13	1.3-1
1.4-1 through 1.4-8	1.4-1 through 1.4-2
2.0-1	2.0-1
3.0-1 through 3.0-5	3.0-1 through 3.0-3
3.1-1 through 3.1-26	3.1-1
3.2-1 through 3.2-3	3.2-1
3.3-1 through 3.3-81	3.3-1
3.4-1 through 3.4-25	3.4-1
3.5-1 through 3.5-16	3.5-1
3.6-1 through 3.6-43	3.6-1
3.7-1 through 3.7-20	3.7-1 through 3.7-9
3.8-1 through 3.8-31	3.8-1
3.9-1 through 3.9-15	3.9-1
3.10-1 through 3.10-23	3.10-1
4.0-1 through 4.0-3	4.0-1 through 4.0-2
5.0-4 through 5.0-25	5.0-4 through 5.0-17

Appendix B, Additional Conditions

REMOVE	INSERT
-1-	-1-
-2-	--

NEXTERA ENERGY DUANE ARNOLD, LLC
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE
DOCKET 50-331
DUANE ARNOLD ENERGY CENTER
RENEWED FACILITY LICENSE

Renewed License No. DPR-49

1. The Nuclear Regulatory Commission (the Commission) having found that:

   A. The application for license filed by FPL Energy Duane Arnold, LLC,* Central Iowa Power Cooperative and Corn Belt Power Cooperative (the licensees) 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 and all required notifications to other agencies or bodies have been duly made;

   B. Deleted;

   C. The facility will be maintained in conformity with the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

   D. There is reasonable assurance: (i) that the activities authorized by this renewed license can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   E. NextEra Energy Duane Arnold, LLC is technically qualified and NextEra Energy Duane Arnold, LLC, Central Iowa Power Cooperative and Corn Belt Power Cooperative are financially qualified to engage in the activities authorized by this renewed license in accordance with the rules and regulations of the Commission;

   F. The licensees have satisfied the applicable provisions of 10 CFR Part 140, “Financial Protection Requirements and Indemnity Agreements,” of the Commission’s regulations;

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

   H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of renewed Facility License No. DPR-49 is in accordance with 10 CFR Part 50, Appendix D, of the Commission’s regulations and all applicable requirements of said Appendix D have been satisfied;

*On April 16, 2009, the name “FPL Energy Duane Arnold, LLC” was changed to “NextEra Energy Duane Arnold, LLC.”

I. Deleted.

2. Renewed Facility License No. DPR-49 is hereby issued to NextEra Energy Duane Arnold, LLC, Central Iowa Power Cooperative (CIPCO) and Corn Belt Power Cooperative (Corn Belt) to read as follows:

   A. This renewed license applies to the Duane Arnold Energy Center, a permanently defueled boiling water reactor and associated equipment (the facility), owned by NextEra Energy Duane Arnold, LLC, Central Iowa Power Cooperative and Corn Belt Power Cooperative and operated by NextEra Energy Duane Arnold, LLC. The facility is located on NextEra Energy Duane Arnold, LLC's, Central Iowa Power Cooperative's and Corn Belt Power Cooperative's site near Palo in Linn County, Iowa. This site consists of approximately 500 acres adjacent to the Cedar River and is described in the "Final Safety Analysis Report" as supplemented and amended (Amendments 1 through 14) and the Environmental Report as supplemented and amended (Supplements 1 through 5).

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

      (1) NextEra Energy Duane Arnold, LLC, pursuant to Section104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess and use the facility as required for nuclear fuel storage; and CIPCO and Corn Belt to possess the facility at the designated location in Linn County, Iowa, in accordance with the procedures and limitations set forth in this license;

      (2) NextEra Energy Duane Arnold, LLC, pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Updated Final Safety Analysis Report, as supplemented and amended as of June 1992 and as supplemented by letters dated March 26, 1993, and November 17, 2000.

      (3) NextEra Energy Duane Arnold, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source or sealed sources for radiation monitoring equipment calibration, and to possess any byproduct, source and special nuclear material as sealed neutron sources previously used for reactor startup or reactor instrumentation; and fission detectors;

      (4) NextEra Energy Duane Arnold, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated radioactive apparatus components;

      (5) NextEra Energy Duane Arnold, LLC, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear materials that were produced by the operation of the facility.

C. This renewed 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) Deleted

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 311, are hereby incorporated in the license. NextEra Energy Duane Arnold, LLC shall maintain the facility in accordance with the Permanently Defueled Technical Specifications.

(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.

Risk-Informed Changes that May Be Made Without Prior NRC Approval

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and maintained plant; and reflect the operating experience at the plant. Acceptable methods to assess the risk of the change may include methods that have been used in the peer-reviewed fire PRA model, methods that have been approved by NRC through a plant-specific license amendment or NRC approval of generic methods specifically for use in NFPA 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

(a) Prior NRC review and approval is not required for changes that clearly result in a decrease in risk. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

(b) Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10^{-7}/year (yr) for CDF and less than 1×10^{-8}/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

Other Changes that May Be Made Without Prior NRC Approval

1. Changes to NFPA 805, Chapter 3, Fundamental Fire Protection Program. Prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3, fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to NFPA 805, Chapter 3 element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

   The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3 elements are acceptable because the alternative is "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are as follows:

   - Fire Alarm and Detection Systems (Section 3.8);
   - Automatic and Manual Water-Based Fire Suppression Systems (Section 3.9);
   - Gaseous Fire Suppression Systems (Section 3.10); and,
   - Passive Fire Protection Features (Section 3.11).

   This License Condition does not apply to any demonstration of equivalency under Section 1.7 of NFPA 805.

2. Fire Protection Program Changes that Have No More than Minimal Risk Impact

   Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation report dated September 10, 2013 to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

Transition License Conditions

(1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) and (3) below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in (2) above.

(2) The licensee shall implement the modifications to its facility, as described in Enclosure 2, Attachment S, Table S-1, "Plant modifications Committed," of DAEC letter NG-13-0287, dated July 2, 2013, to complete the transition to full compliance with 10 CFR 50.48(c) by December 31, 2014. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

(3) The licensee shall implement the items listed in Enclosure 2, Attachment S, Table S-2, "Implementation Items," of DAEC letter NG-13-0287, dated July 2, 2013, by March 9, 2014.

(4) Deleted.

(5) Physical Protection

NextEra Energy Duane Arnold, LLC shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification,

(11) Deleted.

(12) Deleted.

(13) Deleted.

D. This license is effective as of the date of issuance and is effective until the Commission notifies the licensee in writing that the license is terminated.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by Eric J. Leeds

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Enclosures:

1. Appendix A Technical Specifications
2. Appendix B Additional Conditions

Date of Issuance: December 16, 2010

1.0 USE AND APPLICATION

1.1 Definitions

---NOTE---
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
CERTIFIED FUEL HANDLER	A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the Certified Fuel Handler training program required by Technical Specification Section 5.3.2.
NON-CERTIFIED OPERATOR	A NON-CERTIFIED OPERATOR is an individual who complies with the provisions of Technical Specifications Section 5.3.1.

1.0 USE AND APPLICATION

1.2 DELETED

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions for Operation (LCOs) specify minimum requirements for safely maintaining the facility. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Times(s).
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability.
IMMEDIATE COMPLETION TIME	When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR. The “specified Frequency” consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements. The use of “met” or “performed” in these instances conveys specific meanings. A Surveillance is “met” only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being “performed,” constitutes a Surveillance not “met.” “Performance” refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

(continued)

1.4 Frequency (continued)

EXAMPLE

The following example illustrates the manner in which Frequencies are specified.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified, then SR 3.0.3 becomes applicable.

2.0 DELETED

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1	LCOs shall be met during the specified conditions in the Applicability, except as provided in LCO 3.0.2.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met.
LCO 3.0.3	Deleted.
LCO 3.0.4	Deleted.
LCO 3.0.5	Deleted.
LCO 3.0.6	Deleted.
LCO 3.0.7	Deleted.
LCO 3.0.8	Deleted.
LCO 3.0.9	Deleted.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as “once,” the above interval extension does not apply.

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

3.0 SR APPLICABILITY (continued)

SR 3.0.4 Entry into a specified condition in the Applicability of an LCO shall only be made when the LCO's Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3.

This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with ACTIONS.

3.1 DELETED

3.2 DELETED

3.3 DELETED

3.4 DELETED

3.5 DELETED

3.6 DELETED

3.7 PLANT SYSTEMS

3.7.1 Deleted

3.7.2 Deleted

3.7.3 Deleted

3.7.4 Deleted

3.7.5 Deleted

3.7.6 Deleted

3.7 PLANT SYSTEMS

3.7.7 Deleted

3.7 PLANT SYSTEMS

3.7.8 Spent Fuel Storage Pool Water Level

LCO 3.7.8 The spent fuel storage pool water level shall be ≥ 36 ft.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies in the spent fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify the spent fuel storage pool water level is ≥ 36 ft.	In accordance with the Surveillance Frequency Control Program

3.7.9 Deleted

3.8 DELETED

3.9 DELETED

3.10 DELETED

4.0 DESIGN FEATURES

4.1 Site Location

The plant site, which consists of approximately 500 acres, is adjacent to the Cedar River approximately 2.5 miles northeast of the Village of Palo, Iowa. The boundary of the exclusion area defined in 10 CFR 100 is delineated by the property lines. The distance to the outer boundary of the low population zone is 6 miles. The plan of the site is shown on UFSAR Figures 1.2-1 and 1.2-2.

4.2 Deleted

4.3 Fuel Storage

4.3.1 Criticality

The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having the following limits for maximum k-infinity in the normal reactor core configuration at cold conditions and maximum lattice-average U-235 enrichment weight percent:

	k-∞	wt %
i) 7x7 and 8x8 pin arrays (Legacy Fuel Assemblies only; Holtec and PaR racks)	$\leq$ 1.29	$\leq$ 4.6
ii) 10x10 pin arrays (Holtec and PaR racks)	$\leq$ 1.29	$\leq$ 4.95

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in 9.1 of the UFSAR; and

c. A nominal 6.060 inches for HOLTEC designed and 6.625 inches for PaR designed center to center distance between fuel assemblies placed in the storage racks.

d. The Boral neutron absorber shall have a ^{10}B areal density greater than or equal to 0.0162 grams $^{10}B/cm^2$ with an uncertainty of 0.0012 grams $^{10}B/cm^2$.

(continued)

4.0 DESIGN FEATURES (continued)

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 831 ft. – 2 3/4 in.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2563 fuel assemblies in a vertical orientation, including no more than 152 fuel assemblies stored in the cask pit in accordance with UFSAR Section 9.1.

5.0 ADMINISTRATIVE CONTROLS

5.3 Facility Staff Qualifications

5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications referenced for comparable positions in ANSI/ANS 3.1-1978. The radiation protection manager shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

5.3.2 The CERTIFIED FUEL HANDLER shall be qualified to the NRC-approved training and retraining program for CERTIFIED FUEL HANDLERS. The NRC-approved training and retraining program for CERTIFIED FUEL HANDLERS shall be maintained.

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Quality assurance for effluent and environmental monitoring;

d. [Deleted]; and

e. All programs specified in Specifications 5.5.

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs shall be established, implemented and maintained.

5.5.1 Offsite Dose Assessment Manual (ODAM)

a. The ODAM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The ODAM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and Radioactive Material Release Report required by Specification 5.6.2 and Specification 5.6.3.

c. Licensee initiated changes to the ODAM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   a. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   b. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent dose or setpoint calculations;

2. Shall become effective after the approval of the plant manager; and

3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODAM as a part of or concurrent with the Radioactive Material Release Report for the period of the report in which any change in the ODAM was made. Each change shall be identified by markings in the margins of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

(continued)

5.5 Programs and Manuals (continued)

5.5.2 Deleted

5.5.3 Deleted

5.5.4 Radioactive Effluent Controls Program

This program, conforming to 10 CFR 50.36a, provides for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODAM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODAM;

b. Limitations on the concentrations of radioactive material released in liquid effluents from the site to unrestricted areas, conforming to ten times (10x) the concentrations listed in Appendix B, Table 2, Column 2 to 10 CFR 20.1001 – 20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents pursuant to 10 CFR 20.1302 and with the methodology and parameters in the ODAM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODAM at least every 31 days;

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems which were used to establish compliance with the design objectives in 10 CFR 50, Appendix I, Section II be used when specified to provide reasonable assurance that releases of radioactive material in liquid and gaseous effluents be kept as low as reasonably achievable;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

1. For noble gases: less than or equal to a dose rate of 500 mrem/yr to the whole body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and

2. For iodine-131, iodine-133, tritium, and for all radionuclides in particulate form with half lives > 8 days: less than or equal to a dose rate of 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

(continued)

5.5 Programs and Manuals (continued)

5.5.5 Deleted

5.5.6 Deleted

5.5.7 Deleted

5.5.8 Storage Tank Radioactivity Monitoring Program

This program provides controls for the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures".

The program shall include a surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is ≤ 50 curies, excluding tritium and dissolved or entrained noble gases. The liquid radwaste storage tanks in the Low-Level Radwaste Processing and Storage Facility are considered unprotected outdoor tanks.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.9 Deleted

5.5.10 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license; or

(continued)

5.5.10 Technical Specifications (TS) Bases Control Program (continued)

2. A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.10b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.11 Deleted

5.5.12 Deleted

5.5.13 Deleted

5.5.14 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Spent Fuel Pool Neutron Absorber Monitoring Program

This program provides routine monitoring and actions to ensure that the condition of Boral in the spent fuel pool racks is appropriately monitored to ensure that the Boral neutron attenuation capability described in the criticality safety analysis of UFSAR Section 9.1 is maintained. The program shall include the following:

a. Neutron attenuation in situ testing for the PaR racks shall be performed at a frequency of not more than 10 years, or more frequently based on observed trends or calculated projections of Boral degradation. The acceptance criterion for minimum Boral areal density will be that value assumed in the criticality safety analysis.

b. Neutron attenuation testing of a representative Boral coupon for the Holtec racks shall be performed at a frequency of not more than 6 years, or more frequently based on observed trends or calculated projections of Boral degradation. The acceptance criterion for minimum Boral density will be that value assumed in the criticality safety analysis.

c. Description of appropriate corrective actions for discovery on nonconforming Boral.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 DELETED

5.6.2 <u>Annual Radiological Environmental Operating Report</u>

The Annual Radiological Environmental Operating Report covering the operation of the facility during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Assessment Manual (ODAM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODAM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in Regulatory Guide 4.8. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

(continued)

5.6 Reporting Requirements (continued)

5.6.3 Radioactive Material Release Report

The Radioactive Material Release Report covering the operation of the facility during the previous calendar year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the facility. The material provided shall be consistent with the objectives outlined in the ODAM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 DELETED

5.6.5 DELETED

5.6.6 DELETED

5.6.7 DELETED

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously transmits dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

(continued)

5.7 High Radiation Area (continued)

(i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

(ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 <u>High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation</u>

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

(continued)

b. Access to, and activities in, each area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

   1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached with an appropriate alarm setpoint, or

   2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

   3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

      (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

      (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area.

(continued)

5.7 High Radiation Area (continued)

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the “As Low As is Reasonably Achievable” principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible light shall be activated at the area as a warning device.

APPENDIX B

ADDITIONAL CONDITIONS

LICENSE NO. DPR-49

NextEra Energy Duane Arnold, LLC (the term licensee in Appendix B refers to NextEra Energy Duane Arnold, LLC or prior license holders) shall comply with the following conditions on the schedule noted below:

Amendment Number	Additional Conditions	Implementation Date
223 275	NextEra Energy Duane Arnold, LLC is authorized to relocate certain requirements included in Appendix A to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's application dated October 30, 1996, as supplemented and consolidated in its March 31, 1998, submittal. These relocations were evaluated in the NRC staff's Safety Evaluation enclosed with this amendment.	This amendment is effective immediately and shall be implemented within 180 days of the date of this amendment.
260 (1) 275	NextEra Energy Duane Arnold shall take all necessary steps to ensure that the external trust fund is established at the time of the closing of the transfer of the license from Interstate Power (IPL) to FPLE Duane Arnold is maintained in accordance with the requirements of the December 23, 2005 order approving the license transfer, NRC regulations, and consistent with the safety evaluation supporting the order. The trust agreement shall be in a form acceptable to the NRC.	This amendment is effective immediately and shall be implemented within 30 days of the date of this amendment.
260 (2) 279	DELETED	
260 (3)	DELETED	

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 311

TO RENEWED 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 June 20, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19176A356), as supplemented by letters dated September 12, 2019, and November 4, 2019 (ADAMS Accession Nos. ML19261A141 and ML19308A085, respectively), NextEra Energy Duane Arnold, LLC (NEDA, or the licensee) requested changes to Renewed Facility Operating License (RFOL) No. DPR-49 and technical specifications (TSs) for the Duane Arnold Energy Center (DAEC). NEDA requested an amendment to revise the DAEC RFOL and the associated TS to Permanently Defueled Technical Specifications (PDTS) consistent with the permanent cessation of reactor operation and permanent defueling of the reactor.

The supplemental letter dated September 12, 2019, expanded the scope of the application as originally noticed and published in the *Federal Register* on August 29, 2019 (84 FR 45544). The supplement superseded, in their entirety, the attachments on the fuel handling accident (FHA) and associated TS changes contained in the June 20, 2019, submittal.

A revised proposed no significant hazards consideration was issued which replaced the original notice in its entirety. The supplemental letter dated November 4, 2019, provided additional information that clarified the application, did not expand the scope of the application as re-noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or Commission) staff's revised proposed no significant hazards consideration determination as published in the *Federal Register* on December 3, 2019 (84 FR 66232).

2.0 BACKGROUND

By letter dated January 18, 2019 (ADAMS Accession No. ML19023A196), NEDA submitted Notification of Permanent Cessation of Power Operations for DAEC. In this letter, NEDA notified the NRC of its intent to permanently cease operations at DAEC in the fourth quarter of 2020. By letter dated March 2, 2020 (ADAMS Accession No. ML20094F603), the licensee submitted its revised Notification of Permanent Cessation of Power Operations for DAEC. In this letter, NEDA notified the NRC of its intent to permanently cease operations at DAEC no later than October 30, 2020. After certifications of permanent cessation of power operations and permanent removal of fuel from the reactor vessel for DAEC are submitted in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Sections 82(a)(1)(i) and (ii), the 10 CFR Part 50 license will no longer authorize reactor operation or placement or retention of fuel in the reactor vessel.

By letter dated August 28, 2019 (ADAMS Accession No. ML19204A287), the NRC staff approved a Certified Fuel Handler (CFH) training and continuing training program for DAEC.

By letter dated January 2, 2020 (ADAMS Accession No. ML19310C204), the NRC staff issued Amendment No. 309 for DAEC. This amendment revised and removed certain requirements from the Section 5.0, "Administrative," portions of the DAEC TSs that are not applicable to the facility in a permanently defueled condition as well as revised and made editorial changes to Section 1.1, "Definitions."

By letter dated May 12, 2020 (ADAMS Accession No. ML20028F053), the NRC staff issued a partial exemption to eliminate the requirements to maintain records that will no longer be necessary once Duane Arnold Energy Center permanently shuts down and submits its certification for permanent fuel removal in accordance with 10 CFR 50.82(a)(1).

Once the certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel are submitted to the NRC pursuant to 10 CFR 50.82(a)(1)(i) and (ii), "Termination of license", the NRC regulations stipulated in 10 CFR 50.82(a)(2) will no longer authorize operation of the reactor or emplacement of fuel into the reactor vessel under the 10 CFR Part 50 license. In support of this condition, the DAEC RFOL and associated TSs are being proposed for revision to reflect the planned permanent shutdown and defueled condition in accordance with 10 CFR 50.51(b) and 10 CFR 50.36(c)(6).

Specifically, NEDA requested a revision to the 10 CFR Part 50 license and associated TSs consistent with the permanent cessation of reactor operations and the reduced energy contained in the stored nuclear fuel. The proposed license amendment, if approved, would implement a change to the DAEC TS to revise the license conditions, definitions, and TS sections to align with those required for the post defueled technical specifications (PDTS) that will reflect decommissioning requirements.

The existing DAEC TSs contain limiting conditions for operation (LCOs) that provide for appropriate functional capability of equipment required for safe operation of the facility, including the plant being in a defueled condition. Since the safety functions related to safe storage and management of spent fuel at an operating plant is similar to the corresponding function at a permanently defueled facility, the applicable existing TSs provide an appropriate level of control. However, the majority of the existing TSs are only applicable when the reactor is in an operational MODE. Once NEDA submits its certification of permanent cessation of operations and permanent removal of fuel from the reactor vessel for DAEC, consistent with 10 CFR 50.82(a)(2), the DAEC 10 CFR Part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel in the reactor vessel; therefore, the LCOs (and associated surveillance requirements (SRs)) that do not apply in a defueled condition are being proposed for deletion. The proposed amendment would revise the RFOL and associated TSs to reflect the permanent cessation of operations and the permanent removal of fuel from the reactor vessel at DAEC. In general, the changes would eliminate those TSs applicable in operating MODES; MODES where fuel is emplaced in the reactor vessel, and certain TSs required for movement of spent fuel assemblies. Changes were also proposed to TS definitions, administrative controls, and related to programs and procedures. The proposed amendment would also revise the RFOL to clarify or remove certain conditions no longer relevant to the permanently shutdown and defueled condition and would add conditions consistent with other permanently shutdown and defueled reactors.

3.0 REGULATORY EVALUATION

3.1 Regulatory Requirements

Section 182a of the Atomic Energy Act of 1954, as amended, requires applicants for nuclear power plant operating licenses to include TSs as part of the application. The NRC's regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36, “Technical specifications.” Pursuant to 10 CFR 50.36, each operating license issued by the Commission includes TSs and includes items in the following categories: (1) safety limits (SLs), limiting safety systems settings and control settings, (2) LCOs, (3) SRs, (4) design features, (5) administrative controls, (6) decommissioning, (7) initial notification, and (8) written reports.

Section 50.36 of 10 CFR provides four criteria to define the scope of equipment and parameters to be included in the TS LCOs. These criteria were developed for licenses authorizing operation (i.e., operating reactors) and focus on instrumentation to detect degradation of the reactor coolant system (RCS) pressure boundary and process variables, design features, operating restrictions, or structures, systems, or components (SSCs) that affect the integrity of fission product barriers during design-basis accidents (DBAs) or transients. They also focus on SSCs which operating experience or probabilistic risk assessment have shown to be significant to public health and safety. A general discussion of how these criteria were evaluated to ensure that the TS LCOs proposed for deletion are no longer required to be included in TSs, is provided below

Criterion 1 of 10 CFR 50.36(c)(2)(ii)(A) states that TS LCOs must be established for “installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.” Once no fuel is present in the reactor or RCS at the DAEC, this criterion is not applicable.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) states that TS LCOs must be established for a “process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.” The purpose of this criterion is to capture those process variables that have initial values assumed in the DBA and transient analyses, and which are monitored and controlled during power operation. The scope of DBAs applicable to a permanently shutdown and defueled reactor is reduced from those postulated for an operating reactor, and most TSs satisfying Criterion 2 are no longer applicable.

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) states that TS LCOs must be established for an SSC “that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.” The intent of this criterion is to capture into TSs those SSCs that are part of the primary success path of a safety sequence analysis. Also captured by this criterion are those support and actuation systems that are necessary for items in the primary success path to successfully function. The primary success path of a safety sequence analysis consists of the combination and sequences of equipment needed to operate (including consideration of the single failure criterion), so that the plant response to DBAs and transients limits the consequences of these events to within the appropriate acceptance criteria. There are no transients that continue to apply to permanently shutdown and defueled reactors. The scope of applicable DBAs that continue to apply to DAEC is discussed in more detail in Section 4.0 of this SE.

Criterion 4 of 10 CFR 50.36(c)(2)(ii)(D) states that TS LCOs must be established for SSCs “which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.” The intent of this criterion is that risk insights and operating experience be factored into the establishment of TS LCOs.” There are no longer any DBAs at DAEC in a permanently shutdown and defueled condition that can result in a significant offsite radiological risk to public health and safety.

Section 50.36(c)(3) of 10 CFR, *Surveillance requirements,* states, “Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.”

Section 50.36(c)(4) of 10 CFR, *Design features,* states, “Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section.”

Section 50.36(c)(5) of 10 CFR, *Administrative controls,* states, in part, “Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.”

Section 50.36(c)(6) of 10 CFR, *Decommissioning*, states that TS involving safety limits, limiting safety system settings, and limiting control system settings; LCOs; SRs; design features; and administrative controls for decommissioning facilities will be developed on a case-by-case basis.

Section 50.60 of 10 CFR, “Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation,” states that: (a) Except as provided in paragraph (b) of this section, all light-water nuclear power reactors, other than reactor facilities for which the certifications required under § 50.82(a)(1) have been submitted, must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H to this part. (b) Proposed alternatives to the described requirements in Appendices G and H of this part or portions thereof may be used when an exemption is granted by the Commission under § 50.12.

Appendix G to 10 CFR Part 50, “Fracture Toughness Requirements,” states, in part, that:

> This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.
>
> The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the ASME [American Society of Mechanical Engineers] Code, supplemented by the additional requirements set forth below, for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. Reactor vessels may continue to be operated only for

> that service period within which the requirements of this section are satisfied. For the reactor vessel beltline materials, including welds, plates and forgings, the values of RTNDT and Charpy upper-shelf energy must account for the effects of neutron radiation, including the results of the surveillance program of appendix H of this part. The effects of neutron radiation must consider the radiation conditions (i.e., the fluence) at the deepest point on the crack front of the flaw assumed in the analysis.

Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements," states, in part, that:

> The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in section IV of appendix G to part 50.

The NRC staff notes that in the course of this evaluation, information contained in DRAFT NUREG-1625, "Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants," March 1998, was also considered. This draft NUREG provides examples of TSs that the staff found acceptable during previous TS reviews for permanently shutdown and defueled reactors. The NRC staff also considered information contained in NUREG-1431, "Standard Technical Specifications, General Electric Plants (BWR/4)."

3.2 Radiological Consequences from Design-Basis Accidents

Radiological accidents considered in licensing nuclear power plants are classified as DBAs and severe (beyond design basis) accidents. DBAs are those accidents that both the licensee and the NRC staff evaluate to ensure that the plant can withstand normal and abnormal transients and a broad spectrum of postulated accidents without undue hazard to the health and safety of the public. Severe accidents are those that are beyond the design basis of the plant. They are more severe than DBAs because they may result in substantial damage to the fuel, whether or not there are serious offsite consequences. For the most part, DBAs focus on reactor operation and are not applicable to plants undergoing decommissioning. The only DBAs or severe accidents applicable to a decommissioning plant are typically those involving the spent fuel pool (SFP). These postulated accidents are not expected to occur during the life of the plant, but are evaluated to establish the design basis for the preventive and mitigative safety systems of the spent fuel storage facility.

Regulations governing accidents that must be addressed by nuclear power facilities, both operating and shutdown, are found in 10 CFR Part 50 and 10 CFR Part 100. The environmental impacts of DBAs, including those associated with the SFP, are evaluated during the initial licensing process. The ability of the plant to withstand these accidents is demonstrated to be acceptable before issuance of the operating license. The results of these evaluations are found in license documentation, such as the staff's safety evaluation report, the final environmental statement, and in the licensee's Updated Final Safety Analysis Report (UFSAR) or equivalent. The consequences for these events are evaluated for the hypothetical maximally exposed individual. The licensee is required to maintain the acceptable design and performance criteria throughout the life of the plant.

The NRC staff evaluated the radiological consequences of the postulated FHA DBA against the dose criteria specified in 10 CFR 50.67, "Accident source term," and using the guidance described in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (ADAMS Accession No. ML003716792). The RG 1.183 provides guidance to licensees on acceptable application of alternate source term (AST) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

On April 16, 2001, the NRC issued Amendment No. 237 to RFOL No. DPR-49 for DAEC (ADAMS Accession No. ML011070147), which implemented an alternative source term (AST) per 10 CFR 50.67 to perform the radiological consequence analysis of the design-basis FHA to support changes to the TSs. The analysis of the FHA that supported these changes assumed the FHA occurred over the reactor core 60 hours after reactor shutdown from full power and assumed control building emergency ventilation (CBEV) functions to mitigate dose consequences to control room occupants. This analysis did not credit secondary containment, secondary containment isolation or filtration by the standby gas treatment (SBGT) system.

The FHA-specific dose acceptance criteria are specified in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000 (ADAMS Accession No. ML003734190).

The dose acceptance criteria for the FHA are a total effective dose equivalent (TEDE) of 6.3 roentgen equivalent man (rem) at the exclusion area boundary (EAB) for the worst 2 hours, 6.3 rem at the outer boundary of the low population zone (LPZ), and 5 rem in the control room (CR) for the duration of the accident.

The regulations in 10 CFR 50.67 state, in part, that the NRC may issue the amendment only if the licensee's analysis demonstrates with reasonable assurance that:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [Sievert] (25 rem) total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

Appendix A to 10 CFR Part 50, "General Design Criteria [GDC] for Nuclear Power Plants," Criterion 19, "Control room," states, in part:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe

> condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

The emergency planning requirements of 10 CFR 50.47, “Emergency plans,” and Appendix E to 10 CFR Part 50, “Emergency Planning and Preparedness for Production and Utilization Facilities,” continue to apply to a nuclear power reactor after permanent cessation of operations and removal of fuel from the reactor vessel. There are no explicit regulatory provisions distinguishing emergency planning requirements for a power reactor that has been permanently shut down from those for an operating power reactor. The NRC staff notes that the risk of an offsite radiological release is significantly lower and the types of possible accidents are significantly fewer at a nuclear power reactor that has permanently ceased operations and removed fuel from the reactor vessel than at an operating power reactor.

Nuclear Energy Institute (NEI) topical report NEI 99-01, “Development of Emergency Action Levels for Non-Passive Reactors,” Revision 6 (ADAMS Accession No. ML12326A805), provides guidance for the development of emergency action levels (EALs) for reactors in a permanently defueled condition. The NEI 99-01 topical report was endorsed by the NRC in a letter dated March 28, 2013 (ADAMS Accession No. ML12346A463). Revision 6 of NEI 99-01 states that the accident analysis necessary to adopt the permanently defueled EAL scheme must confirm that the source terms and release motive forces are not sufficient to warrant classification of a site area emergency (SAE) or general emergency. An SAE would be declared for any events where exposure levels beyond the site area boundary are expected to exceed 10 percent of the Environmental Protection Agency (EPA) Protective Action Guides (PAGs). The EPA PAG for sheltering or evacuation of the public is a projected dose of one to five rem total effective dose (TED[1]) in 4 days. In addition, the EPA PAG for recommending the administration of potassium iodide (KI) (as a thyroid blocking agent) is a projected dose of 5 rem to the child thyroid from radioactive iodine. Correspondingly, NEI 99-01 established the SAE classification threshold as 100 millirem (mrem) TEDE or 500 mrem thyroid committed dose equivalent.

RG 1.183 provides the methodology for analyzing the radiological consequences of several DBAs to show compliance with 10 CFR 50.67. RG 1.183 provides guidance to licensees on acceptable application of AST submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

SRP Section 15.0.1 provides review guidance to the staff for the review of AST amendment requests. Section 15.0.1 states that the NRC reviewer should evaluate the proposed change against the guidance in RG 1.183. The dose acceptance criteria for the FHA are a TEDE of 6.3 rem at the EAB for the worst 2 hours, 6.3 rem at the outer boundary of the LPZ, and 5 rem in the CR for the duration of the accident.

Regulatory Issue Summary (RIS) 2006-04, “Experience with Implementation of Alternative Source Terms,” dated March 7, 2006 (ADAMS Accession No. ML053460347), discusses

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[1] For the purposes of this safety evaluation, the terms “TED” and “TEDE” are used interchangeably as both describing the combined effects of internal and external radiation exposure.

experiences with analyzing an accident involving a release from off-gas or waste systems. As part of full AST implementation, some licensees have included an accident involving a release from their off-gas or waste gas system. For this type of accident, licensees have proposed acceptance criteria of 500 mrem TEDE. The acceptance criterion for this event is that associated with the dose to an individual member of the public as described in 10 CFR Part 20, "Standards for Protection Against Radiation." When the NRC revised 10 CFR Part 20 to incorporate a TEDE dose, the offsite dose to an individual member of the public was changed from 500 mrem whole body to 100 mrem TEDE. Therefore, any licensee who chooses to implement AST for an off-gas or waste gas system release should base its acceptance criteria on 100 mrem TEDE. Licensees may also choose not to implement AST for this accident and continue with their existing analysis and acceptance criteria of 500 mrem whole body.

Branch Technical Position 11-5, "Postulated Radioactive Release Due to a Waste Gas System Leak or Failure," of SRP Chapter 11, "Radioactive Waste Management," provides guidance to the reviewer for assessing the analysis of an accidental release from the waste gas system.

3.3 Spent Fuel Pool Criticality

GDC 61, "Fuel storage and handling and radioactivity control," which states that "These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety."

GDC 62, "Prevention of criticality in fuel storage and handling," requires that, "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations."

Per 10 CFR 50.68(a), each holder of an operating license shall comply with either 10 CFR 70.24 or the requirements in 10 CFR 50.68(b). The licensee has elected to meet 10 CFR 50.68(b) and, accordingly, must comply with the following requirements:

(1) Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

(2) If no credit for soluble boron is taken, the estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

The regulations in 10 CFR 50.36(b) require TSs to be derived from the analyses and evaluation included in the safety analysis report and amendments thereto. As required by 10 CFR 50.36(c)(4), the TSs will include design features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c)(1), (2), and (3) of 10 CFR 50.36.

4.0 TECHNICAL EVALUATION

4.1 Accident Analysis

During normal power reactor operations, the forced inlet flow of water through the RCS removes the heat from the reactor by generating steam. The steam system, operating at high temperatures and pressures, transfers this heat to the turbine generator. The most severe postulated accidents for nuclear power plants involve damage to the nuclear reactor core and the release of large quantities of fission products to the RCS. Many of the accident scenarios postulated in the UFSAR involve failures or malfunctions of systems, which could affect the reactor core. With the termination of reactor operations at DAEC and the permanent removal of the fuel from the reactor core, most of the DBAs postulated in the UFSAR will no longer be possible. The spent fuel will be stored in the SFP and the Independent Spent Fuel Storage Installation (ISFSI). The reactor, RCS, steam system, and turbine generator are no longer in operation and have no function related to the storage of the spent fuel. Therefore, the postulated accidents involving failure or malfunction of the reactor, RCS, steam system, or turbine generator are no longer applicable.

Chapter 15 of the UFSAR describes the safety analysis aspects of the DAEC that were evaluated to demonstrate the plant could be operated safely and radiological consequences from postulated accidents do not exceed regulatory limits. The spectrum of Chapter 15 events is divided into three classes: transients, accidents, and special events. Table 2.1 from the LAR provides a list of the UFSAR Chapter 15 events and whether each applies to the permanently defueled condition. The licensee has stated, and the NRC staff agrees, that while spent fuel remains in the SFP, the accident that remain applicable to DAEC in the permanently shut down and defueled condition is the FHA within the SFP.

4.1.1 FHA Analysis

After the reactor has been completely defueled following permanent shutdown, an FHA in the reactor cavity is no longer a credible accident. The DBA FHA in the SFP is applicable when DAEC is in a permanently shut down and defueled condition. The licensees' analysis applied the accident source term guidelines outlined in RG 1.183 and was performed to determine the dose to the public at the exclusion area boundary as a function of time after shutdown. The analysis assumes, in part: a period of decay of 19 days after shutdown; and, no credit for mitigating safety systems.

The FHA is defined as the dropping of a single spent fuel assembly in the SFP during fuel handling activities, such that 151 fuel rods in the assembly suffer mechanical damage to the cladding. The gap activity in the damaged rods is instantaneously released into the SFP. The release occurs under 23 feet of water, which acts as a filter. TS LCO 3.7.8 will ensure the minimum water level (36 feet or greater) in the SFP is established prior to fuel handling and maintained. The proposed changes would not take effect until DAEC has certified to the NRC that it has permanently ceased operation, entered a permanently defueled condition, and a period of 19 days has transpired since shutdown.

The UFSAR-described FHA analyses for DAEC shows that, provided the SFP water level requirement of TS LCO 3.7.8 is met, the dose consequences are acceptable without relying on secondary containment or the SGTS system. NEDA performed a supplemental analysis and determined that, following a decay period of 19 days, CBEV is also not required to maintain FHA dose consequences for CR occupants below the acceptance criteria of 10 CFR 50.67(b)(2)(iii), justifying deletion of the TSs associated with CBEV.

In performing this review, the NRC staff relied upon information provided by the licensee and NRC staff experience in performing similar reviews. The staff concludes that the FHA consequences submitted for the permanently defueled DAEC meet the applicable radiological dose criteria specified in 10 CFR 50.67 at the exclusion area boundary, low population zone, and CR. The licensee's analysis did not credit secondary containment, secondary containment isolation, or filtration by the SBGT system. The analysis determined that following a 19-day decay period, CBEV is not required to maintain dose consequences for CR occupants within the criteria of 10 CFR 50.67(b)(2)(iii). Consequently, TS, LCOs, and SRs associated with CBEV and support equipment were proposed for deletion in TS by the licensee.

4.1.2 Spent Fuel Criticality Analysis

Although reactor operations at DAEC will cease, and the reactor will be defueled, the licensee will still store spent fuel in the DAEC SFP. The licensee credits use of a neutron absorbing material (NAM) in its SFP criticality safety analysis and, therefore, the NRC staff has reviewed the program in place to monitor the condition of the NAM.

DAEC uses two types of racks to store spent fuel in the SFP. Both rack designs credit the ability of the Boral NAM to absorb neutrons consistent with the licensee's criticality safety analysis. In order to ensure the Boral doesn't experience degradation that would impact its ability to absorb neutrons, the licensee established a monitoring program that covers both rack designs. This monitoring program contains provisions to measure the Boron-10 (^{10}B) areal density (AD) of the Boral (as ^{10}B performs the neutron attenuation function) and take corrective actions if degradation is detected. The monitoring program is described further in TS 5.5.15, "Spent Fuel Pool Neutron Absorber Monitoring Program," and the requirement for the minimum ^{10}B AD is found in TS 4.3.1, "Criticality."

During the period in which DAEC has ceased operations and is permanently defueled, there may be spent fuel in the SFP which requires criticality control. In order to ensure 10 CFR 50.68 is met, in part, the condition of the NAM is monitored to ensure that it does not degrade below the minimum ^{10}B AD that the licensee assumes in its criticality safety analysis.

In 2017, the NRC staff reviewed and approved a LAR for DAEC (ADAMS Accession No. ML17072A232), which incorporated its NAM monitoring program into the plant TS. As described below, the current amendment did not propose any changes to the NAM monitoring program that was approved in the 2017 LAR.

4.2 Proposed Changes to RFOL

4.2.1 License Title

The current license title is "Renewed Facility Operating License."

The licensee proposed to delete "Operating" from the title, so that it reads: "Renewed Facility License."

The proposed change to the title to delete "Operating" would provide a more accurate description of the facility during the permanently shutdown and defueled condition. Once the certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Therefore, the proposed change is consistent with 10 CFR 50.82(a)(2) and is acceptable to the NRC staff.

4.2.2 License Condition 1.B

Currently, License Condition 1.B reads:

> Construction of the Duane Arnold Energy Center (facility) has been substantially completed in conformity with Construction Permit No. DPPR-70; the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

The licensee proposes to delete License Condition 1.B because the decommissioning of DAEC does not depend on the conformity with Construction Permit No. DPPR-70. On February 22, 1974 (ADAMS Accession No. ML021860274), the Atomic Energy Commission issued Facility Operating License No. DPR-49 to the licensee. Construction Permit No. DPPR-70 was superseded by Facility Operating License No. DPR-49, which eventually became Renewed Facility Operating License (NFOL) No. DPR-49, dated December 16, 2010 (ADAMS Accession No. ML110980120). Therefore, the NRC staff finds it acceptable to delete License Condition 1.B.

4.2.3 License Condition 1.C

Currently, License Condition 1.C reads:

> The facility will operate in conformity with the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

The licensee proposes License Condition 1.C to read:

> The facility will be maintained in conformity with the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

The proposed language for License Condition 1.C is revised to align with the permanently defueled condition. Once the certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Thus, removal of the reference to operating provides accuracy in the 10 CFR Part 50 license description. Therefore, the changes are consistent with the requirements associated with a permanently shut down and defueled condition and, therefore, the NRC staff approves the revision to the License Condition 1.C.

4.2.4 License Condition 1.D

Currently, License Condition 1.D reads:

> There is reasonable assurance: (i) that the activities authorized by this renewed operating license can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

The licensee proposes License Condition 1.D to read:

> There is reasonable assurance: (i) that the activities authorized by this renewed license can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

The revision to License Condition 1.D eliminates the reference to “operating.” Once certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Therefore, the NRC staff approves of this change to the License Condition 1.D.

4.2.5 License Condition 1.E

Currently, License Condition 1.E reads:

> NextEra Energy Duane Arnold, LLC is technically qualified and NextEra Energy Duane Arnold, LLC, Central Iowa Power Cooperative and Corn Belt Power Cooperative are financially qualified to engage in the activities authorized by this renewed operating license in accordance with the rules and regulations of the Commission;

The licensee proposes License Condition 1.E to read:

> NextEra Energy Duane Arnold, LLC is technically qualified and NextEra Energy Duane Arnold, LLC, Central Iowa Power Cooperative and Corn Belt Power Cooperative are financially qualified to engage in the activities authorized by this renewed license in accordance with the rules and regulations of the Commission;

The revision to License Condition 1.E eliminates the reference to “operating.” Once certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Therefore, the NRC staff approves of this change to the License Condition 1.E.

4.2.6 License Condition 1.G

Currently, License Condition 1.G reads:

> The issuance of this renewed operating license will not be inimical to the common defense and security or to the health and safety of the public;

The licensee proposes License Condition 1.G to read:

> The issuance of this renewed license will not be inimical to the common defense and security or to the health and safety of the public;

The revision to License Condition 1.G eliminates the reference to "operating." Once certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Therefore, the NRC staff approves of this change to the License Condition 1.E.

4.2.7 License Condition 1.H

Currently, License Condition 1.H reads:

> After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of renewed Facility Operating License No. DPR-49 is in accordance with 10 CFR Part 50, Appendix D, of the Commission's regulations and all applicable requirements of said Appendix D have been satisfied;

The licensee proposes License Condition 1.H to read:

> After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the issuance of renewed Facility License No. DPR-49 is in accordance with 10 CFR Part 50, Appendix D, of the Commission's regulations and all applicable requirements of said Appendix D have been satisfied;

The revision to License Condition 1.H eliminates the reference to "operating." Once certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Therefore, the NRC staff approves of this change to the License Condition 1.H.

4.2.8 License Condition 1.I

Currently, License Condition 1.I reads:

> The receipt, possession, and use of source, by-product and special nuclear material as authorized by this renewed operating license will be in accordance with the Commission's regulations in 10 CFR Part 30 and 70, including 10 CFR Section 30.33, 70.23 and 70.31.

The licensee proposes to delete License Condition 1.I since the DAEC license will no longer authorize operation of the facility pursuant to 10 CFR 50.82(a)(2). The Commission's finding regarding possession and use of byproduct, source, and special nuclear material is not dependent on decommissioning of the facility. Additionally, possession and use of byproduct, source, and special nuclear material at DAEC during decommissioning activities is covered by License Condition 2.B, which will remain in effect. Therefore, License Condition 1.I is not needed. Therefore, the NRC staff approves deletion of the License Condition 1.I.

4.2.9 <u>License Item 2</u>

<u>Current</u>

> Renewed Facility Operating License No. DPR-49 is hereby issued to NextEra Energy Duane Arnold, LLC, Central Iowa Power Cooperative (CIPCO) and Corn Belt Power Cooperative (Corn Belt) to read as follows:

<u>Proposed</u>

> Renewed Facility License No. DPR-49 is hereby issued to NextEra Energy Duane Arnold, LLC, Central Iowa Power Cooperative (CIPCO) and Corn Belt Power Cooperative (Corn Belt) to read as follows:

The word “operating” is removed in the proposed revision above. After certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2).

4.2.10 License Condition 2.A

Currently, License Condition 2.A reads:

> Renewed Facility Operating License No. DPR-49 is hereby issued to NextEra Energy Duane Arnold, LLC, Central Iowa Power Cooperative (CIPCO) and Corn Belt Power Cooperative (Corn Belt) to read as follows:
>
> A. This renewed operating license applies to the Duane Arnold Energy Center, a boiling water reactor and associated equipment (the facility), owned by NextEra Energy Duane Arnold, LLC, Central Iowa Power Cooperative and Corn Belt Power Cooperative and operated by NextEra Energy Duane Arnold, LLC. The facility is located on NextEra Energy Duane Arnold, LLC's, Central Iowa Power Cooperative's and Corn Belt Power Cooperative's site near Palo in Linn County, Iowa. This site consists of approximately 500 acres adjacent to the Cedar River and is described in the “Final Safety Analysis Report” as supplemented and amended (Amendments 1 through 14) and the Environmental Report as supplemented and amended (Supplements 1 through 5).

The licensee proposes License Condition 2.A to read:

> Renewed Facility License No. DPR-49 is hereby issued to NextEra Energy Duane Arnold, LLC, Central Iowa Power Cooperative (CIPCO) and Corn Belt Power Cooperative (Corn Belt) to read as follows:
>
> A. This renewed license applies to the Duane Arnold Energy Center, a permanently defueled boiling water reactor and associated equipment (the facility), owned by NextEra Energy Duane Arnold, LLC, Central Iowa Power Cooperative and Corn Belt Power Cooperative and operated by NextEra Energy Duane Arnold, LLC. The facility is located on NextEra Energy Duane Arnold, LLC’s, Central Iowa Power Cooperative’s and Corn Belt Power Cooperative’s site near Palo in Linn County, Iowa. This

site consists of approximately 500 acres adjacent to the Cedar River and is described in the "Final Safety Analysis Report" as supplemented and amended (Amendments 1 through 14) and the Environmental Report as supplemented and amended (Supplements 1 through 5).

Once the certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50, the license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). The removal of the word "operating" provides accuracy in the 10 CFR Part 50 license description. Moreover, addition of the words "permanently defueled" for the boiling water reactor provides accurate description of the status of the reactor. Therefore, the NRC staff finds the revision to the License Condition 2.A to be acceptable.

4.2.11 License Condition 2.B.(1)

Currently, License Condition 2.B.(1) reads:

(1) NextEra Energy Duane Arnold, LLC, pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use and operate the facility; and CIPCO and Corn Belt to possess the facility at the designated location in Linn County, Iowa, in accordance with the procedures and limitations set forth in this license;

The licensee proposes License Condition 2.B.(1) to read:

(1) NextEra Energy Duane Arnold, LLC, pursuant to Section 104b of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, and use the facility as required for nuclear fuel storage; and CIPCO and Corn Belt to possess the facility at the designated location in Linn County, Iowa, in accordance with the procedures and limitations set forth in this license;

The proposed language change associated with operating the facility in this license condition is removed. After certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). This item is revised to align with the permanently defueled condition, and therefore NRC staff finds this license condition to be acceptable. Therefore, the NRC staff finds revision to the License Condition 2.B.(1) to be acceptable.

4.2.12 License Condition 2.B.(2)

Currently, License Condition 2.B.(2) reads:

(2) NextEra Energy Duane Arnold, LLC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended as of June 1992 and as supplemented by letters dated March 26, 1993, and November 17, 2000.

The licensee proposes License Condition 2.B.(2) to read:

(2) NextEra Energy Duane Arnold, LLC, pursuant to the Act and 10 CFR Part 70, to possess at any time special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, as described in the Updated Final Safety Analysis Report, as supplemented and amended as of June 1992 and as supplemented by letters dated March 26, 1993, and November 17, 2000.

The proposed language for this license condition reflects a condition in which the special nuclear material for fuel used for reactor operations can no longer be received, only stored. Once the certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Therefore, the changes are consistent with the requirements associated with a permanently shutdown and defueled condition, and the NRC staff approves the proposed change to License Condition 2.B.(2).

4.2.13 License Condition 2.B.(3)

Currently, License Condition 2.B.(3) reads:

(3) NextEra Energy Duane Arnold, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

The licensee proposes License Condition 2.B.(3) to read:

(3) NextEra Energy Duane Arnold, LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source or sealed sources for radiation monitoring equipment calibration, and to possess any byproduct, source and special nuclear material as sealed neutron sources previously used for reactor startup or reactor instrumentation; and fission detectors;

The licensee proposes to delete the requirements regarding receipt of sealed neutron sources for reactor startup and nuclear instrumentation from this license condition. This license condition is revised to reflect authorization only for continued possession of those sources previously used for reactor startups, produced as a byproduct, and those required for calibrations. Once certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Therefore, the use of startup sources will no longer be needed. The changes are consistent with the requirements associated with a permanently shut down and defueled condition. The use of sources for radiation monitoring will continue to be required. Therefore, the NRC staff approves of the revision to the License Condition 2.B.(3).

4.2.14 License Condition 2.B.(5)

Currently, License Condition 2.B.(5) reads:

> (5) NextEra Energy Duane Arnold, LLC, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

The licensee proposes License Condition 2.B.(5) to read:

> (5) NextEra Energy Duane Arnold, LLC, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear materials that were produced by the operation of the facility.

The licensee proposes to revise this license condition to replace “as may be” with “that were.” This revision allows possession of byproduct and special nuclear materials that were produced by operation of the DAEC reactor. Once certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Since the changes are consistent with the requirements of associated with a permanently shut down and defueled condition, the NRC staff approves of the modification to the License Condition 2.B.(3).

4.2.15 License Condition 2.C

Currently, License Condition 2.C reads

> 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:

The licensee proposes License Condition 2.C to read:

> C. This renewed 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:

The licensee proposes to delete License Condition 2.C to reflect the permanently defueled condition of the facility. Once certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor pursuant to 10 CFR 50.82(a)(2). The license need not limit reactor core power levels since the reactor will not be authorized to operate. Therefore, the NRC staff approves revision to the License Condition 2.C.

4.2.16 License Condition 2.C.(1)

Currently, License Condition 2.C.(1) reads:

> (1) 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).

The licensee proposes to delete this license condition in its entirety to reflect the permanently defueled condition of the facility. Once certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor pursuant to 10 CFR 50.82(a)(2). The license need not limit reactor core power levels since the reactor will not be authorized to operate. Therefore, the NRC staff approves deletion of the License Condition 2.C.(1).

4.2.17 License Condition 2.C.(2)

Currently, License Condition 2.C.(2) reads:

> (2) <u>Technical Specifications</u>
>
> The Technical Specifications contained in Appendix A, as revised through Amendment No. 309, 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.

The licensee proposes License Condition 2.C.(2) to read:

> (2) <u>Technical Specifications</u>
>
> The Technical Specifications contained in Appendix A, as revised through Amendment No. 311, are hereby incorporated in the license. NextEra Energy Duane Arnold, LLC shall maintain the facility in accordance with the Permanently Defueled Technical Specifications.
>
> (a) Deleted.
>
> (b) Deleted.

This license condition is revised to reflect the permanently defueled condition of the facility. After certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). The license condition is revised to remove the reference to reactor operation as well as the reference to the now deleted License Condition 2.C.(1). Therefore, the NRC staff approves revision to the License Condition 2.C.(2).

4.2.18 License Condition 2.C.(4)

Currently, License Condition 2.C.(4) reads:

> (4) The licensee is authorized to operate the Duane Arnold Energy Center following installation of modified safe-ends on the eight primary recirculation system inlet lines which are described in the licensee letter dated July 31, 1978, and supplemented by letter dated December 8, 1978.

The licensee proposes to delete this license condition in its entirety to reflect the permanently defueled condition of the facility. Once certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor pursuant to 10 CFR 50.82(a)(2). Thus, the requirement for installation of equipment to support operation is not necessary. Therefore, the NRC staff approves deletion of the License Condition 2.C.(4).

4.2.19 License Condition 2.C.(11)

Currently, License Condition 2.C.(11) reads:

> (11) The information in the UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be incorporated into the UFSAR no later than the next scheduled update required by 10 CFR 50.71(e) following the issuance of this renewed operating license. Until this update is complete, the licensee may not make changes to the information in the supplement. Following incorporation into the UFSAR, the need for prior Commission approval of any changes will be governed by 10 CFR 50.59.

The licensee proposes to delete this license condition in its entirety to reflect the permanently defueled condition of the facility. Once certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor pursuant to 10 CFR 50.82(a)(2). Thus, the requirement for the addition of UFSAR information to support operation is not necessary. Therefore, the NRC staff approves deletion of the License Condition 2.C.(11).

4.2.20 License Condition 2.C.(12)

Currently, License Condition 2.C.(12) reads:

> (12) The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), and as supplemented by Appendix A of NUREG-1955, “Safety Evaluation Report Related to the License Renewal of Duane Arnold Energy Center,” dated November 2010, as supplemented by letter from the licensee to the NRC dated November 23, 2010, describes

certain programs to be implemented and activities to be completed before the period of extended operation.

a. NextEra Energy Duane Arnold, LLC shall implement those new programs and enhancements to existing programs no later than February 21, 2014.

b. NextEra Energy Duane Arnold, LLC shall complete those activities no later than February 21, 2014.

The licensee shall notify the NRC in writing within 30 days after having accomplished item (a) above and include the status of those activities that have been or remain to be completed in item (b) above.

The licensee proposes to delete this license condition in its entirety to reflect the permanently defueled condition of the facility. This license condition imposed program requirements, necessary for the DAEC to continue operation beyond February 21, 2014. Once certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor pursuant to 10 CFR 50.82(a)(2). Thus, the program requirements to support extended operation are not necessary. Therefore, the NRC staff approves deletion of the License Condition 2.C.(11).

4.2.21 License Condition 2.C.(13)

Currently, License Condition 2.C.(13) reads:

(13) The licensee shall implement the most recent staff-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as the method to demonstrate compliance with the requirements of 10 CFR Part 50, Appendix H. Any changes to the BWRVIP ISP capsule withdrawal schedule must be submitted for staff review and approval. Any changes to the BWRVIP ISP capsule withdrawal schedule which affects the time of withdrawal of any surveillance capsules must be incorporated into the licensing basis. If any surveillance capsules are removed without the intent to test them, these capsules must be stored in a manner which maintains them in a condition which would support re-insertion into the reactor pressure vessel if necessary.

The licensee proposes to delete License Condition 2.C(13). The licensee has already submitted its certification for permanent cessation of operations in accordance with 10 CFR 50.82(a)(1)(i) by letter dated January 18, 2019. This license condition is proposed for deletion in its entirety to reflect the permanently defueled condition of the facility once the certification of permanent removal of fuel from the reactor vessel is submitted.

The NRC staff noted that License Condition 2.C(13) is related to the reactor vessel material surveillance program required by Appendix H to 10 CFR Part 50 and was revised with a license condition as a result of its renewed license to operate an additional 20 years past its original license. The license condition was imposed with the assumption that DAEC would be operating. The NRC staff noted that the requirements in Appendix H to 10 CFR Part 50 are only relevant to nuclear power plants that are authorized to operate in the reactor-critical operating mode because this is the plant operating mode that produces high energy neutrons as a result of the reactor's nuclear fission process, and the requirements are set in place to provide assurance that the reactor pressure vessel (RPV) will maintain adequate levels of fracture toughness throughout the operating life of the reactor. Further, 10 CFR 50.60(a) stipulates that reactor facilities, for which the certifications required under 10 CFR 50.82(a)(1) have been submitted, no longer need to meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in 10 CFR Part 50, Appendices G and H.

The NRC staff finds continued implementation of this license item will no longer be necessary because the licensee has decided to permanently cease power operations at DAEC, and from a fracture toughness perspective, the RPV will cease to be exposed to further irradiation by high energy neutrons or subjected to any high thermal stress environments.

Based on its review of the proposed deletion, the NRC staff concludes that continued implementation of License Condition 2.C(13) will no longer be necessary for DAEC in accordance with 10 CFR 50.60(a) because power operation will no longer be authorized once the 10 CFR 50.82(a)(1) certifications have been docketed. Therefore, the NRC staff finds the deletion of License Condition 2.C(13) acceptable.

4.2.22 License Condition 2.D

Currently, License Condition 2.D reads:

> D. This license is effective as of the date of issuance and shall expire at midnight February 21, 2034.

The licensee proposes License Condition 2.D to read:

> D. This license is effective as of the date of issuance and is effective until the Commission notifies the licensee in writing that the license is terminated.

Once certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor pursuant to 10 CFR 50.82(a)(2). Thus, February 21, 2034, no longer has significance and does not govern termination of the license. Therefore, the NRC staff approves the revision to the License Condition 2.D.

4.3 Changes to Appendix B – Additional Conditions

The licensee has provided the following revision to APPENDIX B, ADDITIONAL CONDITIONS in the submittal dated June 20, 2019.

> OPERATING LICENSE NO. DPR-49 is changed to LICENSE NO. DPR-49.

The following table shows the current and the proposed revision to the additional conditions. Also provided is the basis for each revision.

Appendix B Additional Conditions	
Current Amendment 260 (1)	Proposed Amendment 260 (1)

<table>
<tr><th colspan="2">Appendix B Additional Conditions</th></tr>
<tr><td>At the time of the closing of the transfer of the license from Interstate Power and Light Company (IPL) to FPLE Duane Arnold*, IPL shall transfer to FPLE Duane Arnold* IPL's decommissioning funds accumulated as of such time, with a aggregate minimum value of at least $186 million, and FPLE Duane Arnold* shall deposit such funds in an external decommissioning trust fund established by FPLE Duane Arnold* for DAEC. NextEra Energy Duane Arnold shall take all necessary steps to ensure that this external trust fund is maintained in accordance with the requirements of the order approving the license transfer, NRC regulations, and consistent with the safety evaluation supporting the order. The trust agreement shall be in a form acceptable to the NRC.

* On April 16, 2009, the name "FPL Energy Duane Arnold, LLC was changed to "NextEra Energy Duane Arnold, LLC."</td><td>NextEra Energy Duane Arnold shall take all necessary steps to ensure that the external trust fund established at the time of the closing of the transfer of the license from Interstate Power and Light Company (IPL) to FPLE Duane Arnold is maintained in accordance with the requirements of the December 23, 2005 order approving the license transfer, NRC regulations, and consistent with the safety evaluation supporting the order. The trust agreement shall be in a form acceptable to the NRC.</td></tr>
<tr><th colspan="2">Basis</th></tr>
<tr><td colspan="2">The revisions to the condition pertaining to decommissioning trust funding are to clarify current requirements and remove statements that are no longer relevant.</td></tr>
</table>

Appendix B Additional Conditions	
Current Amendment [No.] 260 (3) NextEra Energy Duane Arnold shall take no action to cause FPL Group Capital, or its successors and assigns, to void, cancel, or modify its $50 million contingency commitment to NextEra Energy Duane Arnold, as represented in the license transfer application, or cause it to fail or perform or impair its performance under the commitment, or remove or interfere with NextEra Energy Duane Arnold's ability to draw upon the commitment, without the prior written consent from the NRC. An executed copy of the Support Agreement shall be submitted to the NRC no later than 30 days after completion of the license transfer. Also, NextEra Energy Duane Arnold shall inform the NRC in writing any time that it draws upon the $50 million commitment.	Proposed DELETED
Basis	
As stated in the NRC's Safety Evaluation for the DAEC license transfer, dated December 23, 2005 (ADAMS Accession No. ML053420246), the purpose of the support agreement is to ensure the licensee has sufficient funds to cover operating costs. Moreover, the Support Agreement, which was provided as Enclosure 9 to the August 1, 2005 License Transfer Application (ADAMS Accession No. ML052160266), states that the agreement terminates when the plant ceases commercial operations. After certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor pursuant to 10 CFR 50.82(a)(2). Thus, DAEC will cease commercial operation and this License Condition will no longer apply.	

The NRC staff finds the above revisions and the basis for the revisions to be acceptable.

4.4 Changes to Appendix A, Technical Specification

4.4.1 Table of Contents

The licensee proposed to revise the Table of Contents to reflect proposed additions, deletions, and changes to the TSs as described in this SE. The changes to the Table of Contents are editorial and do not change any technical content. The NRC staff finds the changes to the Table of Contents acceptable.

4.4.2 Definitions Proposed for Deletion

The licensee proposes deletion of the following definitions since the terms are not used in any PDTS specification and do not apply to a facility in the permanently defueled condition. The licensee also proposed deletion of Table 1.1-1 MODES. Once DAEC dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR Part 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel, pursuant to 10 CFR 50.82(a)(2):

- AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

  The APLHGR shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod for all the rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at the height.

- CHANNEL CALIBRATION

  A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with Resistance Temperature Detector (RTD) or thermocouple sensors may consist of an in place qualitative assessment of sensor behavior and normal calibration of the remining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.

- CHANNEL CHECK

  A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

- CHANNEL FUNCTIONAL TEST

  A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.

- CORE ALTERATION

  CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate power range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

- CORE OPERATING LIMITS REPORT (COLR)

  The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

- DOSE EQUIVALENT I-131

  DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/ml), that alone would produce the same dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The dose conversion factors used for this calculation shall be those listed in Federal Guidance Report (FGR) 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” 1989 and FGR 12, “External Exposure to Radionuclides in Air, Water, and Soil,” 1993.

- DRAIN TIME

  The DRAIN TIME is the time it would take for the water inventory in and above the Reactor Pressure Vessel (RPV) to drain to the Reactor Vessel Water Level Safety Limit of T.S. 2.1.1.3 assuming:

  a) The water inventory above the T.S. 2.1.1.3 Safety Limit is divided by the limiting drain rate;

  b) The limiting drain rate is the larger of the drain rate through a single penetration flow path with the highest flow rate, or the sum of the drain rates through multiple penetration flow paths susceptible to a common mode failure (e.g., seismic event, loss of normal power, single human error), for all penetration flow paths below the T.S. 2.1.1.3 Safety Limit except:

     1. Penetration flow paths connected to an intact closed system, or isolated by manual or automatic valves that are locked, sealed, or otherwise secured in the closed position, blank flanges, or other devices that prevent flow of reactor coolant through the penetration flow paths;
     2. Penetration flow paths capable of being isolated by valves that will close automatically without offsite power prior to the RPV water level being equal to the T.S. 2.1.1.3 Safety Limit when actuated by RPV water level isolation instrumentation; or

3. Penetration flow paths with isolation devices that can be closed prior to the RPV water level being equal to the T.S. 2.1.1.3 Safety Limit by a dedicated operator trained in the task, who is in continuous communication with the control room, is stationed at the controls, and is capable of closing the penetration flow path isolation device without offsite power.

c) The penetration flow paths required to be evaluated per paragraph b) are assumed to open instantaneously and are not subsequently isolated, and no water is assumed to be subsequently added to the RPV water inventory;

d) No additional draining events occur; and

e) Realistic cross sectional areas and drain rates are used.

A bounding DRAIN TIME may be used in lieu of a calculated value.

- END OF CYCLE RECIRCULATION PUMP TRIP (EOC RPT) SYSTEM RESPONSE TIME

  The EOC RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine stop valve limit switch or form when the turbine control valve hydraulic oil control oil pressure drops below the pressure switch setpoint to actuation of the breaker secondary (auxiliary) contact. 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.

- INSERVICE TESTING

  The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f).

- LEAKAGE

  LEAKAGE shall be:

  a. <u>Identified LEAKAGE</u>

     1. LEAKAGE into the drywell, such as that from pump seals or valve packing, that is captured and conducted to a sump or collecting tank; or
     2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known not to interfere with the operation of leakage detection systems;

  b. <u>Unidentified LEAKAGE</u>
     All LEAKAGE into the drywell that is not identified LEAKAGE;

  c. <u>Total LEAKAGE</u>
     Sum of the identified and unidentified LEAKAGE.

- LOGIC SYSTEM FUNCTIONAL TEST

  A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.

- MINIMUM CRITICAL POWER RATIO (MCPR)

  The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience transition boiling, divided by the actual assembly operating power. Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and 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.

The NRC staff noted that this definition is associated with the requirements of 10 CFR Part 50, Appendices G and H. 10 CFR 50.60(a) stipulates that reactor facilities for which the certifications required under 10 CFR 50.82(a)(1) have been submitted, no longer need to meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in 10 CFR Part 50, Appendices G and H. The staff finds that once DAEC is permanently shut down and defueled, this definition and requirements of Appendices G and H to 10 CFR Part 50 are no longer applicable.

- 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.

- SHUTDOWN MARGIN (SDM)

  SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that:

  a. The reactor is xenon free;
  b. The moderator temperature is ≥ 68°F (20°C), corresponding to the most reactive state; and
  c. All rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn with the core in its most reactive state during the operating cycle. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

- THERMAL POWER

  THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

- TURBINE BYPASS SYSTEM RESPONSE TIME

  The TURBINE BYPASS SYSTEM RESPONSE TIME consists of two components:

  a. The time from initial movement of the main turbine stop valve or control valve until 80% of the turbine bypass capacity is established; and

  b. The time from initial movement of the main turbine stop valve or control valve until initial movement of the turbine bypass valve.

  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.

NRC Assessment

The NRC staff examined the TS definitions proposed for deletion and Table 1.1-1 and the licensee’s basis for the deletion and concludes that all the terms and table proposed for deletion are only meaningful to a reactor authorized to operate and need not be retained in the defueled TSs. Therefore, the NRC staff finds the deletion of the definition in TS Section 1.1 acceptable.

4.4.3 TS Section 1.2 – “Logical Connectors”

Logical connectors are used in TSs to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these constitutes logical conventions with specific meanings.

Because the defueled TS do not have any conditions to include any logical connects, the NRC staff finds deletion of this section to be acceptable.

4.4.4 TS Section 1.3 – “Completion Times”

This section establishes the Completion Time convention and provides guidance for its use, including several illustrative examples.

The current background for TS Section 1.3 states the following:

> Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Times(s).

The licensee proposed to revise the background to state the following:

> Limiting Conditions for Operation (LCOs) specify minimum requirements for safely maintaining the facility. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Times(s).

The current description for TS Section 1.3 states the following in the first paragraph:

> The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

The licensee proposed to revise the description to delete the paragraphs following the first paragraph and the examples so that it states the following:

> The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS

> Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability.

NRC Assessment

The NRC staff examined the licensee's proposal to modify the background and description sections of TS Section 1.3. Once DAEC permanently shuts down and defuels, DAEC will be authorized only to possess special nuclear material and following a 19-day decay period, the only required LCO will be 3.7.8, Spent Fuel Pool Level Indication. Because this LCO has a required action with a completion time of immediately, the need for the explanatory test and examples contained in TS Section 1.3 are no longer needed. The NRC staff reviewed the licensee's provided basis for these revisions and concludes that the revisions are acceptable.

4.4.5 TS Section 1.4 – "Frequency"

This section defines the proper use and application of Frequency requirements, including several illustrative examples.

The current description of TS Section 1.4 states the following:

> Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.
>
> The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.
>
> Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both. Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.
>
> The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria. Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restriction of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed

prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discussed these special situations.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

<u>EXAMPLE 1.4-1</u>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Examples 1.4-3 and 1.4-4), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable. The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the MODE or other specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

The current description also includes additional examples, which the licensee proposed for deletion. Because they are proposed for deletion, the NRC staff is not providing the detail from the following examples:

EXAMPLE 1.4-2
EXAMPLE 1.4-3
EXAMPLE 1.4-4
EXAMPLE 1.4-5
EXAMPLE 1.4-6

The licensee proposed to revise the description to state the following:

> Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.
>
> The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.
>
> The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.
>
> EXAMPLE
>
> The following example illustrates the manner in which Frequencies are specified.
>
> <u>EXAMPLE 1.4-1</u>
>
> SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK	12 hours

> Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the facility is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified, then SR 3.0.3 becomes applicable.

NRC Assessment

The NRC staff examined the licensee's proposal to modify the description of TS Section 1.4. Once DAEC permanently shuts down and defuels, DAEC will be authorized only to possess special nuclear material and following a 19-day decay period, the only required LCO will be 3.7.8, Spent Fuel Pool Level Indication. Because this LCO has a straightforward SR to check SFP level on a Frequency in accordance with the Surveillance Frequency Control Program, much of the explanatory text and examples in Section 1.4 are no longer needed. The NRC staff reviewed the licensee's provided basis for these revisions and concludes that the revisions are acceptable.

4.4.6 TS Section 2.0 – "Safety Limits"

The licensee proposed deletion of this section in its entirety. It includes the following TSs:

TS 2.1 – Safety Limits
TS 2.2 – Safety Limit Violation

The fuel cladding, RPV and primary system piping are principal barriers to prevent the release of radioactive materials to the environs during operations. TS 2.1 establishes Safety Limits to protect the integrity of these barriers during normal plant operations and anticipated transients. TS 2.2 defines the actions to take if there is a non-compliance with a safety limit.

These TSs are being proposed for deletion in their entirety. Since fuel is no longer allowed in the reactor vessel for a permanently defueled plant, the need for safety limits no longer exists. As such, TSs 2.1 and 2.2 do not apply to a reactor that is in a permanently defueled condition. Once the licensee dockets the certifications required by 10 CFR 50.82(a)(1), the 10 CFR 50 license will no longer authorize operation of the reactor, or emplacement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Further, these specifications apply to the reactor vessel and therefore, do not apply to the storage and handling of spent fuel in the spent fuel pool storage.

These safety limits apply to an operating reactor. They do not have a function in the permanently defueled condition. Therefore, based on its review, the NRC staff finds the deletion of Section 2, TS 2.1, "Safety Limits," and TS 2.2, "Safety Limits Violation," in their entirety, acceptable.

4.4.7 TS Section 3.0 – "LCO Applicability"

The LCOs and SRs provide for appropriate control of process variables, design features, or operating restrictions needed for appropriate functional capability of RCS equipment required for safe operation of the facility. The licensee proposed deletions in this TS section for the following LCOs and SRs:

This section contains LCOs, which specify the lowest functional capability or performance levels of equipment required for safe operation of the facility and contains the general requirements applicable to all LCOs and applies at all times unless otherwise stated in TS.

The licensee proposed revisions to LCOs 3.0.1 and 3.0.2 and deletion of LCOs 3.0.3, 3.0.4, 3.0.5, 3.0.6, 3.0.7, 3.0.8, and 3.0.9.

The current LCO 3.0.1 states the following:

> LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, LCO 3.0.8, and LCO 3.0.9.

The licensee proposed to revise LCO 3.0.1 to state the following:

> LCOs shall be met during the specified conditions in the Applicability, except as provided in LCO 3.0.2.

The current LCO 3.0.2 states the following:

> Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.
>
> If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

The licensee proposed to revise LCO 3.0.2 to state the following:

> Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met.
>
> If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.

NRC Assessment

The NRC staff reviewed the proposed changes to LCO 3.01 and concludes that the changes are consistent with the transition to a permanently shutdown and defueled facility. Since 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant or placing fuel in the reactor vessel, the references to Modes are no longer applicable. In addition, the references to LCOs 3.0.7, 3.0.8, and 3.0.9 pertain to special tests and operations required for an operating reactor and actions required for equipment, respectively, that will no longer be operating once DAEC permanently shuts down and defuels. Therefore, the NRC staff finds the changes acceptable.

The NRC staff reviewed the proposed change to LCO 3.0.2 and concludes that the change is consistent with the transition to a permanently shutdown and defueled facility. The reference to LCO 3.0.5 pertains to allowing flexibility in declaring equipment operable to allow testing to demonstrate operability, on equipment that will no longer be operating once DAEC permanently shuts down and defuels. The reference to LCO 3.0.6 pertains to an exception to LCO 3.0.2 for support systems that have an LCO specified in the TS and will no longer be needed once DAEC permanently shuts down and defuels. Therefore, the NRC staff finds the change acceptable.

The NRC staff reviewed the proposed deletions of LCOs 3.0.3, 3.0.4, 3.0.5, 3.0.6, 3.0.7, 3.0.8, and 3.0.9. These LCOs pertain to an operating reactor, operability testing, special tests and operations, support systems, snubbers, and barriers. Once DAEC permanently shuts down and defuels, these LCOs will no longer be applicable. Therefore, the NRC staff finds the deletions acceptable.

4.4.8 TS Section 3.0, "Surveillance Requirement Applicability"

This section contains the general requirements applicable to all SRs and applies at all times unless otherwise stated in a TS. The licensee proposed revisions to SRs 3.0.1, 3.0.2, and 3.0.4.

The current SR 3.0.1 states the following:

> SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

The licensee proposed to revise SR 3.0.1 to state the following:

> SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

The current SR 3.0.2 states the following:

> The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of Frequency is met.
>
> For Frequencies specified as "once," the above interval extension does not apply.
>
> If a Completion Time requires periodic performance on a "once per…" basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.

The licensee proposed to revise SR 3.0.2 to state the following:

> The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the

previous performance or as measured from the time a specified condition of Frequency is met.

For Frequencies specified as “once,” the above interval extension does not apply.

The current SR 3.0.4 states the following:

Entry into a MODE or other specified condition in the Applicability of an LCO shall only be made when the LCO’s Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with LCO 3.0.4. This provision shall not prevent entry into MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The licensee proposed to revise SR 3.0.4 to state the following:

Entry into a specified condition in the Applicability of an LCO shall only be made when the LCO’s Surveillances have been met within their specified Frequency, except as provided by SR 3.0.3.

This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with ACTIONS.

NRC Assessment

The NRC staff reviewed the proposed changes to SRs 3.01, 3.0.2, and 3.0.4 and concludes that the changes are consistent with the transition to a permanently shutdown and defueled facility. Since 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant or placing fuel in the reactor vessel, the references to Modes, the discussion of specific Completion Times performed on a “once per…” basis and exceptions to this Specification, and the discussion about shutting down the unit, are no longer applicable. Therefore, the proposed changes to delete these references would reflect the plant status and are appropriate and acceptable.

4.4.9 TS Section 3.1.1 - “Shutdown Margin (SDM)”

TS 3.1.1 establishes that the shutdown margin is specified to ensure:

a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Accidents;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

TS 3.1.1 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS Section 3.1.2 – “Reactivity Anomalies”

TS 3.1.2 establishes reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and abnormal operational transients. TS 3.1.2 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS Section 3.1.3 – “Control Rod Operability”

TS 3.1.3 establishes that the performance of the control rods in the event of a Design Basis Accident or transient meets the assumptions used in the safety analyses. TS 3.1.3 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS Section 3.1.4 – “Control Rod Scram Times”

TS 3.1.4 establishes the control rod drive (CRD) system controls reactivity changes during abnormal operational transients and DBAs to ensure that specified acceptable fuel design limits are not exceeded. TS 3.1.4 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS Section 3.1.5 – “Control Rod Scram Accumulators”

TS 3.1.5 establishes that control rod scram accumulators are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulates store sufficient energy to fully insert a control rod at any reactor vessel pressure. TS 3.1.5 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS Section 3.1.6 – “Rod Pattern Control”

TS 3.1.6 establishes that the control rod patterns are consistent with the assumptions of the control rod drop accident analyses in the thermal power ≤ 10 percent reactor thermal power. TS 3.1.6 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS Section 3.1.7 – “Standby Liquid Control [SLC]) System”

TS 3.1.7 establishes the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon free state without taking credit for control rod movement. TS 3.1.7 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS Section 3.1.8 – “Scram Discharge Volume [SDV] Vent and Drain Valves”

TS 3.1.8, Scram Discharge Volume Vent and Drain Valves, function to: (1) limit the amount of reactor coolant discharged during a reactor scram; and (2) ensure there is sufficient volume to accept the reactor coolant discharged during a scram. TS 3.1.8 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

4.4.10 Proposed Deletion of TS 3.2 Power Distribution Limits

TS Section 3.2.1 – “Average Planar Linear Heat Generation Rate” [APLHGR]

TS 3.2.1, APLHGR places limits to establish that the fuel design limits are not exceeded during abnormal operational transients; and, also to assure that the peak cladding temperature during the postulated design basis loss of coolant accident. TS 3.2.1 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS Section 3.2.2 – “Minimum Critical Power Ratio” [MCPR]

TS 3.2.2 places limits on the MCPR to establish that no fuel damage results during abnormal operational transients. TS 3.2.2 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

4.4.11 Proposed Deletion of TS Section 3.3 “Instrumentation”

Section 3.3 of TSs, “Instrumentation,” contains the LCOs, actions, and SRs that provide for appropriate functional capability of the instrumentation required for safe operation of the facility. TS Section 3.3 contains the following:

- TS 3.3.1.1, “Reactor Protection System [RPS] Instrumentation”
  - Applicability – MODES 1, 2 and 5
- TS 3.3.1.2, “Source Range Monitor [SRM] Instrumentation”
  - Applicability – MODES 2, 3, 4 and 5
- TS 3.3.2.1, “Control Rod Block Instrumentation”
  - Applicability – MODES 1, 2, 3, 4 and 5
- TS 3.3.3.1, “Post Accident Monitoring [PAM] Instrumentation”
  - Applicability – MODES 1 and 2
- TS 3.3.3.2, “Remote Shutdown System”
  - Applicability – MODES 1 and 2
- TS 3.3.4.1, “End of Cycle Recirculation Pump Trip [EOC-RPT] Instrumentation”
  - Applicability – Reactor Power ≥ 26%
- TS 3.3.4.2, “Anticipated Transient Without Scram Recirculation Pump Trip [ATWS-RPT] Instrumentation”
  - Applicability – MODE 1
- TS 3.3.5.1, “Emergency Core Cooling System [ECCS] Instrumentation”
  - Applicability – MODES 1, 2 and 3
- TS 3.3.5.2, “Reactor Pressure Vessel [RPV] Water Inventory Control Instrumentation”
  - Applicability – MODES 4 and 5
- TS 3.3.5.3, “Reactor Core Isolation Cooling [RCIC] Instrumentation”
  - Applicability – MODES 1, 2 and 3
- TS 3.3.6.1, “Primary Containment Isolation Instrumentation”
  - Applicability – MODES 1, 2 and 3
- TS 3.3.6.2, “Secondary Containment Isolation Instrumentation”
  - Applicability – MODES 1, 2 and 3
- TS 3.3.6.3, “Low-Low Set [LL] Instrumentation”
  - Applicability – MODES 1, 2 and 3
- TS 3.3.7.1, “Standby Filter Unit [SFU] Instrumentation”
  - Applicability – MODES 1, 2 and 3, during movement of irradiated fuel assemblies in secondary containment, and during CORE ALTERATIONS

- TS 3.3.8.1, “Loss of Power [LOP] Instrumentation
    - Applicability – MODES 1, 2 and 3 and when the associated Diesel Generator is required to be Operable by LCO 3.8.2.
- TS 3.3.8.2, “Reactor Protection System [RPS] Electrical Power Monitors
    - Applicability – MODES 1, 2, 3, 4 and 5

(Note – some Applicability conditions, described above, have additional conditions and are not shown.)

The licensee proposed to delete all of the above TSs, since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled conditions of DAEC.

TS Section 3.3.1

The NRC staff reviewed the proposed changes to TS Section 3.3.1. The staff concludes that these above TSs provide the LCOs, actions, and SRs necessary to maintain the instrumentation associated with the RPS and SRM.

The LAR describes that TS 3.3.1.1 “ensures safe operation of the reactor by specifying limited safety system settings (LSSS) in terms of parameters directly monitored by the RPS.” The LAR describes that TS 3.3.1.2 “is used by the operator to monitor the approach to criticality and determine when criticality is achieved.”

TS Section 3.3.1 is required for safe operation of the facility only when the reactor is in MODES 1 through 5. Once DAEC is placed in a defueled condition in accordance with 10 CFR 50.82(a)(2), the 10 CFR Part 50 license will no longer authorize operation, placement or retention of fuel in the reactor vessel and entry into any of the listed applicable MODES will no longer be possible.

Therefore, the staff finds that the instrumentation addressed by the TSs in Section 3.3.1 are no longer applicable to a reactor that is permanently shutdown and can therefore be deleted.

TS Section 3.3.2

The NRC staff reviewed the proposed change to TS Section 3.3.2. The staff concludes that these TSs provide the LCOs, actions, and SRs necessary to maintain the instrumentation associated with the Control Rod Block.

The LAR describes that TS 3.3.2.1 “ensures that specified fuel limits are not exceeded for postulated transients and accidents.” TS Section 3.3.2 is required for safe operation of the facility only when the reactor is in Modes 1 through 5. Once DAEC is placed in a defueled condition in accordance with 10 CFR 50.82(a)(2), the 10 CFR Part 50 license will no longer authorize operation, placement or retention of fuel in the reactor vessel and entry into any of the listed applicable modes will no longer be possible.

Therefore, the NRC staff finds that the instrumentation addressed by the TS in Section 3.3.2 is no longer applicable to a reactor that is permanently shutdown and this TS section can, therefore, be deleted.

TS Section 3.3.3

The NRC staff reviewed the proposed changes to TS Section 3.3.3. The staff concludes that these TSs provide the LCOs, actions, and SRs necessary to maintain the instrumentation associated with PAM and remote shutdown.

The LAR describes that TS 3.3.3.1 “ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident.”

The LAR describes that TS 3.3.3.2 “provides the control room operator with sufficient instrumentation and controls to place and maintain the plant in a safe shutdown condition from a location other than the control room…to protect against…the control room becoming inaccessible.”

TS Section 3.3.3 is required for safe operation of the facility only when the reactor is in MODES 1 and 2. Once DAEC is placed in a defueled condition in accordance with 10 CFR 50.82(a)(2), the 10 CFR Part 50 license will no longer authorize operation, placement or retention of fuel in the reactor vessel and entry into any of the listed applicable modes will no longer be possible.

Therefore, the NRC staff finds that the instrumentation addressed by the TSs in Section 3.3.3 are no longer applicable to a reactor that is permanently shutdown and this TS section can therefore be deleted.

TS Section 3.3.4

The NRC staff reviewed the proposed changes to TS Section 3.3.4. and concludes that the above TSs provide the LCOs, action and SRs necessary to maintain the instrumentation associated with EOC-RPT and ATWS-RPT.

The LAR describes that TS 3.3.4.1 “initiates a recirculation pump trip [RPT] to reduce the peak reactor pressure and power resulting from turbine trip or generator load rejection transients to provide additional margin to core thermal minimum critical power ratio safety limits.”

The LAR describes that TS 3.3.4.2 “ensures initiation of an RPT,” to add negative reactivity to lessen the effects of an anticipated transient without scram (ATWS) event. TS Section 3.3.4 is required for safe operation of the facility only when the reactor thermal power is ≥ 26 percent (TS 3.3.4.1) or in Mode 1 (TS 3.3.4.2). Once DAEC is placed in a defueled condition in accordance with 10 CFR 50.82(a)(2), the 10 CFR Part 50 license will no longer authorize operation, placement, or retention, of fuel in the reactor vessel and entry into any of the listed applicable MODES will no longer be possible.

Therefore, the NRC staff finds that none of the instrumentation addressed by the TSs in Section 3.3.4 are applicable to a reactor that is permanently shutdown and this TS section can therefore be deleted.

TS Section 3.3.5

The NRC staff reviewed the proposed change to TS Section 3.3.5. The staff concludes that these above TSs provide the LCOs, actions, and SRs necessary to maintain the instrumentation associated with ECCS, RPV Water Inventory Control, and RCIC.

The LAR describes that TS 3.3.5.1 "ensures initiation of appropriate responses from emergency systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient."

The LAR describes that TS 3.3.5.2 "supports the requirements of LCO 3.5.2…and the definition of DRAIN TIME," and supports operation of core spray (CS) and low-pressure coolant injection (LPCI).

The LAR describes that TS 3.3.5.3 "ensures initiation of actions to ensure adequate core cooling when the reactor vessel is isolated from its primary heat sink…and normal coolant makeup flow from the reactor feedwater system is unavailable."

TS Section 3.3.5 is required for safe operation of the facility only when the reactor is in Modes 1 through 5. Once DAEC is placed in a defueled condition in accordance with 10 CFR 50.82(a)(2), the 10 CFR Part 50 license will no longer authorize operation, placement, or retention, of fuel in the reactor vessel and entry into any of the listed applicable modes will no longer be possible.

Therefore, the NRC staff finds that none of the instrumentation addressed by the TSs in Section 3.3.5 are applicable to a reactor that is permanently shutdown and this TS section can therefore be deleted.

<u>TS Section 3.3.6</u>

The NRC staff reviewed the proposed change to TS Section 3.3.6. The staff concludes that the above TSs provide the LCOs, actions, and SRs necessary to maintain the instrumentation associated with primary and secondary containment isolation and low-low set.

The LAR describes that TS 3.3.6.1, "Primary Containment Isolation Instrumentation," initiates automatic closure of appropriate Primary Containment Isolation Valves [PCIVs] to limit fission product release during and following postulated design basis accidents."

The LAR describes that TS 3.3.6.2, "Secondary Containment Isolation Instrumentation," initiates automatic closure of appropriate secondary containment isolation valves and/or dampers (SCIV/Ds) and starts the SBGT system to limit fission product release during and following postulated DBAs.

TS 3.3.6.3, "Low-Low Set [LLS] Instrumentation," ensures the safety relief valve (SRV) discharge lines are not adversely impacted by thrust loads caused by valve actuation.

The licensee states that these TS are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in Modes 1 through 3. After the certifications required by 10 CFR 50.82(a)(1) are docketed, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel.

Therefore, the licensee states that the proposed deletion of all TS in Section 3.3.6, including associated SRs, is acceptable and with the TS section deleted in its entirety, the corresponding TS Bases will also be deleted.

Therefore, the NRC staff finds that none of the instrumentation addressed by the TSs in Section 3.3.6 are applicable to a reactor that is permanently shutdown and this TS section can therefore be deleted.

The NRC staff finds the deletion of TSs 3.3.6.1, 3.3.6.2, and 3.3.6.3, acceptable.

TS Section 3.3.7

The NRC staff reviewed the proposed change to TS Section 3.3.7. The staff concludes that the above TSs provide the LCOs, actions, and SRs, necessary to maintain the instrumentation associated with the SFU.

TS 3.3.7.1 - Standby Filter Unit (SFU) Instrumentation

The LAR describes that TS 3.3.7.1 ensures automatic action to pressurize the control building envelope to minimize the consequences of radioactive material in the control building envelope, by using the SFU, to ensure habitability of the CR during all plant conditions. The SFU is support equipment for the CBEV and the specific applicability condition, "during movement of irradiated fuel in secondary containment," is no longer required.

TS Section 3.3.7 is required for safe operation of the facility only when the reactor is in Modes 1 through Modes 3, during movement of irradiated fuel in secondary containment, and during CORE ALTERATIONS. Once DAEC is placed in a defueled condition in accordance with 10 CFR 50.82(a)(2), the 10 CFR Part 50 license will no longer authorize operation, placement or retention, of fuel in the reactor vessel and entry into any of the listed applicable modes and CORE ALTERATIONS will no longer be possible.

After the reactor has been completely defueled following permanent shut down, the licensee states that an FHA over the reactor core is no longer a credible accident and that the FHA in the SFP is the bounding accident. The licensee performed a revised analysis for an FHA in the SFP which determined that, following a 19-day decay period, CBEV is not required to maintain dose consequences for CR occupants within the criteria of 10 CFR 50.67(b)(2)(iii). Therefore, TS LCOs and SRs associated with CBEV and support equipment, specifically, SFU instrumentation, are proposed for deletion. The licensee also states that with this TS deleted, the corresponding TS Bases will also be deleted.

Therefore, the NRC staff finds that none of the instrumentation addressed by the TS in Section 3.3.7 are applicable to a reactor that is permanently shutdown and this TS section can therefore be deleted.

The NRC staff finds the deletion of TS 3.3.7.1 acceptable.

TS Section 3.3.8

The NRC staff reviewed the proposed change to TS Section 3.3.8. The staff concludes that these above TSs provide the LCOs, actions, and SRs necessary to maintain the instrumentation associated with LOP and RPS electric power monitors.

The LAR describes that TS 3.3.8.1 "monitors the 4.16kV emergency bus voltages and the startup and standby transformer secondary winding voltages." The LAR describes that TS 3.3.8.2 "provides protection of electrical loads supplied from the RPS bus against unacceptable voltage and frequency conditions."

TS Section 3.3.8 is required for safe operation of the facility only when the reactor is in Modes 1 through 3 and when the associated diesel generator is required to be operable by LCO 3.8.2 (TS 3.3.8.1) and Modes 1 through 5 (TS 3.3.8.2).

The LOP is support equipment for the CBEV and the specific applicability, “when the associated diesel generator is required to be Operable by LCO 3.8.2,” is no longer required. Therefore, the NRC staff finds that none of the instrumentation addressed by the TSs in Section 3.3.8 are applicable to a reactor that is permanently shutdown and this TS section can therefore be deleted.

The NRC staff has reviewed the instrumentation TSs proposed for deletion to ensure that these LCOs and SRs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs. The staff notes that these TSs indicate modes and conditions for which the TSs are applicable. The reference to modes for a permanently shutdown and defueled reactor, such as DAEC, is no longer applicable and no longer required.

Once DAEC is placed in a defueled condition in accordance with 10 CFR 50.82(a)(2), the 10 CFR Part 50 license will no longer authorize operation, placement or retention of fuel in the reactor vessel and DAEC is no longer in a configuration under which the TS modes apply.

Therefore, the NRC staff finds the proposed deletion of TS Section 3.3 to be acceptable as they are no longer required to be included in TSs for a reactor that is permanently shutdown.

4.4.12 Proposed Change to TS 3.4 – “Reactor Coolant System” (RCS)

3.4.1 - Recirculation Loops Operating
3.4.2 - Jet Pumps
3.4.3 - Safety Relief Valves and Safety Valves
3.4.4 - RCS Operational Leakage
3.4.5 - RCS Leakage Detection Instrumentation
3.4.6 - RCS Specific Activity
3.4.7 - Residual Heat Removal Shutdown Cooling System – Hot Shutdown
3.4.8 - Residual Heat Removal Shutdown Cooling System – Cold Shutdown
3.4.9 - RCS Pressure and Temperature Limits
3.4.10 - Reactor Steam Dome Pressure

The TS 3.4 RCS contains LCOs that provide for appropriate control of process variables, design features, or operating restrictions needed for appropriate functional capability of RCS equipment required for safe operation of the facility. The licensee proposes to delete the following section:

TS 3.4.1, “Recirculation Loops Operating,” is designed to provide a forced coolant flow through the core to remove heat from the fuel. The recirculation system also controls reactivity over a wide span of reactor power by varying the recirculation flow rate to control the void content of the reactor coolant. TS 3.4.1 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.4.2, Jet Pumps, are part of the reactor coolant recirculation system and provides forced circulation through the core to remove heat from the fuel. TS 3.4.2 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.4.3 - SRVs and safety valves, ensure the reactor vessel is protected from overpressure during upset conditions by self-actuated safety valves. TS 3.4.3 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.4.4 - RCS operational leakage, limits on RCS operational leakage to ensure appropriate action is taken before the integrity of the RCPB is impaired. TS 3.4.4 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.4.5 - "RCS Leakage Detection Instrumentation," which is applicable in MODES 1, 2, and 3, provides a means for detecting and, to the extent practical, identifying the location of the source of RCS leakage.

The licensee states that with the reactor in a permanently shut down and defueled condition, reactor pressure and temperature limits do not apply and that after the certifications required by 10 CFR 50.82(a)(1) are docketed, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel. The licensee also states that the corresponding TS Bases will also be deleted.

Systems for separating the leakage of an identified source from an unidentified source are necessary during operation to provide prompt and quantitative information to the operators to permit them to take immediate corrective action. TS 3.4.5 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

The NRC staff finds the deletion of TS 3.4.5 acceptable.

TS 3.4.6 - RCS specific activity, provides limits on the maximum allowable level of radioactivity in the reactor coolant to ensure that in the event of a release of any radioactive material to the environment during a design basis accident, radiation doses are maintained within the limits of 10 CFR 50.67. TS 3.4.6 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.4.7 – "Residual Heat Removal Shutdown Cooling System - Hot Shutdown," ensures decay heat, produced by irradiated fuel in the shutdown reactor core, is removed to reduce the temperature of the reactor coolant to ≤ 212°F. TS 3.4.7 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.4.8 - "Residual Heat Removal Shutdown Cooling System - Cold Shutdown," also ensures decay heat, produced by irradiated fuel in the shutdown reactor core, is removed to reduce the temperature of the reactor coolant to ≤ 212 °F. TS 3.4.8 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.4.9 - RCS pressure and temperature limits establishes the pressure and temperature changes during RCS heatup and cooldown remain within the design assumptions and the stress limits for cyclic operation.

The NRC staff noted that this section is specifically related to the requirements for the RCPB set forth in Appendices G and H to 10 CFR Part 50. Section 10 CFR 50.60(a) stipulates that reactor facilities which have submitted the certifications required under 10 CFR 50.82(a)(1) no longer need to meet the fracture toughness and material surveillance program requirements for the RCPB set forth in Appendices G and H to 10 CFR Part 50. Based on its review of the proposed deletion, the NRC staff concludes that continued implementation of TS 3.4.9 will no longer be necessary for DAEC in accordance with 10 CFR 50.60(a) because these LCOs will not apply in a permanently shutdown and defueled condition, and power operation will no longer be authorized once the 10 CFR 50.82(a)(1) certifications have been docketed.

Therefore, the NRC staff finds the deletion of TS Section 3.4.9 acceptable.

TS 3.4.10 – “Reactor Steam Dome Pressure”, establishes compliance with an assumed initial condition of design basis accidents and transients and is conservative to the value used in the determination of compliance with reactor pressure vessel overpressure protection criteria. TS 3.4.10 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

4.4.13 Proposed Deletion of TS 3.5 ECCS, RPV Water Inventory Control, and RCIC System

3.5.1 - Emergency Core Cooling Systems - Operating
3.5.2 - Reactor Pressure Vessel Water Inventory Control
3.5.3 - Reactor Core Isolation Cooling System

TS 3.5.1. – “Emergency Core Cooling Systems - Operating,” limits the release of radioactive materials to the environment following a loss of coolant accident. The ECCS network consists of the HPCI system, the CS system, the LPCI mode of the RHR, and the automatic depressurization system. TS 3.5.1 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.5.2 – “Reactor Pressure Vessel Water Inventory Control,” ensures the RPV water level remains above the top of the active irradiated fuel at all times to prevent elevated fuel cladding temperatures when the reactor is in cold shutdown or refueling. TS 3.5.2 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.5.3, - “Reactor Core Isolation Cooling System,” is designed to operate either automatically or manually following RPV isolation in accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of the RPV water level. TS 3.5.3 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

4.4.14 Proposed Deletion of TS 3.6 Containment Systems

Deletion of TS Section 3.6.1

TS 3.6.1.1 Primary Containment

TS 3.6.1.2 Primary Containment Air Lock

TS 3.6.1.3 Primary Containment Isolation Valves (PCIVs)

TS 3.6.1.4, “Drywell Air Temperature,” limits the drywell average air temperature to a value below that used in the UFSAR Chapter 15 safety analyses.

TS 3.6.1.5, “Low-Low Set (LLS) Valves,” is to prevent excessive short duration SRV cycles that would occur with valve actuation at the relief setpoint.

TS 3.6.1.6, “Reactor Building-to-Suppression Chamber Vacuum Breakers,” ensures vacuum is relieved when primary containment depressurizes below reactor building pressure.

TS 3.6.1.7, “Suppression Chamber-to-Drywell Vacuum Breakers,” relieve vacuum in the drywell. The licensee states that these TS are related to assuring the appropriate functional capability of plant, equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in MODES 1 through 3. After the certifications required by 10 CFR 50.82(a)(1) are docketed, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel. Therefore, the licensee states that these TS are no longer applicable and the proposed deletion of all TS in Section 3.6.1, including associated SRs, is acceptable. The licensee also states that with the TS section deleted in its entirety, the corresponding TS Bases will also be deleted.

The NRC staff finds the deletion of all TS in Section 3.6.1 acceptable.

Deletion of TS Section 3.6.2

3.6.2.1 - Suppression Pool Average Temperature
3.6.2.2 - Suppression Pool Water Level
3.6.2.3 - Residual Heat Removal Suppression Pool Cooling
3.6.2.4 - Residual Heat Removal Suppression Pool Spray

TS 3.6.2.1 - “Suppression Pool Average Temperature,” ensures limitations on the suppression pool average temperature are enforced to provide assurance that the containment conditions assumed for the safety analyses are met. TS 3.6.2.1 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.6.2.2 - “Suppression Pool Water Level,” ensures limitations on the suppression pool water level are enforced to provide assurance that the primary containment conditions assumed for the safety analyses are met. TS 3.6.2.2 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.6.2.3 - “Residual Heat Removal Suppression Pool Cooling,” is a system that can be used to remove heat from the suppression pool following a DBA. TS 3.6.2.3 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.6.2.4 - “Residual Heat Removal Suppression Pool Spray,” reduces pressure in the suppression chamber. Although not required by the accident analysis to ensure that the suppression chamber remains within the analyzed design pressure and temperature limits, condensing the steam in the suppression chamber airspace reduces the long-term pressure response in the primary containment. TS 3.6.2.4 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

The licensee states that these TS are related to assuring the appropriate functional capability of plant, equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in Modes 1 through 3. After the certifications required by 10 CFR 50.82(a)(1) are docketed, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel. Therefore, the licensee states that these TS are no longer applicable and the proposed deletion of all TS in Section 3.6.2, including associated SRs, is acceptable. The licensee also states that with the TS section deleted in its entirety, the corresponding TS Bases will also be deleted.

The NRC staff finds the deletion of all TS in Section 3.6.2 acceptable.

<u>Deletion of TS Section 3.6.3</u>

TS 3.6.3.1 Containment Atmosphere Dilution (CAD) System
TS 3.6.3.2 Primary Containment Oxygen Concentration
Deletion of TS 3.6.3.1 “Containment Atmosphere Dilution (CAD) System” and TS 3.6.3.2, “Primary Containment Oxygen Concentration.”

TS 3.6.3.1 was previously deleted by Amendment 265.

TS 3.6.3.2 - “Primary Containment Oxygen Concentration,” maintains the containment atmosphere with a low concentration of oxygen, rendering it inert to combustion. TS 3.6.3.2 is applicable in Mode 1 from 24 hours after reactor thermal power is > 15 percent following startup and to 24 hours prior to reducing reactor thermal power to < 15 percent prior to reactor shutdown.

The licensee states that TS 3.6.3.2 is related to assuring the appropriate functional capability of plant, equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in Mode 1. After the certifications required by 10 CFR 50.82(a)(1) are docketed, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel.

Therefore, the licensee states that these TS are no longer applicable and the proposed deletion of all TS in Section 3.6.3, including associated SRs, is acceptable. The licensee also states that with the TS section deleted in its entirety, the corresponding TS Bases will also be deleted.

The NRC staff finds the deletion of all TSs in Section 3.6.3 acceptable.

<u>Deletion of TS Section 3.6.4</u>

Deletion of TS 3.6.4.1, “Secondary Containment,” TS 3.6.4.2, “Secondary Containment Isolation Valves/Dampers (SCIV/Ds),” and TS 3.6.4.3, “Standby Gas Treatment System.”

TS 3.6.4.1 - “Secondary Containment,” states the function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a DBA.

TS 3.6.4.2 - “Secondary Containment Isolation Valves/Dampers (SCIV/Ds),” limits fission product release during and following postulated DBAs.

TS 3.6.4.3 - “Standby Gas Treatment System,” ensures that radioactive materials that leak from the primary containment into the secondary containment following a DBA are filtered and adsorbed prior to exhausting to the environment.

The licensee states that these TS are related to assuring the appropriate functional capability of plant, equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in Modes 1 through 3. After the certifications required by 10 CFR 50.82(a)(1) are docketed, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel.

Therefore, the licensee states that these TS are no longer applicable and the proposed deletion of all TS in Section 3.6.4, including associated SRs, is acceptable. The licensee also states that with the TS section deleted in its entirety, the corresponding TS Bases will also be deleted.

The NRC staff finds the deletion of all TS in Section 3.6.4 acceptable.

4.4.15 Proposed Deletion of TS 3.7 Plant Systems

Deletion of TS 3.7.1, “Residual Heat Removal Service Water (RHRSW) System,” TS 3.7.2, “River Water Supply (RWS) System and Ultimate Heat Sink (UHS),” TS 3.7.3, “Emergency Service Water System (ESW),” TS 3.7.4, “Standby Filter Unit (SFU) System,” TS 3.7.5, “Control Building Chiller (CBC) System,” TS 3.7.6, “Main Condenser Offgas,” TS 3.7.7, “Main Turbine Bypass System,” and TS 3.7.9, “Control Building/Standby Gas Treatment Instrument Air System”

TS 3.7.1 - “Residual Heat Removal Service Water System,” states the system is designed to provide cooling water for the RHR system heat exchangers, required for a safe reactor shutdown following a DBA or transient.

TS 3.7.2 - “River Water Supply System and Ultimate Heat Sink,” states the RWS system provides cooling water required support for various systems required for a safe reactor shutdown following a DBA or transient and the UHS ensures that sufficient water inventory is available for the RWS System.

TS 3.7.3 - “Emergency Service Water System,” states the system provides cooling water for the removal of heat from equipment and various minor heat loads required for a safe reactor shutdown following a DBA or transient and provides cooling to unit components, as desired, during normal operation.

TS 3.7.4 - “Standby Filter Unit System,” states the system provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals or smoke. Specifically, the SFU System provides emergency treatment of outside supply air and a CBE boundary that limits the in-leakage of unfiltered air.

TS 3.7.5 - “Control Building Chiller System,” states the system provides temperature control for the control building heating ventilation air conditioning (HVAC) system under both normal and accident conditions.

TS 3.7.6 - “Main Condenser Off gas,” reduces the gaseous radwaste emissions during plant operation.

TS 3.7.7 - “Main Turbine Bypass System,” states the system is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown.

Modification of TS 3.7.8 - “Spent Fuel Pool Water Level”

TS 3.7.8 - “Spent Fuel Pool Water Level,” is applicable during movement of irradiated fuel assemblies in the SFP. The minimum water level in the spent fuel storage pool ensures the FHA analysis assumptions are met.

The licensee states that TS 3.7.8 is proposed to be maintained with a revision to remove reference to LCO 3.0.3 since LCO 3.0.3 has been deleted.

The NRC staff finds the modification of TS 3.7.8 acceptable.

TS 3.7.9 - "Control Building/Standby Gas Treatment Instrument Air System," provides compressed air to systems and components that function to limit fission product release and control the environment from which the unit can be safely operated following a DBA.
The licensee states that these TS are related to assuring the appropriate functional capability of plant, equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in Modes 1 through 3. After the certifications required by 10 CFR 50.82(a)(1) are docketed, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel.

Additionally, the CR ventilation functions that are credited in the mitigation of the FHA until 19 days after permanent shutdown will no longer be required at the effective date of this amendment, which is no earlier than 19 days following permanent shutdown. Therefore, the licensee states that the listed TS are no longer applicable and the proposed deletion of the listed TS in Section 3.7, including associated SRs, is acceptable. The licensee also states that corresponding TS Bases will also be deleted.

The NRC staff finds the deletion of the listed TS in Section 3.7 acceptable.

4.4.16 Proposed Changes to TS 3.8 - "Electrical Power Systems"

The LAR dated June 20, 2019, proposed to either retain, revise, or delete, the DAEC electrical power systems TS in the permanently defueled condition. However, a supplement dated September 12, 2019, proposed to delete all of TS Section 3.8.

In particular, the subsections of TS Section 3.8 proposed to be deleted are listed as follows:

- TS 3.8.1 – [Alternating Current] AC Sources – Operating - Deleted
    - Applicability: MODES 1, 2, and 3

- TS 3.8.2 – AC Sources – Shutdown - Deleted
    - Modes 4 and 5, and during movement of irradiated fuel assemblies in the secondary containment

- TS 3.8.3 – Diesel Fuel Oil, Lube Oil, and Starting Air - Deleted
    - Applicability: When associated DG is required to be OPERABLE.

- TS 3.8.4 – [Direct Current] DC Sources – Operating - Deleted
    - Applicability: MODES 1, 2, and 3

- TS 3.8.5 – DC Sources – Shutdown - Deleted
    - Applicability: Modes 4 and 5, and during movement of irradiated fuel assemblies in the secondary containment

- TS 3.8.6 – Battery Cell Parameters - Deleted
    - Applicability: When associated DC electrical power sources are required to be OPERABLE.

- TS 3.8.7 – Distribution Systems– Operating - Deleted
    - Applicability: Modes 1, 2, and 3

- TS 3.8.8 – Distribution Systems– Shutdown - Deleted
    - Applicability: Modes 4 and 5, and during movement of irradiated fuel assemblies in the secondary containment

The DAEC electrical power systems TS 3.8 contain LCOs, actions and SRs that specify the requirements for assuring the appropriate functional capability of plant electrical power systems required for safe operation and shutdown of the facility. The requirements of the TS 3.8 Actions and SRs ensure that the TS 3.8 LCOs are met. The TS 3.8 LCOs are established for electrical power systems that satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C), which states that TS LCOs must be established for SSCs that are part of the primary success path and which function or actuate to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

In the LAR dated June 20, 2019, the licensee proposed to delete the DAEC electrical power systems TSs 3.8.1, 3.8.4, and 3.8.7 (relating to Operating Modes 1 through 3). The licensee stated that these TSs are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in Modes 1 through 3. After the reactor is defueled, these TSs will no longer be applicable.

The NRC staff reviewed the proposed deletion of the above-mentioned TS subsections to ensure that these will no longer be required for the electrical power systems based on Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) after the DAEC is permanently shut down and defueled. Since the reference to Modes 1 through 3 will no longer be relevant to DAEC in the permanently shut down and defueled condition, the NRC staff finds that the proposed deletion of TSs 3.8.1, 3.8.4, and 3.8.7 from the DAEC TS in the permanently shut down and defueled condition is acceptable.

The DAEC TSs 3.8.2, 3.8.3, 3.8.5, 3.8.6, and 3.8.8, were retained or changed in the LAR dated June 20, 2019, generally to allow the movement of irradiated fuel assemblies in the secondary containment. Applicability in Modes 4 and 5 was proposed to be deleted in TSs 3.8.2, 3.8.5, and 3.8.8; however, applicability of these TSs "during movement of irradiated fuel assemblies in the secondary containment" was proposed to be retained.

TS 3.8.3 was not changed in the LAR dated June 20, 2019.

<u>Deletion of TS 3.8.3, "Diesel Fuel Oil, Lube Oil, and Starting Air"</u>

TS 3.8.3, "Diesel Fuel Oil, Lube Oil, and Starting Air," is applicable when the associated diesel generator is required to be operable. These systems ensure proper operation of the emergency diesel generators by maintaining the quality of the fuel and lube oils and adequate starting air capacity.

The licensee states that TS 3.8.3 is related to assuring the appropriate functional capability of plant, equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility only when the reactor is in Modes 1 through 3. After the certifications required by 10 CFR 50.82(a)(1) are docketed, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel. Therefore, the licensee states that TS 3.8.3 is no longer applicable and the proposed deletion of TS 3.8.3, including associated SRs, is acceptable. The licensee also states that corresponding TS Bases will also be deleted.

The NRC staff finds the deletion of the listed TS in Section 3.8.3 acceptable.

TS 3.8.6 was retained in the LAR dated June 20, 2019, except for the removal of the requirement to verify the battery cell parameters for the 250 VDC batteries, since the 250 VDC batteries were not needed to mitigate the consequences of a postulated FHA.

By supplement dated September 12, 2019, the licensee proposed to delete the TSs in Section 3.8 in their entirety. In the supplement, the licensee stated that subsequent to submitting the LAR dated June 20, 2019, NEDA performed an analysis of an FHA in the SFP. This analysis determined that after a decay period of 19 days, CBEV is not required to maintain FHA dose consequences for CR occupants below the acceptance criteria of 10 CFR 50.67(b)(2)(iii). In the supplement, the licensee stated, "NEDA requests that the approved amendment, as supplemented, become effective not less than 19 days after plant shutdown and following docketing of the certifications required by 10 CFR 50.82(a)(1)(i) and (ii)." Parts of TSs 3.8.2, 3.8.3, 3.8.5, 3.8.6, and 3.8.8, and their related SRs associated with CBEV and support equipment were initially retained in the LAR dated June 20, 2019, but based on the results of the FHA analysis they were proposed for deletion in the supplement dated September 12, 2019.

With the DAEC in the permanently shut down and defueled condition for at least a 19-day decay period, (1) the electrical power systems in TSs 3.8.2, 3.8.3, 3.8.5, 3.8.6, and 3.8.8, will no longer be required to mitigate an FHA during movement of irradiated fuel assemblies and (2) operating Modes 4 and 5 will no longer be relevant to DAEC. As such, the NRC staff finds that the subject electrical power systems will no longer need to satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C). Therefore, the NRC staff finds that the proposed deletion of the rest of the TS subsections in Section 3.8 (i.e., TSs 3.8.2, 3.8.3, 3.8.5, 3.8.6, and 3.8.8) is also acceptable.

The NRC staff reviewed the proposed deletion of the DAEC electrical power systems TSs 3.8.1, 3.8.2, 3.8.3, 3.8.4, 3.8.5, 3.8.6, 3.8.7, and 3.8.8, to support the DAEC planned permanent shutdown and defueled condition. The NRC staff concludes that the proposed deletion of the above-mentioned TSs is acceptable because once the DAEC is permanently shut down and defueled, the electrical power systems will no longer be needed to mitigate a DBA or transient, as required by Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C), and as such, these electrical power systems TS will no longer be required for inclusion in the DAEC TS.

4.4.17 Proposed Deletion of TS 3.9 - "Refueling Operations"

<u>Deletion of TS 3.9.1 - "Refueling Equipment Airlocks"</u>

TS 3.9.1, "Refueling Equipment Interlocks," is applicable during in-vessel fuel movement with equipment associated with the interlocks when the reactor mode switch is in the Refuel position. Refueling equipment interlocks restrict the operation of the refueling equipment or the withdrawal of control rods to serve as a backup to procedural core reactivity controls to prevent the reactor from achieving criticality during refueling.

The licensee states that TS 3.9.1 is related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe refueling operation of the facility only when the reactor is in Mode 5 or during movement of fuel assemblies within the RPV. After the certifications required by 10 CFR 50.82(a)(1) are docketed, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel. Consequently, the licensee states that entry into Mode 5 and fuel movement within the RPV are not authorized. Therefore, the licensee states that TS 3.9.1 is no longer applicable and the proposed deletion of TS 3.9.1, including associated SRs, is acceptable. The licensee states that the corresponding TS Bases will also be deleted.

The NRC staff finds the deletion of TS 3.9.1 acceptable.

4.4.18 Proposed Deletion of TS 3.10 - "Special Operations"

This section contains LCOs that provide for appropriate functional capability of parameters and equipment required for mitigation of DBAs during specific operating scenarios.

The licensee proposed to delete the following TSs and their corresponding LCOs and associated SRs.

- TS 3.10.1 – System Leakage and Hydrostatic Testing Operation
- TS 3.10.2 – Reactor Mode Switch Interlock Testing
- TS 3.10.3 – Single Control Rod Withdrawal – Hot Shutdown
- TS 3.10.4 – Single Control Rod Withdrawal – Cold Shutdown
- TS 3.10.5 – Single Control Rod Drive (CRD) Removal – Refueling
- TS 3.10.6 – Multiple Control Rod Withdrawal – Refueling
- TS 3.10.7 – Control Rod Testing – Operating
- TS 3.10.8 – Shutdown Margin (SDM) Test – Refueling

TS 3.10.1, "System Leakage and Hydrostatic Testing Operation," allows certain reactor coolant pressure (RCP) tests to be performed in Mode 4 when the metallurgical characteristics of the RPV require the pressure testing at temperatures > 212 °F or to allow completing these reactor pressure tests when the initial conditions do not require temperatures > 212 °F. TS 3.10.1 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.10.2, "Reactor Mode Switch Interlock Testing," permits operation of the reactor mode switch from one position to another to confirm certain aspects of associated interlocks during periodic tests and calibrations in Modes 3, 4, and 5. TS 3.10.2 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.10.3, "Single Control Rod Withdrawal - Hot Shutdown," is applicable in Mode 3 with the reactor mode switch in the Refuel position. The purpose is to permit the withdrawal of a single control rod for testing while in Hot Shutdown by imposing certain restrictions. TS 3.10.3 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.10.4, "Single Control Rod Withdrawal - Cold Shutdown," is applicable in Mode 4 with the reactor mode switch in the Refuel position. The purpose to permit the withdrawal of a single control rod for testing while in Cold Shutdown, by imposing certain restrictions. TS 3.10.4 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.10.5, "Single Control Rod Drive Removal - Refueling," permits the removal of a single CRD during refueling operations by imposing certain administrative controls. TS 3.10.5 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.10.6, “Multiple Control Rod Withdrawal - Refueling,” is applicable in Mode 5 with LCOs 3.9.3, 3.9.4, or 3.9.5, not met. The purpose is to permit multiple control rod withdrawal during refueling by imposing certain administrative controls. TS 3.10.6 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.10.7, “Control Rod Testing - Operating,” permits control rod testing, while in Modes 1 and 2, by imposing certain administrative controls. TS 3.10.7 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

TS 3.10.8, “Shutdown Margin Test - Refueling,” permits SDM testing to be performed for those plant configurations in which the RPV head is either not in place or the head bolts are not fully tensioned. TS 3.10.8 does not apply once the reactor is permanently defueled; therefore, the TS is proposed to be deleted.

The licensee proposes the deletions of the above TSs in their entirety. These TSs and associated LCOs and SRs are needed for appropriate functional capability of RCS equipment required for safe operation of the facility. The staff reviewed the TSs proposed for deletion as well as the associated bases to ensure that they no longer needed to satisfy the 10 CFR 50.36 criteria for inclusion in TSs. With the TS sections deleted in their entirety, the applicable bases and surveillance section can also be removed. These specifications do not apply to the safe storage and handling of spent fuel in the SFP. Once the DAEC license no longer authorizes the use of the facility for power operation, or emplacement, or retention of fuel into the reactor vessel as provided in 10 CFR 50.82(a)(2), the TSs described above are no longer applicable.

NRC Assessment

The NRC staff examined the TSs proposed for deletion. The TSs are related to assuring the appropriate functional capability of plant equipment, and control of process variables, design features, or operating restrictions required for safe operation of the facility during specific operating scenarios. Since 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant or placing fuel in the reactor vessel, these TSs are no longer applicable. Therefore, the proposed deletions are appropriate and acceptable.

4.5 TS Section 4.0 – “Design Features”

This section contains a brief description of the DAEC location, description, and requirements for the DAEC reactor core and a description of and requirements for fuel storage at DAEC.

The current TS 4.1, “Site Location,” states that following:

> The plant site, which consists of approximately 500 acres, is adjacent to the Cedar River approximately 2.5 miles northeast of the Village of Palo, Iowa. Distance from the reactor centerline to the nearest site boundary is approximately 2000 ft. The boundary of the exclusion area defined in 10 CFR 100 is delineated by the property lines. The distance to the outer boundary of the low population zone is 6 miles. The plan of the site is shown on UFSAR Figures 1.2-1 and 1.2-2.

The licensee proposed to revise TS 4.1, Site Location, to state the following:

> The plant site, which consists of approximately 500 acres, is adjacent to the Cedar River approximately 2.5 miles northeast of the Village of Palo, Iowa. The boundary of the exclusion area defined in 10 CFR 100 is delineated by the property lines. The distance

to the outer boundary of the low population zone is 6 miles. The plan of the site is shown on UFSAR Figures 1.2-1 and 1.2-2.

The licensee proposed to delete TS 4.2, "Reactor Core."

The licensee proposed deletion of the following TS design features related to the reactor core, in their entirety:

TS 4.2 – "Reactor Core"

TS 4.2 provides a description and requirements regarding the fuel assemblies and control rod assemblies. Since the design features described above no longer provide a function and are no longer required, this TS may be deleted. This TS is not proposed for inclusion in the PDTS since the equipment is not required in the permanently defueled condition and the DAEC license will no longer be authorized for power operation in a permanently defueled condition.

Because fuel will no longer be authorized, the licensee proposed deletion of Section 4.2 in its entirety.

Therefore, based on the above, the NRC staff finds the deletion of Section 4.2 in its entirety, to be acceptable.

The current TS 4.3, "Fuel Storage," states that following:

4.3.1 Criticality

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum k-infinity of 1.31 in the normal reactor core configuration at cold conditions;
b. $K_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;
c. $K_{eff} \leq 0.90$ if dry, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and
d. A nominal 6.625 inch center to center distance between fuel assemblies placed in storage racks.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2563 fuel assemblies in a vertical orientation, including no more than 152 fuel assemblies stored in the cask pit in accordance with UFSAR Section 9.1.

The new fuel storage vault is equipped with racks for storage of up to 110 fuel assemblies in a vertical orientation.

The licensee proposed to revise TS 4.3 to state the following:

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2563 fuel assemblies in a vertical orientation, including no more than 152 fuel assemblies stored in the cask pit in accordance with UFSAR Section 9.1.

Modification of TS 4.3.3, “Capacity”

TS 4.3.3 contains a description of, and requirements for, capacity of the SFP and the new fuel storage vault. After the certifications required by 10 CFR 50.82(a)(1) are docketed, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel. Consequently, the licensee states that the DAEC will not be receiving or storing new fuel and the reference to the new fuel vault will no longer be applicable.

Therefore, the licensee proposes to revise TS 4.3.3 by deleting this reference which reads “The new fuel storage vault is equipped with racks for storage of up to 110 fuel assemblies in a vertical orientation.” TS 4.3.3 does not have associated Bases.

NRC Assessment

The NRC staff reviewed the proposed changes to TS 4.1 and concludes that the changes are consistent with the transition to a permanently shutdown and defueled facility. With the reactor permanently shut down and defueled, its location within the site boundary is inconsequential. Therefore, the proposed change to remove the reference to the reactor centerline is appropriate and acceptable.

The NRC staff reviewed the proposed deletion of TS 4.2. TS 4.2 contains descriptions of and requirement for fuel assemblies and control rod assemblies in the DAEC reactor core during operation of the unit. Since 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant or placing fuel in the reactor vessel, the operational reactor core as described in TS 4.2 will no longer exist. Therefore, the proposed deletions are appropriate and acceptable.

The NRC staff reviewed the proposed changes to TS 4.3 and concludes that changes are consistent with the transition to a permanently shutdown and defueled facility. The licensee is retaining TS 4.3.1.1 (only the subsection number 4.3.1.1 is deleted), without changes as it details the design and maintenance of the spent fuel storage racks. TS 4.3.1.2 details the design and maintenance of the new fuel storage racks. Since DAEC will not be receiving or storing new fuel, TS 4.3.1.2 will no longer be applicable. TS 4.3.2 ensures SFP water level is maintained to prevent inadvertent draining of the SFP and will be retained without changes. TS 4.3.3 contains a description of and requirements for capacity of the SFP and the new fuel storage vault. Since DAEC will not be receiving or storing new fuel, the reference to the new fuel vault will no longer be applicable and the reference will be deleted from TS 4.3.3. Based on these reasons, the NRC staff finds that the proposed changes are appropriate and acceptable.

4.6 Proposed Change to TS – TS Section 5.5 - "Programs and Manuals"

The TS 5.5, "Programs and Manuals," provides a description and requirements regarding programs and manuals that are to be established, implemented, and maintained. The licensee proposes to delete or modify the following sections:

Deletion of TS 5.5.2 - "Primary Coolant Sources Outside Containment"

This program was established to provide controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable.

This program is proposed for deletion. Once the DAEC is permanently shut down and defueled, there will no longer be any transient or accident conditions associated with primary coolant sources. Thus, TS 5.5.2 will not be retained and NRC staff finds deletion of TS 5.5.2 to be acceptable.

Deletion of TS 5.5.5 - "Component Cyclic or Transient Limit"

The NRC staff noted that this section provides controls to track cyclic and transient occurrences to ensure components are maintained within the design limit. Maintaining these transient occurrences is performed to ensure the reactor vessel integrity.

The NRC staff finds continued implementation of this TS will no longer be necessary because the licensee has decided to permanently cease power operations at DAEC, and the RPV will cease to experience the transient occurrences (e.g., heatup, cooldown) that bound the reactor vessel transient design. Based on its review of the proposed deletion, the NRC staff concludes that continued implementation of TS Section 5.5.5 will no longer be necessary in accordance with 10 CFR 50.60(a) because power operation will no longer be authorized once the 10 CFR 50.82(a)(1) certifications have been docketed. Therefore, the NRC staff finds the deletion of TS Section 5.5.5 acceptable.

This program was established to provide controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable.

Deletion of TS 5.5.7- "Ventilation Filter Testing Program [VFTP]"

The licensee states that this program was established to implement required testing of engineered safety feature (ESF) filter ventilation systems which included the SBGT and SFU systems.

Once the DAEC is permanently shut down and defueled and 19 days of decay have elapsed, these systems are no longer required to mitigate the consequences of an FHA. Therefore, the TS associated with the SBGT system (TS 3.6.4.3) and SFU system (TS 3.7.4) have been deleted and the need for a filter testing program for these systems is no longer required.

The NRC staff finds the deletion of TS 5.5.7 acceptable.

Modification of TS 5.5.8 - "Explosive Gas and Storage Tank Radioactivity Monitoring Program"

The licensee proposes to retain TS 5.5.8 with changes. This program provides controls for potentially explosive gas mixtures contained in the Offgas system downstream of the recombiners and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

In a permanently shut down and defueled condition, the licensee states the Offgas system is no longer required to be operable and has deleted the associated TS. Therefore, the licensee proposes the deletion of the monitoring requirements for potentially explosive gas mixtures contained in the Offgas system from TS 5.5.8. The monitoring requirements for radioactivity contained in unprotected outdoor liquid storage tanks remain unchanged.

The licensee proposes to change the title of TS 5.5.8 from "Explosive Gas and Storage Tank Radioactivity Monitoring Program" to "Storage Tank Radioactivity Monitoring Program." The licensee also provided other edits to TS 5.5.8 that remove all references to explosive gas monitoring.

The NRC staff finds the modification of TS 5.5.8 acceptable.

Deletion of TS 5.5.9 - "Diesel Fuel Oil Testing Program"

This program ensures acceptability of both new fuel oil and stored fuel oil. Once the DAEC is permanently shut down and defueled and 19 days of decay have elapsed, the licensee states the diesel generators are no longer required to mitigate the consequences of an FHA and has therefore deleted the TS associated with the diesel generators and diesel generator fuel. Therefore, the licensee states that a fuel oil testing program is no longer necessary.

The NRC staff finds the deletion of TS 5.5.9 acceptable.

Deletion of TS 5.5.11 - "Safety Function Determination Program [SFDP]

The SFDP ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

Once the DAEC is permanently shut down and defueled, and 19 days of decay have elapsed, the only required TS LCO will be TS 3.7.8, - "Spent Fuel Pool Level Indication." TS 3.7.8 is not included in the SFDP and all TSs included in the SFDP are proposed for deletion after the DAEC is permanently shut down. Therefore, this program is no longer necessary.

The NRC staff agrees with the determination to delete TS 5.5.11.

Deletion of TS 5.5.12 - "Primary Containment Leakage Rate Testing Program"

This program was established to implement the leakage rate testing of the primary containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions.

The licensee states that TS 5.5.12 is proposed for deletion because the 10 CFR Part 50 license will no longer authorize emplacement or retention of fuel in the reactor vessel once the certifications required by 10 CFR 50.82(a)(1) have been docketed. This program pertains only to verifying the operability of the containment systems. The TS for containment systems (TS Section 3.6) are being deleted per this amendment request and, therefore, the licensee states that TS 5.5.12 will no longer be necessary.

The NRC staff finds the deletion of TS 5.5.12 to be acceptable.

Deletion of TS 5.5.13 - “Control Building Envelope Habitability Program”

The control building envelope (CBE) habitability program ensures that the CBE is maintained. Once the DAEC is permanently shut down and defueled and 19 days of decay have elapsed, the licensee states the CBE is no longer required to mitigate the consequences of an FHA. The TS associated with maintaining the CBE (TS within Section 3.7) are being deleted per this amendment request. Therefore, the licensee states that a program to ensure CBE habitability is maintained will no longer be necessary and that TS 5.5.13 may be deleted.

The NRC staff finds the deletion of TS 5.5.13 acceptable.

SFP NAM Monitoring Program Licensee Description

As stated above, TS 5.5.15 describes the licensee’s NAM monitoring program. The program contains provisions for routine monitoring and corrective actions to ensure the Boral can meet its neutron attenuation capability as assumed in the SFP criticality safety analysis. These provisions include a combination of coupon and in-situ neutron attenuation testing to measure the ^{10}B AD of the Boral that is compared against an acceptance criterion set at the value assumed in the criticality safety analysis. This value can also be found in TS 4.3.1. Additionally, the coupon tests are performed at a frequency not to exceed 6 years, and the in-situ testing at a frequency not to exceed 10 years. The TS also states that the program will describe appropriate corrective actions if nonconforming Boral is discovered.

The current amendment did not propose any changes to the NAM monitoring program that was approved in the 2017 LAR.

NRC Staff Evaluation of NAM Monitoring Program

The NRC staff reviewed the DAEC NAM monitoring program as there may be spent fuel in the SFP after DAEC permanently ceases operations and defuels the reactor. The staff noted that the NAM monitoring program contains provisions to ensure the Boral doesn’t degrade and can continue to perform its neutron attenuation function as described in the criticality safety analysis in Section 9.1 of the DAEC UFSAR (TS 4.3.1, “Criticality,” states the requirements for the ^{10}B AD including measurement uncertainty).

The NRC staff finds the NAM monitoring program acceptable because TS 5.5.15 requires the licensee to perform neutron attenuation testing (coupon for the Holtec racks and in-situ for the PaR racks) within the stated intervals. The coupon testing will also allow physical parameters to be measured for signs of degradation. The acceptance criteria as described in TS 4.3.1 and 5.5.15 are acceptable because these require the licensee to take corrective actions if the ^{10}B AD is found to be below the minimum value assumed in the criticality safety analysis.

Additionally, as described in the 2017 license amendment, the licensee will trend the results of the monitoring program to detect potential degradation prior to challenging the acceptance criteria. These aspects of the monitoring program provide reasonable assurance that the ^{10}B AD of the NAM will not fall short of the minimum value assumed in the criticality safety analysis and therefore, in part, demonstrate compliance with 10 CFR 50.68 after DAEC has permanently ceased operations and has defueled the reactor.

The licensee has not proposed any changes to the NAM monitoring program found in TS 5.5.15. The NRC staff finds this acceptable as the staff had previously reviewed and approved this TS and its description of the NAM monitoring program. This TS provides reasonable assurance that the licensee will continue to monitor the condition of the NAM and take appropriate corrective actions if it degrades.

Deletion of TS Section 5.6.6 - "Post Accident Monitoring Report [PAM]"

The primary purpose of the PAM instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to determine that the actions automatically initiated by the ESF equipment have successfully accomplished their safety functions for design basis events.

TS 3.3.3.1 - "Post Accident Monitoring Instrumentation," ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior following an accident, and is applicable in Modes 1 and 2. In accordance with TS section 5.6.6, a PAM report is required by Conditions B or F of LCO 3.3.3.1.

As stated before, TS Section 3.3.3 is required for safe operation of the facility only when the reactor is in Modes 1 and 2. Once DAEC is placed in a defueled condition in accordance with 10 CFR 50.82(a)(2), the 10 CFR Part 50 license will no longer authorize operation, placement or retention of fuel in the reactor vessel, and entry into any of the listed applicable modes will no longer be possible.

Therefore, the NRC staff finds that the required conditions to prepare and submit a PAM report no longer apply and agrees with the deletion of TS 5.6.6.

4.7 Summary

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The staff finds that the licensee's proposed changes use analysis methods and assumptions consistent with the guidance contained in RG 1.183. The NRC staff compared the doses estimated by the licensee to the applicable criteria. The NRC staff finds with reasonable assurance that DAEC, as modified by this proposed change, will continue to provide sufficient safety margins with adequate defense-in-depth to address unanticipated events and to compensate for uncertainties in accident progression and in analysis assumptions and parameters. The staff concludes that the licensee has demonstrated that the dose consequences for postulated accidents at the permanently defueled DAEC would not have consequences that could potentially exceed the applicable 10 CFR 100.11 and 10 CFR 50.67 dose limits and RG 1.183 dose acceptance criteria.

Therefore, the NRC staff finds the proposed changes to be acceptable from a dose consequence perspective.

The NRC staff reviewed the portion of the LAR related to the DAEC NAM monitoring program and has determined that the monitoring program as described in the DAEC TS, and previously approved license amendment, will provide reasonable assurance that the licensee will be able to detect degradation of the neutron absorbing material before its ability to perform its intended safety function is impacted. On this basis, the staff concluded that the continuation of the NAM monitoring program and the contents of the program after permanent cessation of reactor operations meet the applicable requirements of 10 CFR 50.68, and GDCs 61 and 62, and are therefore acceptable.

Once certifications required by 10 CFR 50.82(a)(1) are docketed for DAEC, the 10 CFR Part 50 license will no longer authorize operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). Therefore, based on its review, the NRC staff concludes the deletion of the proposed license item and TSs discussed above are acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Iowa official was notified of the proposed issuance of the amendment on May 11, 2020. On May 19, 2020, the State official confirmed that the State of Iowa had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

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 or changes the 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 and there has been no public comment on such finding (84 FR 66232). Accordingly, the 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 need be prepared in connection with the issuance of the amendment.

7.0 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.

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SUBJECT: DUANE ARNOLD ENERGY CENTER – ISSUANCE OF AMENDMENT NO. 311 RE: PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS (EPID L-2019-LLA-0130) DATED JULY 10, 2020

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