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

March 6, 2018

Ms. Mary J. Fisher
Vice President, Energy Production and
Nuclear Decommissioning
Omaha Public Power District
Fort Calhoun Station
9610 Power Lane, Mail Stop FC-2-4
Blair, NE 68008

SUBJECT: FORT CALHOUN STATION, UNIT 1 – ISSUANCE OF AMENDMENT RE: REVISED TECHNICAL SPECIFICATIONS TO ALIGN TO THOSE REQUIREMENTS FOR DECOMMISSIONING (CAC NO. MF9567; EPID L-2017-LLA-0192)

Dear Ms. Fisher:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 297 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit 1 (FCS). The amendment consists of changes to the FCS renewed facility operating license and the Technical Specifications (TSs) in response to your application dated March 31, 2017, as supplemented by letter dated September 26, 2017.

The amendment revises the renewed facility operating license and associated TSs to reflect the permanent cessation of reactor operations and the permanently defueled condition of the reactor vessel at FCS. In general, the changes eliminate those TSs applicable in operating MODES; MODES where fuel is emplaced in the reactor vessel, and certain TSs required for movement of irradiated fuel assemblies. Changes were also made to the TS definitions, administrative controls, and related to programs and procedures. The amendment also revises the renewed facility operating license to clarify or remove certain conditions no longer relevant and add conditions consistent with other permanently shutdown and defueled reactors.

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

Sincerely,

James Kim

James Kim, Project Manager
Special Projects and Process Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures:
1. Amendment No. 297 to DPR-40
2. Safety Evaluation

cc: Listserv

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

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 297
Renewed License No. DPR-40

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

   A. The application for amendment by the Omaha Public Power District (the licensee), dated March 31, 2017, as supplemented by letter dated September 26, 2017, 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 license 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, Renewed Facility Operating License No. DPR-40 is hereby amended, as follows, and as indicated in the attachment to this license amendment:

- The title "RENEWED FACILITY OPERATING LICENSE NO. DPR-40" is amended to read "RENEWED FACILITY LICENSE NO. DPR-40".

- Paragraph 1.C of Renewed Facility License No. DPR-40 is hereby amended to read as follows:

  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;

- Paragraph 1.H of Renewed Facility License No. DPR-40 is hereby amended to read as follows:

  H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the Commission concludes that the issuance of Renewed License No. DPR-40 is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

- Paragraph 2 of Renewed Facility License No. DPR-40 is hereby amended to read as follows:

  2. On the basis of the forgoing findings regarding this facility, Facility Operating License No. DPR-40, issued August 9, 1973, is superseded by Renewed Facility License No. DPR-40, which is hereby issued to the Omaha Public Power District, to read as follows:

- Paragraph 2.A of Renewed Facility License No. DPR-40 is hereby amended to read as follows:

  A. This renewed license applies to the Fort Calhoun Station, Unit 1, a pressurized water nuclear reactor and associated equipment (the facility), which is owned by the Omaha Public Power District. The facility is located in Washington County, Nebraska, and is described in the Final Safety Analysis Report as supplemented, amended, and updated and the Environmental Report as supplemented and amended.

- Paragraph 2.B.(1) of Renewed Facility License No. DPR-40 is hereby amended to read as follows:

  (1) 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 fuel storage at the designated location in Washington County, Nebraska in accordance with the procedures and limitations set forth in this renewed license;

- Paragraph 2.B.(2) of Renewed Facility License No. DPR-40 is hereby amended to read as follows:

  (2) Pursuant to the Act and 10 CFR Parts 40 and 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 Final Safety Analysis Report, as supplemented, amended, and updated;

- Paragraph 2.B.(3) of Renewed Facility License No. DPR-40 is hereby amended to read as follows:

  (3) 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 and reactor instrumentation; and fission detectors;

- Paragraph 3.A of Renewed Facility License No. DPR-40 is hereby amended to read as follows:

  A. DELETED

- Paragraph 3.B of Renewed Facility License No. DPR-40 is hereby amended to read as follows:

  B. <u>Technical Specifications</u>

  The Technical Specifications contained in Appendix A, as revised through Amendment No. 297, are hereby replaced with the Permanently Defueled Technical Specifications (PDTS). Omaha Public Power District shall maintain the facility in accordance with the Permanently Defueled Technical Specifications.

- Paragraph 3.E of Renewed Facility License No. DPR-40 is hereby amended to read as follows:

  E. DELETED

- Paragraph 3.F of Renewed Facility License No. DPR-40 is hereby amended to read as follows:

  F. DELETED

- Paragraph 4 of Renewed Facility License No. DPR-40 is hereby amended to read as follows:

  4. This license is effective as of the date of issuance and authorizes ownership and possession of Fort Calhoun Station until the Commission notifies the licensee in writing that the license is terminated.

- The Attachments of Renewed Facility License No. DPR-40 are hereby amended to read as follows:

  Attachments: 1. Appendix A – Permanently Defueled Technical Specifications
  2. Appendix B – Deleted

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

N. Salgado for

Douglas A. Broaddus, Chief
Special Projects and Process Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility License No. DPR-40 and Technical Specifications

Date of Issuance: March 6, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 297 TO

RENEWED FACILITY LICENSE NO. DPR-40

FORT CALHOUN STATION, UNIT 1

DOCKET NO. 50-285

Replace the following pages of the Renewed Facility License No. DPR-40, Appendix A - Technical Specifications, and Appendix B – Additional Conditions with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Renewed Facility License No. DPR-40

REMOVE	INSERT
-1- to -8-	-1- to -4-

Appendix A - Technical Specifications

REMOVE	INSERT
TOC - Page 1 to Page 6	TOC – Page 1 to Page 3
Definitions – Page 1 to Page 9	Definitions - Page 1 to Page 2
1.0 – Page 1 to Page 5	--
2.0 – Page 1 to Page 3	--
2.1 – Page 1 to Page 24	--
2.2 – Page 1 to Page 12	--
2.3 – Page 1 to Page 8	--
2.4 – Page 1 to Page 5	--
2.5 – Page 1 to Page 3	--
2.6 – Page 1 to Page 3	--
2.7 – Page 1 to Page 9	--
2.8 – Page 1 to Page 29	2.8 – Page 1 to Page 6
2.9 – Page 1	--
2.10 – Page 1 to Page 20	--
2.12 – Page 1 to Page 9	--
2.13 – Page 1 to Page 3	--
2.14 – Page 1 to Page 5	--
2.15 – Page 1 to Page 18	--
2.16 – Page 1 to Page 2	--
2.20 – Page 1	--
2.21 – Page 1 to Page 5	--
2.22 – Page 1	--
2.23 – Page 1 to Page 4	--
3.0 – Page 1 to Page 5	3.0 – Page 1 to Page 4
3.1 – Page 1 to Page 23	--
3.2 – Page 1 to Page 13	3.2 – Page 1 to Page 3
3.3 – Page 1 to Page 3	3.3 – Page 1
3.4 – Page 1	--

Appendix A - Technical Specifications

REMOVE	INSERT
3.5 – Page 1 to Page 4	--
3.6 – Page 1 to Page 7	--
3.7 – Page 1 to Page 4	--
3.8 – Page 1	--
3.9 – Page 1 to Page 3	--
3.10 – Page 1 to Page 3	--
3.12 – Page 1	--
3.16 – Page 1 to Page 2	--
3.17 – Page 1 to Page 2	--
4.0 – Page 1 to Page 3	4.0 – Page 1 to Page 2
5.0 – Page 1 to Page 21	5.0 – Page 1 to Page 11
6.0 – Page 1	--

Appendix B – Additional Conditions

REMOVE	INSERT
Appendix B – Page 1	--

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT 1

RENEWED FACILITY LICENSE NO. DPR-40

1. The U.S. Nuclear Regulatory Commission (the Commission) having previously made the findings set forth in License No. DPR-40 issued August 9, 1973, has now found that:

   A. The application to renew License No. DPR-40 filed by Omaha Public Power District (the licensee) 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 1, and all required notifications to other agencies or bodies have been duly made;

   B. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21 (a)(1), and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21 (c), such that there is reasonable assurance that the activities authorized by this renewed license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for Fort Calhoun Station, Unit No. 1, and that any changes made to the plant's current licensing basis in order to comply with 10 CFR 54.29(a) are in accord with the Act and the Commission's regulations;

   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: (1) that the activities authorized by this renewed license can be conducted without endangering the health and safety of the public, and (2) that such activities will be conducted In compliance with the rules and regulations of the Commission;

   E. Omaha Public Power District Is technically qualified and financially qualified to engage in the activities authorized by this renewed license in accordance with the rules and regulations of the Commission;

   F. Omaha Public Power District has 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 Commission concludes that the issuance of Renewed License No. DPR-40 is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by the renewed license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40, and 70, including but not necessarily limited to 10 CFR Sections 30.33, 40.32, 70.23 and 70.31.

2. On the basis of the forgoing findings regarding this facility, Facility Operating License No. DPR-40, issued August 9, 1973, is superseded by Renewed Facility License No. DPR-40, which is hereby issued to the Omaha Public Power District, to read as follows:

A. This renewed license applies to the Fort Calhoun Station, Unit 1, a pressurized water nuclear reactor and associated equipment (the facility), which is owned by the Omaha Public Power District. The facility is located in Washington County, Nebraska, and is described in the Final Safety Analysis Report as supplemented, amended, and updated and the Environmental Report as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Omaha Public Power District:

(1) 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 fuel storage at the designated location in Washington County, Nebraska in accordance with the procedures and limitations set forth in this renewed license;

(2) Pursuant to the Act and 10 CFR Parts 40 and 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 Final Safety Analysis Report, as supplemented, amended, and updated;

(3) 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 and reactor instrumentation; and fission detectors;

(4) 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 when associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

3. 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, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and 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:

A. DELETED

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 297, are hereby replaced with the Permanently Defueled Technical Specifications (PDTS). Omaha Public Power District shall maintain the facility in accordance with the Permanently Defueled Technical Specifications.

C. Security and Safeguards Contingency Plans

The Omaha Public Power District shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Fort Calhoun Station Security Plan, Training and Qualification Plan, Safeguards Contingency Plan," submitted by letter dated May 19, 2006.

OPPD shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The OPPD CSP was approved by License Amendment No. 266 and modified by License Amendment No. 284 and Amendment No. 294.

D. DELETED

E. DELETED

F. DELETED

G. Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

4. This license is effective as of the date of issuance and authorizes ownership and possession of Fort Calhoun Station until the Commission notifies the licensee in writing that the license is terminated.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
J. E. Dyer

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments: 1. Appendix A – Permanently Defueled Technical Specifications
2. Appendix B – Deleted

Date of Issuance: November 4, 2003

PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS

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PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS - TABLES

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TABLE **SECTION**

PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS - FIGURES

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PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS

DEFINITIONS

The following terms are defined for uniform interpretation of these Specifications.

FUEL STORAGE AND HANDLING CONDITIONS

Fuel Handling Operations
Any operation involving the shuffling, removal, or replacement of irradiated fuel. The suspension of any FUEL HANDLING OPERATIONS shall not preclude completion of movement of a component to a safe, conservative position.

Certified Fuel Handler (CFH)

A CERTIFIED FUEL HANDLER is an individual who complies with provisions of the CERTIFIED FUEL HANDLER training program required by Technical Specification 5.4.2.

Non-Certified Operator (NCO)

A NON-CERTIFIED OPERATOR is a non-licensed operator who complies with the applicable training requirements of Technical Specification 5.4.1, but is not a CERTIFIED FUEL HANDLER.

MISCELLANEOUS DEFINITIONS

Actions

ACTIONS shall be that part of a specification that prescribes required actions to be taken under designated conditions within specified completion times.

Operable – Operability

A system, subsystem, train, 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 sources, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its specified safety functions(s) are also capable of performing their related support function(s).

DEFINITIONS
MISCELLANEOUS DEFINITIONS (continued)

Offsite Dose Calculation Manual (ODCM)

The document(s) that contain the methodology and parameters used in the calculations of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent radiation monitoring Warn/High (trip) Alarm setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain:

1) The Radiological Effluent Controls and the Radiological Environmental Monitoring Program required by Specification 5.16.

2) Descriptions of the information that should be included in the Annual Radiological Environmental Operating Reports and Annual Radioactive Effluent Release Reports required by Specifications 5.9.4.a and 5.9.4.b.

Unrestricted Area

Any area at or beyond the site boundary access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

Sections 1.0 through 2.7 have been deleted in their entirety.

2.0 LIMITING CONDITIONS FOR OPERATION

2.8 Fuel Handling

2.8.1 DELETED

2.8.2 DELETED

2.8.3 Fuel Handling Operations - Spent Fuel Pool

2.8.3(1) Spent Fuel Assembly Storage

Applicability

Applies to storage of spent fuel assemblies whenever any irradiated fuel assembly is stored in Region 2 (including peripheral cells) of the spent fuel pool.

Objective

To minimize the possibility of an accident occurring during FUEL HANDLING OPERATIONS that could affect public health and safety.

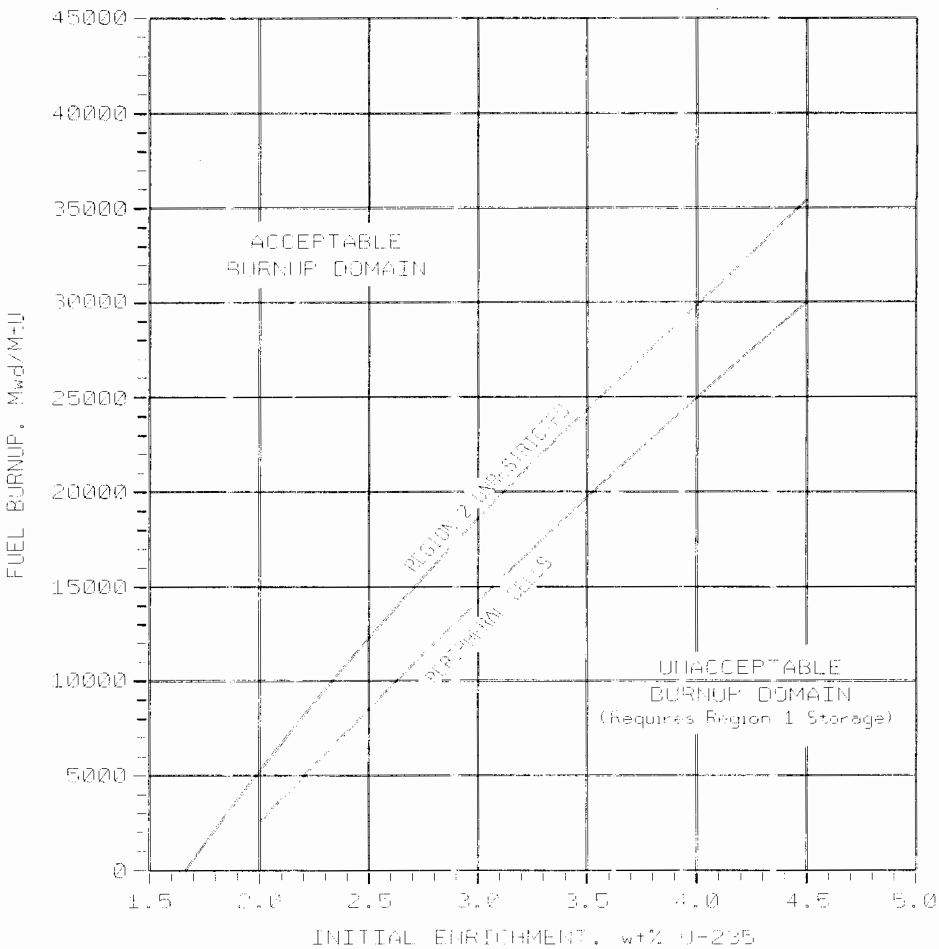
Specification

The combination of initial enrichment and burnup of each spent fuel assembly stored in Region 2 (including peripheral cells) of the spent fuel pool shall be within the acceptable burnup domain of Figure 2-10.

Required Actions

(1) With the requirements of the LCO not met, initiate action to move the noncomplying fuel assembly immediately.

FIGURE 2-10



LIMITING BURNUP CRITERIA FOR ACCEPTABLE STORAGE IN REGION 2

NOTES:

1. Any fuel assembly (≤ 4.5% average U-235 enrichment) mechanically coupled with a full length CEA may be located anywhere in Region 2.

2. Peripheral cells are those adjacent to the Spent Fuel Pool wall or the cask laydown area.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Fuel Handling
2.8.3 Fuel Handling Operations - Spent Fuel Pool (continued)

2.8.3(2) Spent Fuel Pool Water Level

Applicability

Applies to the water level of the spent fuel pool during FUEL HANDLING OPERATIONS in the spent fuel pool.

Objective

To minimize the consequences of a fuel handling accident during FUEL HANDLING OPERATIONS in the spent fuel pool that could affect public health and safety.

Specification

The spent fuel pool water level shall be ≥ 23 ft. above the top of irradiated fuel assemblies seated in the storage racks.

Required Actions

(1) With the spent fuel pool water level not within limits, suspend FUEL HANDLING OPERATIONS in the spent fuel pool immediately.

2.8.3(3) Spent Fuel Pool Boron Concentration

Applicability

Applies to the boron concentration of the spent fuel pool when fuel assemblies are stored in the spent fuel pool.

Objective

To minimize the possibility of an accident that could affect public health and safety from occurring when fuel assemblies are stored in the spent fuel pool.

Specification

The spent fuel pool boron concentration shall be ≥ 500 ppm.

Required Actions

(1) With the spent fuel pool boron concentration < 500 ppm, suspend FUEL HANDLING OPERATIONS in the spent fuel pool immediately, and

(2) Restore spent fuel pool boron concentration to ≥ 500 ppm immediately.

2.0 **<u>LIMITING CONDITIONS FOR OPERATION</u>**

2.8 Fuel Handling

2.8.3 Fuel Handling Operations - Spent Fuel Pool (continued)

2.8.3(4) DELETED

2.8.3(5) DELETED

2.8.3(6) DELETED

2.8.3(1) Spent Fuel Assembly Storage

The spent fuel pool is designed for noncriticality by use of neutron absorbing material. The restrictions on the placement of fuel assemblies within the spent fuel pool, according to Figure 2-10, and the accompanying LCO, ensures that the k_{eff} of the spent fuel pool always remains $<$ 0.95 assuming the pool to be flooded with unborated water.

A spent fuel assembly may be transferred to the spent fuel pool Region 2 provided an independent verification of assembly burnups has been completed and the assembly burnup meets the acceptance criteria identified in Figure 2-10. When the configuration of fuel assemblies stored in Region 2 (including the peripheral cells) is not in accordance with Figure 2-10, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure 2-10. Acceptable fuel assembly burnup is not a prerequisite for Region 1 storage because Region 1 will maintain any type of fuel assembly that the plant is licensed for in a safe, coolable, subcritical geometry.

When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner.

2.8.3(2) Spent Fuel Pool Water Level

The minimum water level in the spent fuel pool meets the assumption of iodine decontamination factors following a fuel handling accident. When the water level is lower than the required level, the movement of irradiated fuel assemblies in the spent fuel pool is immediately suspended. This effectively precludes a fuel handling accident from occurring in the spent fuel pool. Suspension of FUEL HANDLING OPERATIONS shall not preclude completion of movement of a component to a safe, conservative position. When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner.

2.8.3(3) Spent Fuel Pool Boron Concentration

The basis for the 500 ppm boron concentration requirement with Boral poisoned storage racks is to maintain the k_{eff} below 0.95 in the event a misloaded unirradiated fuel assembly is located next to a spent fuel assembly. A misloaded unirradiated fuel assembly at maximum enrichment condition, in the absence of soluble poison, may result in exceeding the design effective multiplication factor. A misloaded irradiated fuel assembly is bounded by this requirement. Soluble boron in the spent fuel pool water, for which credit is permitted under these conditions, would assure that the effective multiplication factor is maintained substantially less than the design condition.

This LCO applies whenever fuel assemblies are stored in the spent fuel pool. The boron concentration is periodically sampled in accordance with Specification 3.2. Sampling is performed periodically when fuel is stored in the spent fuel pool.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.8 Fuel Handling

Bases (continued)

2.8.3(3) Spent Fuel Pool Boron Concentration (continued)

When "immediately" is used as a completion time, the required action should be pursued without delay and in a controlled manner. Suspension of FUEL HANDLING OPERATIONS shall not preclude completion of movement of a component to a safe, conservative position.

References

(1) FSAR, as updated, Section 9.5
(2) FSAR, as updated, Section 14.18

Sections 2.9 through 2.23 have been deleted in their entirety.

3.0 SURVEILLANCE REQUIREMENTS

3.0.1 Each surveillance requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

3.0.2 The surveillance intervals are stated in the individual Specifications.

3.0.3 The provisions of Specifications 3.0.1 and 3.0.2 are applicable to all codes and standards referenced within the Technical Specifications. The requirements of the Technical Specifications shall have precedence over the requirements of the codes and standards referenced within the Technical Specifications.

3.0.4 Surveillance Requirements shall be met during the specified conditions in the individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. 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 OPERABILITY requirements for the corresponding Limiting Condition for Operation. Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specifications 3.0.1 and 3.0.2, shall constitute noncompliance with the OPERABILITY requirements for the corresponding Limiting Condition for Operation except as provided in Specification 3.0.5. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. Surveillance Requirements do not have to be performed on inoperable equipment.

3.0.5 If it is discovered that a Surveillance was not performed within its specified surveillance interval, then compliance with the requirement to declare the OPERABILITY requirements for the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, 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 OPERABILITY requirements for the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION(S) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the OPERABILITY requirements for the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION(S) must be entered.

3.0 SURVEILLANCE REQUIREMENTS

Basis

Specifications 3.0.1 through 3.0.5 establish the general requirements applicable to Surveillance Requirements.

Specification 3.0.1 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of station conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified. The limitation of Specification 3.0.1 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

The provisions of Specification 3.0.2 define the surveillance intervals for use in the Technical Specifications. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications. A few surveillance requirements have uncommon intervals. In such a case the surveillance interval shall be performed as defined by the individual specifications.

Specification 3.0.3 extends the testing interval required by codes and standards referenced by the Technical Specifications. This clarification is provided to remove any ambiguities relative to the frequencies for performing the required testing activities. Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the codes and standards referenced therein. The requirements of regulations take precedence over the TS. Therefore, test intervals governed by regulation cannot be extended by the TS.

Specification 3.0.4 establishes the requirement that Surveillances must be met during other specified conditions in the Specification for which the requirements of the Limiting Condition for Operation apply, unless otherwise specified in the individual Surveillances. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified surveillance interval, in accordance with Specifications 3.0.1 and 3.0.2, constitutes a failure to meet the OPERABILITY requirements for the corresponding Limiting Condition for Operation.

Systems and components are assumed to be OPERABLE when the associated Surveillances have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when either:

a. The systems or components are known to be inoperable, although still meeting the Surveillances or

Basis (continued)

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a specified condition for which the requirements of the associated Limiting Condition for Operation are not applicable unless otherwise specified. The Surveillances associated with a special test exception (STE) are only applicable when the STE is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given Surveillance. In this case, the unplanned event may be credited as fulfilling the performance of the Surveillance. This allowance includes those Surveillances whose performance is normally precluded in a specified condition.

Surveillances, including Surveillances invoked by ACTIONS, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with Specifications 3.0.1 and 3.0.2, prior to returning equipment to OPERABLE status.

Specification 3.0.5 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified surveillance interval. A delay period of up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with Specification 3.0.1, and not at the time that the specified surveillance interval was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with ACTIONS or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a surveillance interval based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., as modified by approved exemptions, etc.) is discovered to not have been performed when specified, Specification 3.0.5 allows for the full delay period of up to the specified surveillance interval to perform the Surveillance.

However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity. Specification 3.0.5 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of changes imposed by ACTIONS.

Failure to comply with specified surveillance intervals for Surveillance Requirements is expected to be an infrequent occurrence. Use of the delay period established by Specification 3.0.5 is a flexibility which is not intended to be used as a convenience to extend surveillance intervals. While up to 24 hours or the limit of the specified surveillance interval is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required to perform the Surveillance) and impact on any analysis assumptions, in addition to station conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management actions. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the corrective action program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the allowable outage time limits of the ACTIONS for the applicable Limiting Condition for Operation begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the allowable outage time limits of the ACTIONS for the applicable Limiting Condition for Operation begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the allowable outage time limits of the ACTIONS, restores compliance with Specification 3.0.4.

3.0 **SURVEILLANCE REQUIREMENTS**

3.1 DELETED

3.2 Equipment and Sampling Tests

Applicability

Applies to plant equipment and conditions related to safe storage and handling of nuclear fuel.

Objective

To specify the minimum frequency and type of surveillance to be applied to critical plant equipment and conditions.

Specifications

Equipment and sampling tests shall be conducted as specified in Tables 3-4 and 3-5.

Basis

The equipment testing and system sampling frequencies specified in Tables 3-4 and 3-5 are considered adequate, based upon experience, to maintain the status of the equipment and systems so as to assure safe storage and handling of nuclear fuel. Thus, those systems where changes might occur relatively rapidly are sampled frequently and those static systems is not subject to changes are sampled less frequently.

TABLE 3-4

MINIMUM FREQUENCIES FOR SAMPLING TESTS

		Type of Measurement and Analysis	Sample and Analysis Frequency
1.	Spent Fuel Pool	Boron Concentration	See Footnote (1) below

(1) Weekly when fuel assemblies are stored in the spent fuel pool.

TABLE 3-5

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

		Test	Frequency
1.	Spent Fuel Pool Racks	Test neutron poison samples for dimensional change, weight, neutron attenuation change and specific gravity change.	1, 2, 4, 7, and 10 years after installation, and every 5 years thereafter.
2.	Spent Fuel Pool Level	Verify spent fuel pool water level is ≥ 23 ft. above the top of irradiated fuel assemblies seated in the storage racks.	Prior to commencing, and weekly during FUEL HANDLING OPERATIONS in the spent fuel pool.
3.	Spent Fuel Assembly Storage	Verify by administrative means that initial enrichment and burnup of the fuel assembly is in accordance with Figure 2-10.	Prior to storing the fuel assembly in Region 2 (including peripheral cells).

3.0 SURVEILLANCE REQUIREMENTS

Sections 3.3 through 3.17 have been deleted in their entirety.

4.0 **DESIGN FEATURES**

4.1 Site

The site for Fort Calhoun Station Unit No. 1 is in Washington County, Nebraska, on the west bank of the Missouri River and approximately nineteen miles north, northwest of the city of Omaha, Nebraska. The exclusion area description, as defined in 10 CFR Part 100, Section 100.3(a), is located in the Final Safety Analysis Report, as updated.

4.2 DELETED

4.3 Fuel Storage

4.3.1 Criticality

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

a. Fuel assemblies having a maximum U-235 enrichment of 4.5 weight percent,

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.5 of the FSAR, as updated,

c. A nominal 8.6 inch center to center distance between fuel assemblies placed in Region 2, the high density fuel storage racks,

d. A nominal 9.8 inches (East-West) by 10.3 inches (North South) center to center distances between fuel assemblies placed in Region 1, the low density fuel storage racks,

e. Partially spent fuel assemblies with a discharge burnup in the "acceptable domain" of Figure 2-10 for "Region 2 Unrestricted" may be allowed unrestricted storage in any of the Region 2 fuel storage racks in compliance with Reference (1).

f. Partially spent fuel assemblies with a discharge burnup between the "acceptable domain" and "Peripheral Cells" of Figure 2-10 may be allowed unrestricted storage in the peripheral cells of the Region 2 fuel storage racks in compliance with Reference (1).

g. Partially spent fuel assemblies with a discharge burnup in the "unacceptable domain" of Figure 2-10 will be stored in Region 1 in compliance with Reference (1).

4.0 **DESIGN FEATURES**
4.3 Fuel Storage
4.3.1 Criticality (continued)

4.3.1.2 DELETED

4.3.1.3 DELETED

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below 23 ft. above the top of irradiated fuel assemblies seated in the 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 1083 fuel assemblies.

References:

(1) Letter from R. Wharton (NRC) to T. Patterson (OPPD), Amendment 174 to Facility Operating License No. DPR-40, (TAC NO. M94789) Dated July 30, 1996, NRC-96-0126.

5.0 **ADMINISTRATIVE CONTROLS**

5.1 Responsibility

5.1.1 The plant manager shall be responsible for the overall facility, and maintenance of fuel, shall delegate in writing the succession to this responsibility during his absence.

5.1.2 The Shift Manager shall be responsible for the shift command function.

5.2 Organization

5.2.1 Onsite and offsite organizations shall be established for the facility and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear fuel.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all organization positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR, as updated.

b. The plant manager shall be responsible for the overall facility and shall have control over those onsite activities necessary for safe storage and maintenance of the nuclear fuel.

c. The corporate officer with responsibility for overall management of nuclear fuel shall take any measures needed to ensure acceptable performance of the staff in controlling, maintaining, and providing technical support to the station to ensure safe management of nuclear fuel.

d. The individuals who train the CERTIFIED FUEL HANDLERs and NON-CERTIFIED OPERATORS, and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their ability to perform their assigned functions.

5.2.2 Facility Staff

The facility staff organization shall be as described in the FSAR, as updated and shall function as follows:

a. The minimum number and type of personnel required onsite for each shift shall be as shown in Table 5.2-1.

5.0 **ADMINISTRATIVE CONTROLS**
5.2 Organization
5.2.2 Facility Staff (continued)

b. An individual qualified in Radiation Protection Procedures shall be onsite during fuel handling operations or movement over storage racks containing fuel. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

c. The Shift Manager shall be a CERTIFIED FUEL HANDLER.

d. Fire protection program responsibilities are assigned to those positions and/or groups designated by asterisks in the FSAR, as updated.

e. DELETED

TABLE 5.2-1

MINIMUM SHIFT CREW COMPOSITION (ii, iii)

Staffing Category	Minimum Staffing
CERTIFIED FUEL HANDLER	1(i)
NON-CERTIFIED OPERATOR	1(iv)

(i) This includes the individual with CERTIFIED FUEL HANDLER qualification supervising FUEL HANDLING OPERATIONS.

(ii) Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 5.2-1 provided no fuel handling operations or movement over storage racks containing fuel is in progress. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crew member being late or absent.

(iii) At least one of these individuals must be in the control room at all times when fuel is in the Spent Fuel Pool.

(iv) The NON-CERTIFIED OPERATOR position may be filled by a CERTIFIED FUEL HANDLER.

5.0 **ADMINISTRATIVE CONTROLS**

5.3 Facility Staff Qualification

5.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions with the requirement exception for those individuals in lieu of holding a Senior Reactor Operator license will be qualified as a CERTIFIED FUEL HANDLER, and with the exception of the Manager - Radiation Protection (MRP), who shall meet the requirements set forth in Regulatory Guide 1.8, Revision 3, dated May 2000, entitled "Qualification and Training of Personnel for Nuclear Power Plants."

5.4 Training

5.4.1 A retraining and replacement training program for the plant staff shall be maintained under the direction of the Plant Manager or designee and shall meet or exceed the requirements of Section 6 of ANSI/ANS 3.1-1993, as modified by Regulatory Guide 1.8, Revision 3, dated May 2000.

5.4.2 An NRC approved training and retraining program for the CERTIFIED FUEL HANDLER shall be maintained.

5.5 Not Used

5.6 Not Used

5.7 Not Used

5.8 Procedures

5.8.1 Written procedures and administrative policies shall be established, implemented and maintained covering the following activities:

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

b. DELETED

c. Not Used

d. All applicable programs specified in Specifications 5.11, 5.16, and 5.20.

5.8 Procedures (continued)

5.8.2 Temporary changes to procedures of 5.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the facility supervisory staff, at least one of whom is qualified as a CERTIFIED FUEL HANDLER.

c. The change is documented, reviewed by a qualified reviewer and approved by either the plant manager or the department head designated by Administrative Controls as the responsible department head for that procedure within 14 days of implementation.

5.8.3 Written procedures shall be implemented which govern the selection of fuel assemblies to be placed in Region 2 of the spent fuel racks (Technical Specification 2.8). These procedures shall require an independent verification of initial enrichment requirements and fuel burnup calculations for a fuel bundle to assure the "acceptance" criteria for placement in Region 2 are met. This independent verification shall be performed by individuals or groups other than those who performed the initial acceptance criteria assessment, but who may be from the same organization.

5.9 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the appropriate NRC Regional Office unless otherwise noted.

5.9.1 Not Used

5.9.2 Not Used

5.9.3 DELETED

5.9.4 Unique Reporting Requirements

a. Annual Radioactive Effluent Release Report

The Annual Radioactive Effluent Release Report covering the station during the previous calendar year shall be submitted before May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be 1) consistent with the objectives outlined in the ODCM and PCP, and 2) in conformance with 10 CFR 50.36a. and Section IV.B.1 of Appendix I to 10 CFR 50.

b. Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the station during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis 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 (1) the ODCM and (2) Section IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR 50.

c. Not Used

5.9.5 DELETED

5.9.6 DELETED

5.10 Record Retention

5.10.1 Records shall be retained as described in the Quality Assurance Program.

5.11 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

5.11.1 In lieu of the "control device" required by paragraph 20.1601(a) of 10 CFR Part 20, and as an alternative method allowed under § 20.1601(c), each high radiation area (as defined in § 20.1601) in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by required issuance of a Radiation Work Permit.* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

*Radiation Protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Manager-Radiation Protection (MRP) in the Radiation Work Permit.

5.11.2 The requirements of 5.11.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr** but less than 500 rads/hr*** (Locked High Radiation Area). In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or the MRP (or designee) with the following exception:

a. In lieu of the above, for accessible localized Locked High Radiation Areas located in large areas such as containment, where no lockable enclosure exists in the immediate vicinity to control access to the Locked High Radiation Area and no such enclosure can be readily constructed, then the Locked High Radiation Area shall be:

i. roped off such that an individual at the rope boundary is exposed to 1000 mrem/hr or less,
ii conspicuously posted, and
iii a flashing light shall be activated as a warning device.

**At 30 centimeters (12 inches) from the radiation source or from any surface penetrated by the radiation.
***At 1 meter from the radiation source or from any surface penetrated by the radiation.

5.12 DELETED

5.13 DELETED

5.14 DELETED

5.15 DELETED

5.0 **ADMINISTRATIVE CONTROLS**

5.16 Radiological Effluents and Environmental Monitoring Programs

The following programs shall be established, implemented, and maintained.

5.16.1 Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for control of radioactive effluents and for maintaining the doses to individuals in UNRESTRICTED AREAs from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functionality of radioactive liquid and gaseous radiation monitoring instrumentation including functionality tests and setpoint determination in accordance with the methodology in the ODCM.

b. Limitations on the concentration of radioactive material, other than dissolved or entrained noble gases, released in liquid effluents to unrestricted areas conforming to ten times 10 CFR 20.1001-20.2401, Appendix B, Table 2, Column 2. For dissolved or entrained noble gases, the concentration shall be limited to 2.0 E-04 μCi/ml total activity.

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM.

d. Limitations on the annual and quarterly doses or dose commitment to individuals in unrestricted areas from radioactive materials in liquid effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

e. Determination of cumulative doses from radioactive effluents for the current calendar quarter and current calendar year in accordance with the ODCM on a quarterly basis.

f. Limitations on the functionality and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity in plant effluents.

g. Limitations on the concentration resulting from radioactive material, other than noble gases, released in gaseous effluents to unrestricted areas conforming to ten times 10 CFR 20.1001-20.2401, Appendix B, Table 2, Column 1. For noble gases. the concentration shall be limited to five times 10 CFR 20.1001-20.2401, Appendix B, Table 2, Column 1.

5.0 **ADMINISTRATIVE CONTROLS**
5.16 Radiological Effluents and Environmental Monitoring Programs
5.16.1 Radioactive Effluent Controls Program (continued)

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

i. Limitations on the annual and quarterly doses to an individual beyond the site boundary from tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50.

j. Limitations on the annual dose or dose commitment to an individual beyond the site boundary due to releases or radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

5.16.2 Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

b. A Land Use Census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required by the results of this census.

c. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

5.0 **ADMINISTRATIVE CONTROLS**

5.17 Offsite Dose Calculation Manual (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by the Quality Assurance Program. This documentation shall contain:

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

   2. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the Plant Operations Review Committee and the approval of the plant manager.

c. Temporary changes to the ODCM may be made in accordance with Technical Specification 5.8.2.

d. Shall be submitted to the Nuclear Regulatory Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed and shall indicate the date (e.g., month/year) the change was implemented.

5.18 DELETED

5.19 DELETED

5.0 ADMINISTRATIVE CONTROLS

5.20 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

   2. A change to the FSAR, as updated 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 FSAR, as updated.

d. Proposed changes that meet the criteria of 5.20.b 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).

5.21 DELETED

5.22 DELETED

5.23 DELETED

5.24 DELETED

6.0 DELETED

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 297 TO RENEWED FACILITY

LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By letter dated June 24, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16176A213), Omaha Public Power District (OPPD, the licensee), submitted a certification to the U.S. Nuclear Regulatory Commission (NRC) indicating it would permanently cease power operations at Fort Calhoun Station, Unit No. 1 (FCS) by December 31, 2016.

By letter dated November 13, 2016 (ADAMS Accession No. ML16319A254), OPPD informed the NRC that in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.82(a)(1)(ii), as of November 13, 2016, all fuel has been permanently removed from the FCS reactor vessel and placed into the FCS spent fuel pool (SFP). Further, the letter stated that OPPD understands and acknowledges that upon docketing these certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel, the 10 CFR Part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel.

By letter dated March 31, 2017 (ADAMS Accession No. ML17093A309), as supplemented by letter dated September 26, 2017 (ADAMS Accession No. ML17269A343), OPPD requested an amendment to revise the FCS renewed facility operating license and the associated technical specifications (TSs) to permanently defueled TSs consistent with the permanent cessation of reactor operation and permanent defueling of the reactor.

The supplemental letter dated September 26, 2017, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 6, 2017 (82 FR 26135).

The existing FCS 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 function related to safe storage and management of irradiated fuel at an operating plant is similar to the corresponding function at a

permanently defueled facility, the 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. Since the facility's 10 CFR Part 50 license no longer authorizes emplacement or retention of fuel in the reactor vessel, the LCOs (and associated surveillance requirements (SRs)) that do not apply in a defueled condition are being proposed for deletion. The proposed amendment revises the operating license and associated TSs to reflect the permanent cessation of reactor operations and the permanently defueled condition of the reactor vessel at FCS. In general, the changes eliminate those TSs applicable in operating MODES; MODES where fuel is emplaced in the reactor vessel, and certain TSs required for movement of irradiated fuel assemblies. Changes were also proposed to TS definitions, administrative controls, and TS related to programs and procedures. The proposed amendment also revises the facility operating license to clarify or remove certain conditions no longer relevant and add conditions consistent with other permanently shutdown and defueled reactors.

Amendment No. 292 for FCS was issued by the NRC on July 28, 2017 (ADAMS Accession No. ML17165A465), to revise certain requirements in the permanently shutdown and defueled facility's TSs, Section 5.0, "Administrative Controls," related to responsibilities, organization, and facility staff qualifications that reflect new staffing and training requirements for operating staff. Issuance of this amendment, in conjunction with the previously issued TS administrative controls amendment, completes the revision to the FCS permanently shutdown and defueled TSs.

2.0 REGULATORY EVALUATION

2.1 Technical Specifications

Section 182a of the Atomic Energy Act 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 focused on instrumentation to detect degradation of the reactor coolant system (RCS) pressure boundary, process variables and equipment, design features, or operating restrictions that affect the integrity of fission product barriers during design-basis accidents (DBAs) or transients. A fourth criterion refers to the use of operating experience and probabilistic risk assessment to identify and include in the TSs those structures, systems, and components (SSCs) shown to be significant to public health and safety.

A general discussion of the criteria that were used by the NRC staff in its evaluation to ensure that 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 "[i]nstalled instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary." Since no fuel is present in the reactor or RCS at the FCS facility, 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 design basis accident 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 reactor permanently shutdown and defueled is reduced from those postulated for an operating reactor, and most TSs satisfying Criterion 2 are no longer applicable. One existing TS that defines the initial condition of the DBA associated with irradiated fuel movement is discussed in Section 3.5 of this safety evaluation.

Criterion 3 of 10 CFR 50.36(c)(2)(ii)(C) states that TS LCOs for operation 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 apply to FCS is discussed in more detail in Section 3.0 of this safety evaluation.

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 that apply to the permanently shutdown and defueled condition at FCS that can result in a significant offsite radiological risk to public health and safety.

The NRC staff notes that in the course of this evaluation, information contained in NUREG-1625, "Proposed Standard Technical Specifications for Permanently Defueled Westinghouse Plants," Draft Report for Comment, March 1998 (ADAMS Accession No. ML082330233), 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.

2.2 Radiological Consequences from Design-Basis Accidents

During normal power reactor operations, the forced flow of water through the RCS removes the heat generated by the reactor. The RCS, operating at high temperatures and pressures, transfers this heat through the steam generator (SG) tubes to the secondary system. 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 and subsequent release of some fission products to the environment. Many of the accident scenarios postulated in the facility's safety analysis report involve failures or malfunctions of systems that could affect the reactor core. With the termination of reactor operations and the permanent removal of the fuel from the reactor core, such accidents are no longer possible. Therefore, the postulated

accidents involving failure or malfunction of the reactor, RCS, or secondary system are no longer applicable. Postulated accidents that could potentially apply to a permanently shutdown and defueled facility include a fuel handling accident (FHA) in the Auxiliary Building, where the SFP is located, a spent fuel cask drop accident, a gas decay tank rupture, and a waste liquid incident. The potential offsite consequences of these events are affected by the time available for decay of fission products in the fuel and, possibly, the availability of engineered safety features, such as ventilation systems to filter fission products from the accident area atmosphere before they are released outside the facility.

The regulations in 10 CFR 50.67, "Accident source term," state, in part, that the NRC may issue the amendment only if the applicant'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 [Sievert (Sv)] (25 [roentgen equivalent man (rem)])[2] 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)," Criterion 19, "Control room," states, in part, that:

> 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.

Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, July 2000 (ADAMS Accession No. ML003716792), provides the methodology for analyzing the radiological consequences of several DBAs to show compliance with 10 CFR 50.67, "Accident source term." Provided in

RG 1.183 is guidance to licensees on acceptable application of alternate source term (AST) submittals, including acceptable radiological analysis assumptions for use in conjunction with the AST.

Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000 (ADAMS Accession No. ML003734190), of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: [Light-Water Reactor (LWR)] Edition" (SRP), provides review guidance to the NRC staff for the review of AST amendment requests. The guidance in SRP Section 15.0.1 states that the NRC reviewer should evaluate the proposed change against the guidance in RG 1.183. As provided in RG 1.183, the dose acceptance criteria for an FHA are a TEDE of 6.3 rem at the exclusion area boundary for the worst 2 hours, 6.3 rem at the outer boundary of the low population zone, and 5 rem in the control room for the duration of the accident.

The emergency planning requirements of 10 CFR 50.47, "Emergency Plans," and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50 continue to apply to a nuclear power reactor after permanent cessation of operation 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. To modify their emergency plans to reflect the risk commensurate with power reactors that have been permanently shut down, power reactor licensees transitioning to decommissioning must seek exemptions from certain emergency planning regulatory requirements before amending these plans. The regulations in 10 CFR 50.12(a)(2)(ii) provide that the NRC may, on application by a licensee or on its own initiative, grant exemptions from the requirements of the regulations in circumstances in which application of the regulation would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. In addition to showing the existence of special circumstances in support of the exemption request, 10 CFR 50.12(a)(1) requires that the exemption must be authorized by law, not present an undue risk to the public health and safety, and be consistent with the common defense and security. 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.

License Amendment No. 201 for FCS, dated December 5, 2001 (ADAMS Accession No. ML013030027), incorporated the AST methodology for analyzing the radiological consequences of DBAs using RG 1.183. License Amendment No. 201 represents a full-scope implementation of the AST as described in RG 1.183.

Nuclear Energy Institute (NEI) topical report NEI 99-01, "Development of Emergency Action Levels for Non-Passive Reactors" Revision 6, November 2012 (ADAMS Accession No. ML12326A805), provides guidance for development of emergency action levels for reactors in a permanently defueled condition. The NRC endorsed NEI 99-01 by letter dated March 28, 2013 (ADAMS Accession No. ML12346A463). The NEI 99-01 topical report states that the accident analysis necessary to adopt the permanently defueled emergency action level 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 1 to

5 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.

3.0 TECHNICAL EVALUATION

3.1 Accident Analysis

Chapter 14, "Safety Analysis," of the FCS Updated Safety Analysis Report (USAR), describes the DBA scenarios that are applicable to FCS during power operations and the accidents with the greatest potential for radiation exposure. 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. When the reactor is permanently defueled and irradiated fuel assemblies are stored in the SFP and the independent spent fuel storage installation, the spectrum of creditable accidents is much smaller than for an operational plant, and most of the accident scenarios postulated in the USAR are no longer possible.

The licensee has stated, and the NRC staff agrees, that while spent fuel remains in the SFP, the only postulated DBA that will remain applicable to the permanently defueled FCS that could contribute a significant dose is the FHA in the Auxiliary Building, where the SFP is located. For completeness, the NRC staff also evaluated the applicability of other DBAs documented in the FCS USAR to ensure that these accidents would not have consequences that could potentially trigger the EPA PAGs. These additional evaluations include a spent fuel cask drop accident, a gas decay tank rupture, and a waste liquid incident.

In addition, the licensee considered a beyond design basis event scenario to evaluate the effects of a loss of water inventory from the FCS SFP. The purpose of evaluating this beyond design basis event is to determine the dose consequences due to loss of water shielding for an event in which the spent fuel assemblies are uncovered following an SFP drain down.

3.2 Fuel Handling Accident in the Auxiliary Building

The FCS FHA is described in USAR Section 14.18, "Fuel Handling Accident (in Spent Fuel Pool and Containment)." The USAR analysis applicable during normal power operations assumed a minimum decay period of 72 hours prior to fuel movement. This analysis determined that the exclusion area boundary, low population zone, and control room doses from an FHA would be 1.5, 0.5, and 0.5 rem TEDE, respectively. The licensee has determined that within 10 days after shutdown, the FHA doses would decrease to a level that would not warrant protective actions under the EPA PAG framework.

The NRC staff notes that the doses from an FHA are dominated by the isotope Iodine 131 (^{131}I) which has an 8-day half-life. The licensee has based its application for revision to the emergency plan and emergency action level scheme on an effective implementation date of April 7, 2018. The date of cessation of power operations occurred on October 24, 2016. Therefore, by the date of implementation of the revised emergency plan and emergency action

[1] For the purpose of this safety evaluation, the terms TED and TEDE are interchangeable, both describing the combined effects of internal and external exposure.

level scheme, the fuel will have decayed for 530 days. With 530 days of decay, the thyroid dose from an FHA would be negligible. After 530 days of decay, the only isotope remaining in significant amounts among those postulated to be released in an FHA, would be Krypton 85 (^{85}Kr), which has a half-life of 10.76 years. Since ^{85}Kr primarily decays by beta emission, the radiological concern from a release of ^{85}Kr would be the dose to the skin. The NRC staff notes that the calculated ^{85}Kr skin dose from an FHA using RG 1.183 assumptions is quite low (on the order of 10s of mrem as compared to the EPA PAG standard of 1 rem (1,000 mrem) at the exclusion area boundary). Therefore, the calculated skin dose from an FHA analysis would make an insignificant contribution to the TEDE, which is the parameter of interest in the determination of the EPA PAGs for sheltering or evacuation.

In performing this review, the NRC staff relied upon information provided by the licensee and NRC staff's experience in performing similar reviews. The NRC staff concludes that the dose consequence from an FHA for the permanently defueled FCS would not approach the EPA PAGs and would not trigger the declaration of an SAE.

3.3 Spent Fuel Cask Drop Accident

Section 14.18 of the FCS USAR discusses the potential for a spent fuel cask drop accident in the SFP located in the Auxiliary Building and states the following:

> The likelihood of dropping a spent fuel cask into the spent fuel pool is extremely low. The auxiliary building crane is licensed as Single-Failure-Proof and is equipped with overload alarms and safety devices. The safety features incorporated into the design of the main hoisting system of the crane preclude a cask drop accident by preventing a load drop in the event of a single failure in the hoisting or braking systems. Interlocks normally prevent the trolley from traversing any part of the spent fuel pool, which precludes the possibility of a load drop on spent fuel.

The NRC staff concludes that due to the fact that the FCS Auxiliary Building crane is licensed as being Single-Failure-Proof, a spent fuel cask drop accident is not considered a credible accident for the permanently defueled FCS.

3.4 Gas Decay Tank Rupture

Section 14.19, "Gas Tank Rupture," of the FCS USAR describes the gas decay tank rupture accident as the uncontrolled or unanticipated release of the radioactive noble gases stored in a waste gas decay tank (WGDT) as a result of a failure of a tank or associated piping. The WGDT accident analysis employs conservative assumptions to maximize the dose consequence of a WGDT rupture for full power operations by assuming that all the RCS noble gas inventories are available for release with no credit taken for radiological decay during the transfer of RCS activity to the decay tank. The dose consequence documented in the USAR analysis is a small fraction of the EPA PAGs.

The USAR dose consequence is dominated by the release of Xenon 133 (^{133}Xe), which has a half-life of 5.245 days. After 530 days of decay, the only isotope remaining in significant amounts among those postulated to be released from a WGDT rupture, would be ^{85}Kr, which has a half-life of 10.76 years. As was the case for an FHA, the resulting skin dose from the release of ^{85}Kr would make an insignificant contribution to the TEDE, which is the parameter of interest in the determination of EPA PAGs for sheltering or evacuation. Therefore, the NRC

staff concludes that the dose consequence from a WGDT rupture for the permanently defueled FCS will not approach the EPA PAGs for sheltering or evacuation and would not trigger the declaration of an SAE.

3.5 Waste Liquid Incident

Section 14.20, "Waste Liquid Incident," of the FCS USAR explains that a waste liquid incident is considered to be any incident that results in the release of waste liquid, and its accompanying activity, to the environment. The FCS radioactive waste disposal system is designed such that any spillage or leakage of radioactive liquid waste would be retained within the facility. As described in the FCS USAR, administrative controls, multiple valving, fail-safe features, and reliable instrumentation and controls provide assurance against the release of radioactive liquid waste to the environment in excess of 10 CFR Part 20 limits.

As explained in the FCS USAR, the radioactive waste systems are designed to retain the release of liquids within the facility. Section 14.20 also includes a dose consequence analysis, which considers the gaseous release of entrained activity as a result of a liquid waste tank failure (LWTF). The analysis incorporates conservative assumptions regarding the amount of activity assumed to be contained in the liquid as well as the amount of activity which is assumed to become airborne as a result of the LWTF. The dose consequence from the gaseous release associated with an LWTF are shown to be well below 10 CFR Part 20 limits.

The NRC staff notes that the USAR LWTF dose consequence is dominated by the release of ^{133}Xe, which has a half-life of 5.245 days. After 530 days of decay, the only isotope remaining in significant amounts among those postulated to be released from the gaseous release associated with an LWTF, would be ^{85}Kr, which has a half-life of 10.76 years. As was the case for an FHA, the resulting skin dose from the release of ^{85}Kr would make an insignificant contribution to the TEDE, which is the parameter of interest in the determination of EPA PAGs for sheltering or evacuation. Therefore, the NRC staff concludes that the dose consequence from an LWTF for the permanently defueled FCS will not approach the EPA PAGs for sheltering or evacuation and would not trigger the declaration of an SAE.

3.6 Consequences of a Beyond Design Basis Event

The licensee analyzed the radiological consequences of a beyond design basis scenario to evaluate the effects of a loss of water inventory from the FCS SFP. The primary purpose of this calculation is to determine the dose rates as a function of time at the exclusion area boundary and in the control room due to loss of shielding for an event in which the spent fuel assemblies are uncovered following drain down. The dose rates determined by this calculation are due to direct and indirect radiation from spent fuel assemblies. The NRC staff notes that while the direct dose rate above the unshielded fuel would be high, radiation protection personnel would restrict access to ensure that no one was subjected to the direct dose from the unshielded fuel. Therefore, the primary concern becomes the dose rate from gamma and neutron radiation that is scattered from interactions with the air above the SFP. The radiation that is scattered due to interactions with air is sometimes referred to as sky shine. The licensee used appropriate methods to evaluate the effects of this source of radiation at the exclusion area boundary and in the control room.

The licensee determined that for this beyond design basis event, the integrated external dose at the exclusion area boundary, assuming continuous exposure, would be well within the 10 CFR 20.1302(b)(ii), "Compliance with dose limits for individual members of the public," limit

of 50 mrem in a year for an individual continuously present in an unrestricted area. The NRC staff notes that the integration of this potential radiation source for an entire year is extremely conservative in that the licensee did not credit any actions that could be taken to reduce the source such as refilling the SFP with water. In addition, the licensee determined that the control room dose rate from this beyond design basis event would be insignificant.

The NRC staff reviewed the license's evaluation and preformed independent analyses which confirmed the licensee's results.

3.7 Accident Analysis Conclusions

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 and to the results of confirmatory analyses by the staff. The NRC staff finds with reasonable assurance that FCS, 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 defueled FCS are well below the EPA thresholds for the implementation of PAGs and would not trigger the declaration of an SAE. Therefore, the NRC staff finds the proposed changes to be acceptable from a dose consequence perspective.

3.8 Proposed TS Changes

3.8.1 "Definitions"

The licensee proposed deleting the following TS definitions because they pertain to an operating reactor. Since FCS is permanently shut down and defueled, the definitions have no relevance and no longer apply:

Rated Power

A steady state reactor core output of 1500 MWt.

Reactor Critical

The reactor is considered critical for purposes of administrative control when the neutron flux logarithmic range channel instrumentation indicates greater than 10^{-4}% of rated power.

Power Operation Condition (Operating Mode 1)

The reactor is in the power operation condition when it is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

Hot Standby Condition (Operating Mode 2)

The reactor is considered to be in a hot standby condition if the average temperature of the reactor coolant (T_{avg}) is greater than 515°F, the reactor is critical, and the neutron flux power range instrumentation indicates less than 2% of rated power.

Hot Shutdown Condition (Operating Mode 3)

The reactor is in a hot shutdown condition if the average temperature of the reactor coolant (T_{avg}) is greater than 515°F and the reactor is subcritical by at least the amount defined in Paragraph 2.10.2.

Cold Shutdown Condition (Operating Mode 4)

The reactor coolant T_{cold} is less than 210°F and the reactor coolant is ≥ SHUTDOWN BORON CONCENTRATION but < REFUELING BORON CONCENTRATION.

Refueling Shutdown Condition (Operating Mode 5)

The reactor coolant T_{cold} is less than 210°F and the reactor coolant is ≥ REFUELING-BORON CONCENTRATION.

The Refueling Boron Concentration

A reactor coolant boron concentration of at least that specified in the CORE OPERATING LIMITS REPORT which corresponds to a shutdown margin of not less than 5% with all CEA's withdrawn.

Shutdown Boron Concentration

The boron concentration required to make the reactor subcritical by the amount defined in Section 2.10.

Refueling Outage or Refueling Shutdown

A plant outage or shutdown to perform refueling operations upon reaching the planned fuel depletion for a specific core.

Plant Operating Cycle

The time period from a REFUELING SHUTDOWN to the next REFUELING SHUTDOWN.

Physics Testing

Testing performed under written procedures approved by Plant Operations Review Committee to determine CEA worths and other core nuclear parameters. Deviations from normal operating practice which are necessary to enable some

of these tests to be performed are permitted in accordance with: 1) the specific provisions of these technical specifications, 2) authorization under the provisions of 10 CFR 50.59, or 3) other approval of the Commission.

Reactor Trip

The de-energizing of the CEDM magnetic clutch holding coils which releases the CEA's and allows them to drop into the core.

Instrument Channel

One of four independent measurement channels complete with the sensors, sensor power supply units, amplifiers, and trip modules provided for each safety parameter.

Reactor Protective System Logic(1)

The system which utilizes relay contact outputs from individual instrument channels to provide the reactor trip signal for de-energizing the magnetic clutch power supplies. The logic system is wired to provide a reactor trip on a 2-of-4 or 2-of-3 basis for any given input parameter.

Reference (1) associated with this definition will also be deleted.

Engineered Safety Feature Logic(2)

The system which utilizes relay contact outputs from individual instrument channels to provide a dual channel signal to independently initiate the actuation of the engineered safety feature equipment. Two logic subsystems, termed A and B, are provided; each subsystem is composed of four channels wired to provide independent safety feature initiation signals on a 2-out-of-4 basis (Containment Radiation High Signal is 1-out-of-2 logic).

Reference (2) associated with this definition will also be deleted.

Degree of Redundancy

The difference between the number of operable channels and the number of channels which when tripped will cause an automatic system trip.

Channel Check

A qualitative determination of acceptable operability by observation of channel behavior during normal plant operation. This determination shall where feasible, include comparison of the channel with other independent channels measuring the same variable.

Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including any alarm and/or trip initiating action.

Channel Calibration

Adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment action, alarms, interlocks or trip, and shall be deemed to include the channel functional test.

In Operation

A system or component is IN OPERATION if it is OPERABLE and is performing its design function.

CEA's

All full length shutdown and regulating control rods.

Non-trippable (NT) CEA's

CEA's which are non-trippable.

Containment Integrity

Containment integrity is defined to exist when all of the following are met:

(1) All nonautomatic containment isolation valves which are not required to be open during accident conditions and blind flanges, except for valves that are open under administrative control as permitted by Specification 2.6(1)a, are closed.

(2) The equipment hatch is properly closed and sealed.

(3) The personnel air lock satisfies Specification 2.6(1)b.

(4) All automatic containment Isolation valves are operable, locked closed, or deactivated and secured in their closed position (or isolated by locked closed valves or blind flanges as permitted by a limiting condition for operation).

(5) The uncontrolled containment leakage satisfies Specification 3.5, and

(6) The sealing mechanism associated with each penetration (e.g., welds, bellows or O-rings) is operable.

Core Alteration

The movement or manipulation of fuel, sources, reactivity control components, or other components affecting reactivity within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE

ALTERATION shall not preclude completion of movement of a component to a safe, conservative position.

Equivalent Full Power Day (EFPD)

The time interval during power operation when the heat generated by the reactor is equivalent to reactor operation at 100% of rated power for 24 hours.

Shutdown Margin

Shutdown Margin shall be the amount of reactivity by which:

(1) the reactor is subcritical; or

(2) the instantaneous amount of reactivity by which the reactor would be subcritical from its present condition assuming:

   a. All known trippable full length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn, and

   b. No change in non-trippable control element assembly position.

Axial Shape Index

The external AXIAL SHAPE INDEX (Y_E) is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The internal AXIAL SHAPE INDEX (Y_I) used for the trip and pre-trip signals in the reactor protection system is the above value (Y_E) modified by the shape annealing factor, SAF, and a constant, B, to determine the true core axial power distribution for that channel.

$$Y_E = \frac{L-U}{L+U} \qquad Y_I = SAF \times Y_E + B$$

Azimuthal Power Tilt – Tq

Azimuthal Power Tilt shall be the power asymmetry between azimuthally symmetric fuel assemblies.

Maximum Radial Peaking Factor (F_R^T)

The Maximum Radial Peaking Factor is the maximum ratio of the individual fuel pin power to the core average pin power integrated over the total core height, including tilt. The F_R^T limit is provided in the Core Operating Limits Report.

Dose Equivalent I-131

That concentration of I-131 (μCi/gm) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. In other words,

$$
\begin{aligned}
\text{Dose Equivalent I-131 } (\mu\text{Ci/gm}) = \; & \mu\text{Ci/gm of I-131} \\
+ \; & 0.0361 \times \mu\text{Ci/gm of I-132} \\
+ \; & 0.270 \times \mu\text{Ci/gm of I-133} \\
+ \; & 0.0169 \times \mu\text{Ci/gm of I-134} \\
+ \; & 0.0838 \times \mu\text{Ci/gm of I-135}
\end{aligned}
$$

$\bar{E}$ - Average Disintegration Energy

$\bar{E}$ is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration, in MEV, for isotopes, other than iodines, with half lives greater than 15 minutes making up at least 95% of the total non-iodine radioactivity in the coolant.

Core Operating Limits Report (COLR)

The Core Operating Limits Report (COLR) is a Fort Calhoun Station Unit No. 1 specific document that provides core operating limits for the current operating cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Section 5.9.5. Plant operation within these operating limits is addressed in the individual specifications.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE
    1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal leakoff), that is captured and conducted to collection systems or a sump or collecting tank,
    2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE, or
    3. LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE),

b. Unidentified LEAKAGE
    All LEAKAGE (except RCP seal leakoff) that is not identified LEAKAGE, and

c. Pressure Boundary LEAKAGE
LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

RCS Pressure-Temperature Limits Report (PTLR)

The PTLR is a fluence dependent document that provides Limiting Conditions for Operation (LCO) in the form of pressure-temperature (P-T) limits to ensure prevention of brittle fracture. In addition, this document establishes power operated relief valve setpoints which provide low-temperature overpressure protection (LTOP) to assure the P-T limits are not exceeded during the most limiting LTOP event. The P-T limits and LTOP criteria in the PTLR are applicable through the effective full power years (EFPYs) specified in the PTLR. NRC approved methodologies are used as the bases for the information provided in the PTLR.

In its application, the licensee proposed the following changes to the TS definitions:

a. "REACTOR OPERATING CONDITIONS," section title will be revised to read "FUEL STORAGE AND HANDLING CONDITIONS."

b. Refueling Operation – The application states, "this definition will be revised to Fuel Handling Operations and will remove the phrase "outside of the reactor pressure vessel," as shown below:

Fuel Handling Operations

Any operation involving the shuffling, removal, or replacement of irradiated fuel ~~outside of the reactor pressure vessel~~. The suspension of any FUEL HANDLING ~~REFUELING~~ OPERATIONS shall not preclude completion of movement of a component to a safe, conservative position.

c. Actions - This definition will be added to clarify its use in determining requirements for surveillance testing as follows:

Actions

ACTIONS shall be that part of a specification that prescribes required actions to be taken under designated conditions within specified completion times.

The NRC staff examined the TS definitions proposed for deletion and concluded that all the terms listed above are only meaningful to a reactor authorized to operate or are not retained in the defueled TS. Since FCS would be permanently shut down and defueled, the staff finds the deletion of these definitions from TSs acceptable.

In addition, the licensee proposed revising certain current TS definitions and adding other definitions as stated above. The NRC staff finds these changes conform to the usage contained in the Administrative Controls section of the FCS permanently defueled TSs and are consistent with the definition in 10 CFR Part 50 and are, therefore, acceptable.

3.8.2 Section 1.0, "Safety Limits"

The licensee proposed deletion of Section 1.0, "Safety Limits," in its entirety. This section establishes safety limits which preclude violation of the fuel design criteria and RCS design pressure. Specification 1.1.1, "Reactor Core SLs," is used to maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the reactor coolant. Specification 1.1.2, "Reactor Coolant System Pressure SL," is used to maintain the integrity of the RCS and to prevent the release of significant amounts of fission product activity to the containment. This section also contains TS 1.2, "Safety Limit Violations," which directs actions to be taken if an SL specified in TS 1.1 is violated.

The NRC staff reviewed the specifications as well as the associated basis. There are three SLs in Section 1.0 related to reactor power, fuel temperature, and RCS pressure. Because FCS has permanently shut down, defueled, and submitted certifications under the provisions of 10 CFR 50.82(a)(2), placing fuel in the reactor vessel and resuming power operations are no longer authorized. In this condition, there will be no reactor power or peak fuel centerline temperature to be monitored and therefore no challenge to RCS integrity or release of fission products. Based on these findings, the NRC staff concludes the SLs no longer apply. Therefore, the NRC staff finds the deletion of TS Section 1.0 in its entirety is acceptable.

3.8.3 Section 2.0, "Limiting Conditions for Operations"

Section 2.0.1, "General Requirements," of the FCS TSs contains the general requirements applicable to all LCOs applies at all times unless otherwise stated in TSs.

Current TS Section 2.0.1 lists the requirements as follows:

<u>Applicability</u>

Applies to the operable status of all systems, subsystems, trains, components, or devices covered by the Limiting Conditions for Operation.

<u>Objective</u>

To specify corrective measures to be employed for system conditions not covered by or in excess of the Limiting Conditions for Operation.

<u>Specification</u>

(1) In the event a Limiting Condition for Operation and/or associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least HOT SHUTDOWN within 6 hours, in at least subcritical and < 300°F within the next 6 hours, and in at least COLD SHUTDOWN within the following 30 hours, unless corrective measures are completed that permit operation under the permissible action requirements for the specified time interval as measured from initial discovery or until the reactor is placed in an Operating Mode in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications.

(2) When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

Basis

(1) This specification delineates corrective measures to be taken for circumstances not directly provided for in the system specific specifications and whose occurrence would violate the intent of the specification. For example, Specification 2.3 requires each Low Pressure Safety Injection (LPSI) pump to be operable and provides explicit corrective measures to be followed if one pump is inoperable. Under the terms of Specification 2.0.1(1), if more than one LPSI pump is inoperable, the unit must be placed in at least HOT SHUTDOWN within 6 hours, in at least subcritical and < 300°F within the following 6 hours, and in at least COLD SHUTDOWN within the following 30 hours, unless at least one LPSI pump were restored to operability. It is assumed that the unit is brought to the required mode within the required times by promptly initiating and carrying out the appropriate measures required by the specification.

(2) LCO 2.0.1(2) establishes conditions under which systems are considered to remain capable of performing their intended safety function when associated snubbers are not capable of providing their associated support function(s). This LCO states that the supported system is not considered to be inoperable solely due to one or more snubbers not capable of performing their associated support function(s). This is appropriate because a limited length of time is allowed for maintenance, testing, or repair of one or more snubbers not capable of performing their associated support function(s) and appropriate compensatory measures are specified in the snubber requirements, which are located outside of the Technical Specifications (TS) under licensee control. The snubber requirements do not meet the criteria in 10 CFR 50.36(c)(2)(ii), and, as such, are appropriate for control by the licensee.

> If the allowed time expires and the snubber(s) are unable to perform their associated support function(s), the affected supported system's LCO(s) must be declared not met and the Conditions and Required Actions entered.
>
> LCO 2.0.1(2)a applies when one or more snubbers are not capable of providing their associated support function(s) to a single train or subsystem of a multiple train or subsystem supported system or to a single train or subsystem supported system. LCO 2.0.1(2)a allows 72 hours to restore the snubber(s) before declaring the supported system inoperable. The 72-hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function and due to the availability of the redundant train of the supported system.
>
> LCO 2.0.1(2)b applies when one or more snubbers are not capable of providing their associated support function(s) to more than one train or subsystem of a multiple train or subsystem supported system. LCO 2.0.1(2)b allows 12 hours to restore the snubber(s) before declaring the supported system inoperable. The 12-hour Completion Time is reasonable based on the low probability of a seismic event concurrent with an event that would require operation of the supported system occurring while the snubber(s) are not capable of performing their associated support function.
>
> LCO 2.0.1(2) requires that risk be assessed and managed. Industry and NRC guidance on the implementation of 10 CFR 50.65(a)(4) (the Maintenance Rule) does not address seismic risk. However, use of LCO 2.0.1(2) should be considered with respect to other plant maintenance activities, and integrated into the existing Maintenance Rule process to the extent possible so that maintenance on any unaffected train or subsystem is properly controlled, and emergent issues are properly addressed. The risk assessment need not be quantified, but may be a qualitative awareness of the vulnerability of systems and components when one or more snubbers are not able to perform their associated support function.

The application states, in part, that "Section 2.0.1, General Requirements, will be deleted in its entirety. The basis for this TS will also be deleted." The application provides the following justification for the deletion of Section 2.0.1 requirements:

> <u>Section 2.0.1</u>, General Requirements, establishes the actions that must be implemented when an LCO is not met. Additionally, this section establishes requirements for snubbers not able to perform their support function. None of the described mode requirements of Section 2.0.1 are applicable with the reactor vessel defueled. There are no snubbers required to be maintained with fuel removed from the reactor vessel. Since the FCS license no longer authorizes 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 proposed deletion of this section in its entirety is acceptable.

With the TS section deleted in its entirety, the applicable basis section and any correlation/reference in other TS sections to this section will also be removed.

The NRC staff has reviewed the proposed changes to the FCS TSs concerning LCO applicability and has determined that the changes are consistent with the transition to a permanently shutdown and defueled facility. When the licensee permanently defuels the facility, 10 CFR 50.82(a)(2) will prohibit the licensee from operating the plant or placing fuel in the reactor vessel. The proposed change to delete the references to "mode" (the TSs reference to HOT SHUTDOWN, COLD SHUTDOWN modes) is appropriate because the term refers to conditions of normal operation. Since the licensee will no longer be operating, reference to operating modes is no longer needed. Similar to modes, the operating cycle interval no longer applies because the licensee will no longer be operating. Therefore, the changes appropriately reflect the change in plant status and are acceptable.

Because 10 CFR 50.82(a)(2) prohibits operation of the plant or placing fuel in the reactor vessel, all systems associated with snubbers are no longer required to be operable and are proposed for deletion from TSs. As such, LCO 2.0.1 no longer applies to any systems remaining in TSs. Therefore, the NRC staff concludes that the proposed deletion of LCO 2.0.1 is acceptable.

The application states that the proposed changes also include a renumbering of pages and sections, where appropriate, to condense and reduce the number of pages in the TS without affecting the technical content. The TS table of contents is also revised accordingly.

The NRC staff has reviewed the proposed changes and has determined that they are of a clarifying nature and do not change any technical requirements; therefore, these changes are acceptable.

3.8.4 Section 2.1, "Reactor Coolant System"

The RCS TS contains the LCOs, Actions, and basis 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. This section contains the following LCOs:

- 2.1.1 - Operable Components
- 2.1.2 - Heatup and Cooldown Rate
- 2.1.3 - Reactor Coolant Radioactivity
- 2.1.4 - Reactor Coolant System Leakage Limits
- 2.1.5 - Maximum Reactor Coolant Oxygen and Halogens Concentrations
- 2.1.6 - Pressurizer and Main Steam Safety Valves
- 2.1.7 - Pressurizer Operability
- 2.1.8 - Reactor Coolant System Vents

The LCO 2.1.1, "Operable Components," specifies certain conditions of the RCS components that shall be in operation including reactor coolant loops and associated reactor coolant pumps. There are several specifications based on reactor operating mode (Power Operation, Hot Standby, etc.). The basis for this specification is to maintain departure from nucleate boiling

ratio above the minimum departure from nucleate boiling limit during all normal operations and anticipated transients in order to protect the fuel.

The LCO 2.1.2, "Heatup and Cooldown Rate," specifies limiting conditions of the RCS heatup and cooldown rates. The limits are associated with maintaining the vessel pressure and temperature including the limitation established with heatup and cooldown rates to prevent encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the reactor coolant pressure boundary.

The LCO 2.1.3, "Reactor Coolant Radioactivity," ensures that the reactor coolant radioactivity is maintained at a level commensurate with occupational and public safety. The limitations on the radioactivity of the reactor coolant ensure that the resulting doses at the site boundary will be within the limits of 10 CFR 50.67 for an SG tube rupture accident.

The LCO 2.1.4, "Reactor Coolant System Leakage Limits," specifies limiting conditions of the reactor coolant system leakage rates to assure safe reactor operation. The RCS leakage is indicative of material deterioration, possibly of the RCS pressure boundary, which can affect the probability of a design basis event.

The LCO 2.1.5, "Maximum Reactor Coolant Oxygen and Halogens Concentrations," specifies the maximum oxygen and halogens concentrations of the reactor coolant system for safe reactor operation. Maintaining the oxygen, chloride, and fluoride concentrations in the reactor coolant within the limits specified, protects the integrity of the RCS materials in contact with the coolant against potential stress corrosion.

The LCO 2.1.6, "Pressurizer and Main Steam Safety Valves," specifies minimum requirements pertaining to the pressurizer and main steam safety valves. The basis for this LCO is to provide adequate overpressure protection for the RCS and main steam system.

The LCO 2.1.7, "Pressurizer Operability," specifies the minimum requirements pertaining to the pressurizer water volume and availability of heaters for accident conditions. The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS.

The LCO 2.1.8, "Reactor Coolant System Vents," ensures capability of venting non-condensible gases from the RCS. The basis is to ensure a method and system is available to remove steam and/or non-condensible gases from the RCS, which may inhibit core cooling during natural circulation.

The NRC staff has reviewed the specifications in Section 2.1 as well as the associated basis. The staff has determined that these TSs are needed to provide the LCOs and SRs necessary to maintain functionality and integrity of the fuel and RCS pressure boundary for an operating reactor. These TSs contain requirements for various RCS parameters such as:

(1) thermal limitations for heatup and cooldown rates during plant operation in order to operate within the analyzed requirements for stress intensity and fatigue limits for the reactor vessel;

(2) pressurization, which established and maintained an equilibrium under saturated conditions for pressure control to prevent bulk boiling in the remainder of the RCS;

(3) coolant chemistry, which included limits on RCS activity to limit potential offsite doses due to postulated events and limits on RCS conductivity, chlorides, and pH to prevent stress-corrosion cracking;

(4) coolant leakage, which established primary system leakage limits to allow prompt identification and isolation of leaks before the integrity of the RCS pressure boundary was impaired;

(5) safety and relief valves, which specifies operability requirements for the safety and relief valves designed to prevent over-pressurization of, and damage to, the primary system boundary; and

(6) structural integrity, which addresses the inservice inspection requirements of the primary system boundary components.

All of these TSs are related to assuring the integrity of the RCS pressure boundary for an operating reactor. The RCS TSs are only significant for a reactor authorized to operate or retain irradiated fuel in the reactor vessel. However, because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel, the RCS is no longer functional or used in any capacity at FCS. In this condition, the NRC staff finds TS Section 2.1 is no longer applicable as the reactor vessel is permanently defueled and the reactor coolant pressure boundary will no longer be used as a fission product barrier.

The NRC staff has also reviewed the RCS TS proposed for deletion to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.0 of this safety evaluation. The staff notes that these TSs indicate modes for which the TS is applicable. Modes, as defined in the TSs, correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning with fuel in the reactor vessel. The reference to modes for a permanently shutdown and defueled reactor, such as FCS, has no meaning and is not relevant. Because OPPD has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and, therefore, FCS is no longer in a configuration or a condition under which the TS modes apply. Furthermore, because irradiated fuel has been permanently removed from the reactor pressure vessel, the RCS is no longer relevant as a fission product barrier. Therefore, based on the above, the NRC staff finds the deletion of TS Section 2.1 in its entirety is acceptable.

3.8.5 Section 2.2, "Chemical and Volume Control System"

The chemical and volume control system TS contain the LCOs, Actions, and basis that provide for appropriate control of process variables, design features, or operating restrictions needed to provide a flow path and fluid storage for reactivity control, RCS makeup, and shutdown margin. This section contains the following LCOs:

- 2.2.1 - Boric Acid Flow Paths – Shutdown
- 2.2.2 - Boric Acid Flow Paths – Operating
- 2.2.3 - Charging Pumps – Shutdown

- 2.2.4 - Charging Pumps – Operating
- 2.2.5 - Boric Acid Transfer Pumps – Shutdown
- 2.2.6 - Boric Acid Transfer Pumps – Operating
- 2.2.7 - Borated Water Source – Shutdown
- 2.2.8 - Borated Water Sources – Operating

The LCO Sections 2.2.1 through 2.2.8 are used to assure operability of equipment required to add negative reactivity. The chemical and volume control system is used in Modes 1, 2, 3, 4, and 5 to provide a flow path and fluid storage for reactivity control, RCS makeup, and shutdown margin. As previously discussed, the reference to modes for a permanently shutdown and defueled reactor, such as FCS, has no meaning and is not relevant. Because OPPD has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and, therefore, FCS is no longer in a configuration or a condition under which the TS modes apply. Therefore, based on the above, the NRC staff finds the deletion of TS Section 2.2 in its entirety is acceptable.

3.8.6 Section 2.3, "Emergency Core Cooling System"

The emergency core cooling system (ECCS) TS contain the LCOs, Actions, and basis that provide for appropriate control of process variables, design features, or operating restrictions needed to assure operability of equipment required to provide core cooling and negative reactivity to ensure the reactor core is protected after a postulated accident.

The NRC staff has reviewed the ECCS TS and has determined that this TS is only needed to provide the LCOs and SRs necessary to maintain functionality of the systems that provide emergency cooling to the reactor core. The TS includes multiple LCOs addressing the safety injection refueling water tank, safety injection tanks, low pressure safety injection train, high pressure safety injection train, shutdown heat exchangers, and all associated piping and valves designed to provide adequate emergency cooling capability to the reactor during accident conditions. There are also specifications related to protection against low-temperature overpressure protection (LTOP) and containment sump buffering agent specification and volume requirements. The LTOP specification establishes restrictions on high-pressure safety injection pump operability at low temperatures, such that in combination with relief valve setpoints, the reactor vessel pressure-temperature limits would not be exceeded in the case of an inadvertent actuation of the operable high-pressure safety injection pump and charging pumps. The containment sump buffering agent specification establishes the required volume of hydrated sodium tetraborate during operating modes 1 and 2 which is required to adjust the pH of the recirculation water. A specific pH value is necessary to prevent significant amounts of iodine, released from fuel failures and dissolved in the recirculation water, from converting to a volatile form and evolving into the containment atmosphere.

The ECCS TS is related to providing cooling for a reactor vessel core. Since FCS is permanently shut down and defueled, there are no accidents of any kind that would require emergency core cooling and the accidents these systems and components were designed to mitigate are no longer possible.

The NRC staff reviewed the ECCS TS proposed for deletion to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in TSs, as described in Section 2.0 of this safety evaluation. Because OPPD has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and FCS is no

longer in a configuration or a condition under which the TS modes apply. Therefore, based on the above, the NRC staff finds the deletion of TS Section 2.3 in its entirety is acceptable.

3.8.7 Section 2.4, "Containment Cooling"

The containment cooling TS contain the LCOs, Actions, and basis that provide for appropriate control of process variables, design features, or operating restrictions needed to assure operability of equipment required to remove heat from the containment during normal operating and emergency situations.

The NRC staff has reviewed the containment cooling TS and has determined that this TS is only needed to provide the LCOs and SRs necessary to maintain functionality of the containment during Modes 1 and 2. This TS includes multiple LCOs addressing containment integrity including the containment spray and containment ventilation system which limits post-accident pressure and temperature in containment and provides iodine removal capability through use of the system's ventilation filters. These systems limit post-accident pressure and temperature in containment to less than the design values during accident conditions. However, 10 CFR 50.82(a)(2) prohibits the licensee from operating the plant or placing fuel in the reactor vessel. Consequently, the operational conditions, transients, and accidents the containment SSCs were designed to contain are no longer possible.

The NRC staff has also reviewed the containment systems TSs proposed for deletion to ensure that these LCOs no longer satisfy the 10 CFR 50.36 criteria for inclusion in the TSs, as described in Section 2.0 of this safety evaluation. As previously discussed, the staff notes that these TSs indicate modes for which the TS is applicable. The reference to modes for a permanently shutdown and defueled reactor, such as FCS, has no meaning and is not relevant. Because OPPD has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and FCS is no longer in a configuration or a condition under which the TS modes apply. Therefore, deletion of TS Section 2.4 in its entirety is acceptable.

3.8.8 Section 2.5, "Steam and Feedwater System"

The steam and feedwater system TS is applicable when the SGs are relied upon for heat removal from the RCS. With the reactor permanently defueled, the SGs and auxiliary feedwater system are not relied on for decay heat removal. Thus, the SGs and the auxiliary feedwater system are neither assumed to operate as an initial condition of a DBA analysis, nor are they credited as part of the primary success path that functions to mitigate a DBA that assumes the failure of the integrity of a fission product barrier. Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied, and deletion of TS Section 2.5 in its entirety is acceptable.

3.8.9 Section 2.6, "Containment System"

The containment system TS is applicable when the RCS is heated above 210 degrees Fahrenheit (°F) with fuel in the reactor vessel (i.e., the reactor is not in the cold shutdown or refueling mode of operation). With the reactor permanently defueled, containment integrity is not required to retain radioactive material because the RCS has no source of energy to drive a radioactive material release. The containment system is not credited as part of the primary success path that functions to mitigate a DBA that assumes the failure of the integrity of a

fission product barrier. Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied, and deletion of TS Section 2.6 in its entirety is acceptable.

3.8.10 Section 2.7, "Electrical Systems"

The electrical systems TS is applicable when the RCS is heated above 300 °F and specifies the requirements for direct current (DC) power to ensure that the DC electrical power subsystems are operable. With the reactor permanently defueled, the emergency diesel generators and their associated support systems (i.e., fuel oil, lubricating oil, and starting air) are not relied on to provide electrical power to support operation of emergency core cooling or essential auxiliary systems to protect fission product barriers associated with fuel in the reactor vessel. The fuel located in the SFP is adequately protected by the large inventory of water contained in the SFP without reliance on emergency power. Thus, the emergency diesel generators and DC electrical power subsystems and their associated support systems are neither assumed to operate as an initial condition of a DBA analysis, nor are they credited as part of the primary success path that functions to mitigate a DBA that assumes the failure of the integrity of a fission product barrier. Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied, and deletion of the TS Section 2.7 in its entirety is acceptable.

3.8.11 Section 2.8, "Refueling"

The current title of Section 2.8 is "Refueling." The licensee is proposing to delete or revise the term "Refueling" to "Fuel Handling," for all of its subsections and bases so that it reflects that refueling of the reactor will no longer be possible.

The NRC staff reviewed this proposed change and concludes that because 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactor or placing fuel in the reactor vessel at FCS, refueling is no longer allowed and that the proposed change accurately describes this condition and is consistent with 10 CFR 50.82 for FCS.

3.8.11.1 Section 2.8.1, "Refueling Shutdown"

The refueling shutdown TS establishes the requirements for refueling shutdown including RCS boron concentration, nuclear instrumentation, and shutdown cooling system train operability with fuel in the reactor vessel. With the reactor permanently defueled, the refueling shutdown requirements are no longer needed. Therefore, the deletion of TS 2.8.1 in its entirety is acceptable.

3.8.11.2 Section 2.8.2, "Refueling Operations - Containment"

The refueling operations – containment TS addresses containment penetration status, refueling water level inside containment, ventilation isolation, and control room ventilation system operation during core alterations or refueling operations inside containment. With the reactor permanently defueled, the core alterations and refueling operations inside containment are no longer permitted by the license. Therefore, the deletion of TS 2.8.2 in its entirety is acceptable.

3.8.11.3 Section 2.8.3, "Refueling Operations – Spent Fuel Pool"

The refueling operations – SFP TS establishes the acceptable spent fuel assembly storage within the SFP, the requirements during fuel movement within the SFP for SFP water level, SFP

boron concentration, SFP area ventilation, and the operating conditions of the control room envelope and ventilation system.

Section 2.8.3(1), "Spent Fuel Assembly Storage," ensures the SFP k_{eff} (effective neutron multiplication factor) remains less than 0.95 with unborated water. It also ensures that placement of fuel within the SFP meets specific burnup requirements or physical restrictions to prevent exceeding storage reactivity requirements. The proposed revision deletes the references to TS 2.0.1, which relates to LCO application to reactor operation and is proposed for deletion, and other references related to reactor operating conditions. In addition, the revision replaces the "Refueling Operation" defined term with the "Fuel Handling Operations" defined term, consistent with the proposed definition change. These changes reflect the permanent shutdown status of the reactor and the change in scope of fuel handling operations, and, therefore, are acceptable.

Section 2.8.3(2), "Spent Fuel Pool Water Level," ensures that the SFP water level is adequate to satisfy the assumed iodine decontamination factor following an FHA. The proposed revision deletes the references to TS 2.0.1, which relates to LCO application to reactor operation and is proposed for deletion, and other references related to reactor operating conditions. In addition, the revision replaces the "Refueling Operation" defined term with the "Fuel Handling Operations" defined term, consistent with the proposed definition change. These changes reflect the permanent shutdown status of the reactor and the change in scope of fuel handling operations, and, therefore, are acceptable.

Section 2.8.3(3), "Spent Fuel Pool Boron Concentration," establishes the minimum boron concentration requirement with Boral poisoned storage racks to maintain the k_{eff} below 0.95 in the event a misloaded unirradiated fuel assembly is located next to a spent fuel assembly. A misloaded unirradiated fuel assembly at maximum enrichment condition, in the absence of soluble poison, may result in exceeding the design effective multiplication factor. Soluble boron in the SFP water, for which credit is permitted under these conditions, would assure that the effective multiplication factor is maintained substantially less than the design condition. No unirradiated fuel assemblies are currently stored in the SFP. The proposed revision deletes the references to TS 2.0.1, which relates to LCO application to reactor operation and is proposed for deletion, and other references related to reactor operating conditions. In addition, the revision replaces the "Refueling Operation" defined term with the "Fuel Handling Operations" defined term, consistent with the proposed definition change. These changes reflect the permanent shutdown status of the reactor and the change in scope of fuel handling operations, and, therefore, are acceptable.

Section 2.8.3(4), "Spent Fuel Pool Area Ventilation," and Section 2.8.3(5), "Control Room Ventilation System (CRVS)," are proposed for deletion in their entirety. With the exception of the FHA, postulated accidents analyzed in the facility's safety analysis report are no longer credible in the permanently defueled condition, and the revised FHA analysis no longer credits any ventilation or filtration system to satisfy the applicable onsite or offsite dose limits. The SFP area ventilation system and the CRVS are neither assumed to operate as an initial condition of a DBA analysis, nor are they credited as part of the primary success path that functions to mitigate a DBA that assumes the failure of the integrity of a fission product barrier. Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied, and deletion of the LCOs for the SFP area ventilation system and the CRVS in their entirety is acceptable.

3.8.12 Section 2.9, "Radioactive Waste Disposal System"

The radioactive waste disposal system LCO establishes requirements for management of potentially explosive gas mixtures in the WGDTs such that concentrations of hydrogen and oxygen are maintained below flammability limits. The licensee stated that the tanks will be isolated and purged. With no potentially explosive gas mixtures in the tank and no further additions of gas mixtures, management of gas mixtures in the WGDTs is no longer necessary, and, therefore, the deletion of TS 2.9 in its entirety is acceptable.

3.8.13 Section 2.10, "Reactor Core"

The reactor core TS contain the LCOs, Actions, and basis for minimum conditions for criticality, reactivity control systems and core physics parameters limits, and power distribution limits. This section contains the following LCOs:

- 2.10.1 – Minimum Conditions for Criticality
- 2.10.2 – Reactivity Control Systems and Core Physics Parameters Limits
- 2.10.4 – Power Distribution Limits

The LCO 2.10.1, "Minimum Conditions for Criticality," specifies conditions to prevent unanticipated power excursions of an unsafe magnitude.

The LCO 2.10.2, "Reactivity Control Systems and Core Physics Parameters Limits," specifies conditions to ensure (1) adequate shutdown margin following a reactor trip, (2) the moderator temperature coefficient is within the limits of the safety analysis, and (3) control element assembly operation is within the limits of the setpoint and safety analysis.

The LCO 2.10.4, "Power Distribution Limits," specifies conditions to ensure that peak linear heat rates, departure from nucleate boiling ratio margins, and radial peaking factors are maintained within acceptable limits during power operation.

Because OPPD has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel so FCS is no longer in a configuration or a condition under which the TS apply. Based on these findings, the NRC staff concludes that the LCOs in TS Section 2.10 no longer apply. Therefore, the NRC staff finds the deletion of TS Section 2.10 in its entirety is acceptable.

3.8.14 Section 2.12, "Control Room Ventilation System"

The control room ventilation system LCO establishes requirements for the functional capability of the control room air filtration system, the control room air conditioning system, and the control room envelope when RCS temperature is equal to or greater than 210 °F. With the exception of the FHA, postulated accidents analyzed in the facility's safety analysis report are no longer credible in the permanently defueled condition, and the revised FHA analysis no longer credits any ventilation or filtration system to satisfy the applicable on-site or off-site dose limits. The SFP area ventilation system and the CRVS are neither assumed to operate as an initial condition of a DBA analysis, nor are they credited as part of the primary success path that functions to mitigate a DBA that assumes the failure of the integrity of a fission product barrier. Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied, and deletion of TS Section 2.12 in its entirety is acceptable.

3.8.15 Section 2.13, "Limiting Safety System Settings, Reactor Protective System"

The limiting safety system settings, reactor protective system LCO establishes the requirements for the limiting safety system settings of the reactor protection system (RPS) instrumentation, which are the settings for automatic protective devices related to variables that have significant safety functions to correct abnormal situations before a safety limit is exceeded. Because OPPD has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and FCS is no longer in a configuration or a condition under which the TS modes apply. Therefore, deletion of TS Section 2.13 in its entirety is acceptable.

3.8.16 Section 2.14, "Engineered Safety Features System Initiating Instrumentation Settings"

The engineered safety features system initiation instrumentation settings LCO establishes requirements for the engineered safety features instrumentation system settings to ensure that the design and function of engineered safety features instrumentation system are provided to mitigate the consequences of postulated accidents. Because OPPD has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and FCS is no longer in a configuration or a condition under which the TS modes apply. Therefore, deletion of TS Section 2.14 in its entirety is acceptable.

3.8.17 Section 2.15, "Instrumentation and Control Systems"

The instrumentation and control systems LCO establishes requirements for the instrumentation and control systems necessary to assure nuclear safety. The following TSs are being proposed for deletion.

The LCO 2.15.1, "Instrumentation and Control Systems," specifies the instrument operating requirements for the appropriate functional capability of plant equipment, control of process variables, design features, or operating restrictions required for safe operation of the facility. Because of the plant's decommissioned status, the delineation of the necessary conditions of the plant instrumentation and control systems will no longer be required to assure reactor safety. Therefore, the facility's instrument operating requirements for the RPS as stated in TS Table 2-2, and for engineered safety features as stated in TS Table 2-3, will no longer need to be maintained.

The LCO 2.15.2, "Reactor Protective System (RPS) Logic and Trip Initiation," specifies the required actions for the RPS logic and trip initiation channels in MODE 1 (power operation), MODE 2 (Hot Standby), and control element assembly (CEA) drive system in MODE 4 (Cold shutdown). These actions effect automatic trip signals received from RPS instrumentation and provide a means to manually trip the reactor. Because of the plant's decommissioned status, the requirements for RPS logic and trip initiation channels will no longer be required to assure reactor safety and the RPS scram function will no longer be applicable for the facility. Therefore, the facility's instrument operating requirements for isolation functions as stated in TS Table 2-4, and for other safety feature functions as stated in TS Table 2-5, will no longer need to be maintained.

The LCO 2.15.3, "Alternate Shutdown and Auxiliary Feedwater Panel," specifies the required actions for the RPS instrumentation and control and applies to the operational status of the

alternate shutdown and auxiliary feedwater panel functions in MODES 1 and 2. The operability of the alternate shutdown panel, including wide range logarithmic power and source range monitors, and emergency auxiliary feedwater panel instrumentation and control circuits is to ensure that sufficient capability is available to permit entry into and maintenance of the MODE 4 (Cold shutdown) from locations outside of the control room. With the reactor permanently defueled, the alternate shutdown and auxiliary feedwater panel functions of RPS will no longer be required to prevent the violation of safety design limits. Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied, and deletion of TS Section 2.15 in its entirety is acceptable.

3.8.18 Section 2.16, "River Level"

The river level LCO specification and required actions establish limits on river level for reactor operation, required actions to shutdown the reactor if the Missouri River level limits are exceeded, and required actions for monitoring of the river's level. With the exception of the FHA, postulated accidents analyzed in the facility's safety analysis report are no longer credible in the permanently defueled condition, and the revised FHA analysis credits no system dependent on the river level to satisfy the applicable onsite or offsite dose limits. The river level is neither an initial condition of a DBA analysis nor is it credited to support the primary success path that functions to mitigate a DBA that assumes the failure of the integrity of a fission product barrier. Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied, and deletion of TS Section 2.16 in its entirety is acceptable.

3.8.19 Section 2.20, "Steam Generator Coolant Radioactivity"

The steam generator coolant radioactivity LCO establishes the requirement for the limitation on the SG coolant's radioactivity to ensure that the resultant offsite doses will be within the limits of 10 CFR 50.67 in the event of a steam line break. With the exception of the FHA, postulated accidents analyzed in the facility's safety analysis report are no longer credible in the permanently defueled condition, and the revised FHA analysis credits no system dependent on SG coolant reactivity to satisfy the applicable onsite or offsite dose limits. The SG coolant reactivity is neither an initial condition of a DBA analysis nor is it credited to support the primary success path that functions to mitigate a DBA that assumes the failure of the integrity of a fission product barrier. Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied, and deletion of TS Section 2.20 in its entirety is acceptable.

3.8.20 Section 2.21, "Post-Accident Monitoring Instrumentation"

The post-accident monitoring instrumentation LCO establishes requirements for post-accident monitoring instrumentation not included as part of the RPS or engineered safety features. During and following an accident, the post-accident monitoring instrumentation in the control room displays the plant variables to provide the information, which helps the operators determining the plant condition. The TS specifies the operability requirements for the post-accident monitoring instrumentation and backup methods following an accident. These requirements support manual operator actions for which no automatic control is provided. This TS is required to be operable in MODES 1, 2, and 3 (Hot Shutdown). With the reactor permanently defueled, the information necessary to assess the effect of an accident (i.e., core damage) will no longer be required for safety systems to achieve their safety functions for a DBA. Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied, and deletion of TS Section 2.21 in its entirety is acceptable.

3.8.21 Section 2.23, "Steam Generator (SG) Tube Integrity"

The SG tube integrity TS is applicable when the RCS is heated above 210 °F with fuel in the reactor vessel (i.e., the reactor is not in the cold shutdown or refueling mode of operation). With the reactor permanently defueled, SG tube integrity is not required to retain radioactive material because the RCS has no source of energy to drive a radioactive material release. The SG tube integrity is not credited as part of the primary success path that functions to mitigate a DBA that assumes the failure of the integrity of a fission product barrier. Therefore, the criteria of 10 CFR 50.36(c)(2)(ii) are no longer satisfied, and deletion of TS Section 2.23 in its entirety is acceptable.

3.8.22 Section 3.0, "Surveillance Requirements"

The TS Section 3.0, "Surveillance Requirements," establishes the standards and periods used to implement the SRs for plant systems. The licensee proposes changes to the following specific sections:

Section 3.0.2, establishes surveillance intervals for individual specifications. The application states that these sections are being revised or deleted as shown below in bold text. Per the application, the basis for these sections will be revised to reflect permanently defueled conditions.

The proposed changes to Section 3.0.2 are as shown as below:

3.0.2 The surveillance intervals are ~~defined as follows:~~

~~Notation~~	~~Title~~	~~Frequency~~
~~S~~	~~Shift~~	~~At least once per 12 hours~~
~~D~~	~~Daily~~	~~At least once per 24 hours~~
~~W~~	~~Weekly~~	~~At least once per 7 days~~
~~BW~~	~~Biweekly~~	~~At least once per 14 days~~
~~M~~	~~Monthly~~	~~At least once per 31 days~~
~~Q~~	~~Quarterly~~	~~At least once per 92 days~~
~~SA~~	~~Semiannual~~	~~At least once per 184 days~~
~~A~~	~~Annually~~	~~At least once per 366 days~~
~~R~~	~~Refueling~~	~~At least once per 18 months~~
~~P~~	~~Start up~~	~~Prior to Reactor Start up, if not completed in the previous week~~

~~Exception to these intervals are~~ stated in the individual Specifications.

Section 3.0.4, establishes the requirement that surveillances must be met during the modes or other specified conditions in the specification for which the requirements of the LCO apply. The application proposes elimination of the term "MODES or other" from the section as shown below in bold text, since the term will no longer provide a correlation to plant conditions. The licensee's application dated March 31, 2017, states, in part, that:

> Since the FCS license no longer authorizes 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 use of the listed term is no longer required. Based on the above, the proposed revision of this section is acceptable.

Also per the application, the applicable basis section will also be revised. The proposed changes to Section 3.0.4 are as shown in the text below:

> 3.0.4 Surveillance Requirements shall be met during the ~~MODES or other~~ specified conditions in the individual Limiting Conditions for Operation, unless otherwise stated in the Surveillance Requirement. 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 OPERABILITY requirements for the corresponding Limiting Condition for Operation. Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specifications 3.0.1 and 3.0.2, shall constitute noncompliance with the OPERABILITY requirements for the corresponding Limiting Condition for Operation except as provided in Specification 3.0.5. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. Surveillance Requirements do not have to be performed on inoperable equipment.

The NRC staff has reviewed the proposed changes to TS SR 3.0.2. The NRC staff agrees that the statements to be deleted are no longer necessary because the defueled TSs do not contain Frequencies and Completion Times discussed above; therefore, the NRC staff finds the proposed changes are acceptable.

The NRC staff has reviewed the proposed changes to TS SR 3.0.4, and determined that the changes acceptably removes the reference to "MODE," which is no longer applicable to a permanently shutdown and defueled facility. Therefore, the NRC staff finds that the licensee's proposed changes to these SRs are acceptable.

The SRs establish the standards and periods used to implement SRs for plant system. The following TS deletions and modifications are being proposed.

3.8.22 Section 3.1, "Instrumentation and Control"

The instrumentation and control section establishes SRs for the RPS and other critical instrumentation and controls. This section contains several system SRs that will no longer be required with the removal of the specified TS section and system operability requirements. All of the surveillance test requirements from TS 3.1, Tables 3-1 through 3-3A, are proposed to be deleted in their entirety. The postulated accidents and events analyzed in USAR Chapter 14 are no longer applicable in the permanently defueled condition, with the exception of the FHA in the auxiliary building, as discussed in the FHA analysis in Section 3.2 of this safety evaluation. The FHA does not require any of the listed SSCs to mitigate the consequences of the event. Because OPPD has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and none of the instruments or support equipment associated with this section are required to function in the defueled conditions which removes all requirements for testing. Therefore, deletion of TS Section 3.1 in its entirety is acceptable.

3.8.23 Section 3.2, "Equipment and Sampling Tests"

The equipment and sampling tests SRs specify the minimum frequency and type of surveillance applicable to plant equipment and conditions related to safety, as specified in Table 3-4,

"Minimum Frequencies for Sampling Tests," and Table 3-5, "Minimum Frequencies for Equipment Tests." The licensee proposed deletion of the following equipment surveillance tests because the associated system operability requirements in the TSs have been proposed for deletion:

Table 3-4 Items:
1. Reactor coolant; Radioactivity and chemical limit
2. SIRWT; Boric acid concentration
3. Concentrated boric acid tanks; Boric acid concentration
4. SITs; Boric acid concentration
6. SG blowdown; Isotopic analysis for DEI

Table 3-5 items:
1. CEA; Drop time
2. CEA; Partial movement
3. Pressurizer safety valve; lift
4. Main steam safety valve; lift
8.a RCS leakage; evaluate leakage
8.b Primary to secondary leakage; evaluate leakage
9.a Diesel generator fuel supply; inventory
9.b Diesel generator lube oil; inventory
9.c Diesel generator fuel properties; properties
9.d Diesel generator air; pressure
9.e Diesel generator fuel storage; check for water
10.a Control room HEPA/Filter; adsorber in place test, lab test, system operation, initiation
10.b SFP filter; adsorber in place test, lab test, system operation, manual initiation
10.c Safety injection room filter; adsorber in place test, lab test, system operation, initiation
11. Containment ventilation system; damper action
12. Diesel generator under voltage; calibrate
13. Safety injection loop valve motor starter; contact pickup
14. Pressurizer heater; control circuit operation
16. Reactor coolant gas vent system; manual isolation, cycle valves, verify flow
18. Shutdown cooling; verify shutdown cooling loops operable; breaker alignment
19. Refueling water level; verify level above reactor vessel flange
21. Containment penetrations; verify correct position
23. Pressure-temperature (P-T) limit curve; verify P-T Limits Report limits

These equipment tests are no longer necessary to demonstrate the necessary quality of the equipment to meet the operability requirements of LCOs because the LCOs have been proposed for deletion. Therefore, the requirements of 10 CFR 50.36(c)(3) no longer require implementation of the above SRs, and deletion of the above SRs is acceptable.

The licensee proposed retention of the following equipment surveillance tests related to fuel storage because of the continued storage of fuel in the SFP and potential for transfer of fuel to dry casks:

Table 3-4 item:

5. SFP boron concentration

Table 3-5 items:

15. Spent fuel pool racks
20. Spent fuel pool level
22. Spent fuel assembly storage

The deleted items in Table 3-4 resulted in renumbering the footnote (4) as footnote (1) and changed the footnote to state "Weekly when fuel assemblies are stored in the spent fuel pool." The revised footnote is appropriate since no fuel assemblies can be placed in the spent fuel pool.

3.8.24 Section 3.3, "Reactor Coolant System and Other Components Subject to ASME XI Boiler & Pressure Vessel Code Inspection and Testing Surveillance"

Section 3.3, "Reactor Coolant System and Other Components Subject to ASME XI Boiler & Pressure Vessel Code Inspection and Testing Surveillance," establishes the SRs for the RCS and components subject to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI.

As stated in ASME Code, Section XI, "[t]he rules of this section constitute requirements to maintain the nuclear power plant and to return the plant to service, following plant outages, in a safe and expeditious manner." With the exception of the FHA, postulated accidents analyzed in the facility's safety analysis report are no longer credible in the permanently defueled condition, and the revised FHA analysis no longer requires any of the listed SSCs to mitigate the consequences of the event. Because OPPD has submitted certifications pursuant to 10 CFR 50.82(a)(2), the provisions of maintenance of ASME Code, Section XI requirements are no longer needed, and deletion of the TS Section 3.3 in its entirety is acceptable.

3.8.24 Section 3.5, "Containment Test"

The containment test section establishes the SRs for manual locked valves, containment leakage, and structural integrity. The specific surveillance tests include an administrative position check of all "locked closed" manual valves and performance of the Type A, B, and C containment tests specified in Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50. These surveillance tests are no longer necessary to demonstrate the necessary quality of the containment to meet the operability requirements of the containment LCO because the LCO has been proposed for deletion. Furthermore, the certification required under 10 CFR 50.82(a)(1) has been submitted, which renders inapplicable the requirements of 10 CFR 50.54(o). Therefore, the requirements of 10 CFR 50.36(c)(3) no longer require implementation of the above SRs, and deletion of the TS Section 3.5 in its entirety is acceptable.

3.8.25 Section 3.6, "Safety Injection and Containment Cooling Systems Tests"

The safety injection and containment cooling systems tests section establishes the SRs for the safety injection system, the containment spray system, and the containment cooling system and air filtration system inside the containment. With the exception of the FHA, postulated accidents analyzed in the facility's safety analysis report are no longer credible in the permanently defueled condition, and the FHA does not require any of the SSCs associated with this TS section to mitigate the consequences of the event. Therefore, the deletion of TS Section 3.6 in its entirety is acceptable.

3.8.26 Section 3.7, "Emergency Power System Periodic Tests"

The emergency power system periodic tests section establishes periodic testing and SRs of the emergency power system. Since this section exists solely to support the emergency electrical system test requirements, the elimination of the need for the electrical systems also obviates the need for their support systems in the associated TS sections. With the exception of the FHA, postulated accidents analyzed in the facility's safety analysis report are no longer credible in the permanently defueled condition, and the FHA does not require any of the SSCs associated with this TS section to mitigate the consequences of the event. Therefore, the deletion of TS Section 3.7 in its entirety is acceptable.

3.8.27 Section 3.8, "Main Steam Isolation Valves"

The main steam isolation valves section establishes the SRs to verify the ability of the main steam isolation valves to close on signal. This surveillance test is no longer necessary to demonstrate the necessary quality of the main steam isolation valves to meet the operability requirements of the associated LCO because the LCO has been proposed for deletion. Therefore, the requirements of 10 CFR 50.36(c)(3) no longer require implementation of the above SRs, and deletion of the TS Section 3.8 in its entirety is acceptable.

3.8.28 Section 3.9," Auxiliary Feedwater System"

The auxiliary feedwater system section establishes the SRs to verify the ability of the auxiliary feedwater system to respond properly when required. This surveillance test is no longer necessary to demonstrate the necessary quality of the auxiliary feedwater system to meet the operability requirements of the associated LCO because the LCO has been proposed for deletion. Therefore, the requirements of 10 CFR 50.36(c)(3) no longer require implementation of the above SRs, and deletion of the TS Section 3.9 in its entirety is acceptable.

3.8.29 Section 3.10, "Reactor Core Parameters"

The reactor core parameters section establishes requirements regarding the reactor core parameters including shutdown margin, moderator temperature coefficient, linear heat rate, and departure from nucleate boiling margin. Because OPPD has submitted certifications pursuant to 10 CFR 50.82(a)(2), it is prohibited from operating the reactor or placing fuel in the reactor vessel and the testing of the reactor core is no longer required. Based on the above, deletion of TS Section 3.10 in its entirety is acceptable.

3.8.30 Section 3.12, "Radioactive Waste Disposal System"

The radioactive waste disposal system section establishes requirements regarding the flammability concentrations in the Waste Gas Decay Tank (WGDT). In the permanently defueled condition, use of WGDT is no longer required and, accordingly, it will be removed from service. Therefore, no explosive gases will be introduced in the WGDT obviating the need for sampling and deletion of the TS Section 3.12 in its entirety is acceptable.

3.8.31 Section 3.16, "Residual Heat Removal System Integrity Testing"

The residual heat removal system integrity testing section establishes requirements for the integrity of the residual heat removal system and associated components. With the exception of the FHA, postulated accidents analyzed in the facility's safety analysis report are no longer credible in the permanently defueled condition, and the residual heat removal system is not credited to mitigate the FHA. Therefore, the deletion of TS Section 3.16 in its entirety is acceptable.

3.8.32 Section 3.17, "Steam Generator (SG) Tube Integrity"

The SG tube integrity section establishes SRs for the SG tubes. With the exception of the FHA, postulated accidents analyzed in the facility's safety analysis report are no longer credible in the permanently defueled condition, and SG tube integrity is no longer applicable, which removes all requirements for testing. Therefore, the deletion of TS Section 3.17 in its entirety is acceptable.

3.8.33 Section 4.0, "Design Features"

Section 4.1, "Site," provides a description regarding the location of FCS. The licensee proposed to revise this design feature to remove excessive details associated with the site boundary. This change is editorial and will provide a more consistent branch reference and does not change the technical content. Therefore, the proposed change to this section is acceptable.

Section 4.2, "Reactor Core," provides a general description of the number of and design material requirements for the fuel and control element assemblies used in the reactor core. The licensee has proposed to delete the design feature descriptions for fuel and control element assemblies, since they are only applicable to an operating reactor and do not apply to the permanently shutdown and defueled condition of FCS.

The NRC staff has reviewed the proposed changes to delete the reactor core fuel and control element assemblies design features from FCS TSs. Since 10 CFR 50.82(a)(2) prohibits the licensee from operating the reactors or placing fuel in the reactor vessels, the design features related to the reactor core fuel assemblies and control rods are no longer relevant at FCS. Therefore, the NRC staff finds that the licensee's proposed change to delete the TS 4.2 reactor core design features is acceptable.

Section 4.3, "Fuel Storage," provides a description and the requirements regarding prevention of criticality of spent fuel, prevention of SFP drainage, and spent fuel capacity limitations. This TS section is being retained in the permanently defueled TSs, with the exception of Section 4.3.1.2, which is the design and maintenance of the new fuel storage racks as discussed below. The licensee has also made editorial changes to the TS references in this section to conform to the certain retained TSs.

Section 4.3.1.1(e) and (g), Criticality, establishes requirements regarding the design, use, and maintenance of spent fuel storage racks. With the plant in a permanently defueled state, License Condition 2.B.(2) is being proposed for revision as part of this amendment to no longer allow receipt of new fuel. With new fuel no longer stored onsite, the design features associated with new fuel in the spent fuel storage racks are no longer applicable. Therefore, the proposed change to this section is acceptable.

Section 4.3.1.2 has been proposed to be deleted because new fuel is no longer stored onsite and License Condition 2.B.(3) of the renewed facility license is being revised to no longer allow receipt of new fuel. The NRC staff has reviewed the proposed changes to remove the new fuel storage rack design features from the TSs. Since the licensee currently has no new fuel stored onsite and since the facility license will no longer allow new fuel to be stored onsite, the requirements for new fuel storage racks are no longer applicable.

Section 4.3.2, "Drainage," establishes the requirements for SFP level and states:

> The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 23 ft.

The licensee proposed to revise this design feature to improve consistency with the SR for SFP level (i.e., Table 3-5, Item 20) to state:

> The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below 23 ft above the top of irradiated fuel assemblies seated in the storage racks.

This is a clarification of the specific design feature consistent with the requirements of 10 CFR 50.36(c)(4) and, therefore, the change is acceptable.

Based on the above, the NRC staff finds the proposed changes to delete the new fuel storage rack design features from FCS TS 4.3.1.2 to be acceptable. The staff also reviewed the references in TS 4.3.1, "Criticality," and determined that the changes are conforming and editorial in nature. Therefore, the NRC staff finds that the licensee's proposed changes to TS 4.3, "Fuel Storage," are acceptable.

3.8.34 Section 5.0, "Administrative Controls"

Section 5.0, "Administrative Controls," establishes the requirements associated with procedure, program, and reporting requirements. This section is proposed to be revised to include only those administrative requirements needed for safe storage and movement of fuel in the SFP. The acronym "USAR" is being replaced by an equivalent description, "FSAR as updated," to more accurately describe the plant basis requirements and to allow for future decommissioning changes. This is an administrative change and does not affect the technical content or requirements of the TS. Therefore, the proposed change to this section is acceptable.

3.8.35 Section 5.1, "Responsibility," Section 5.2, "Organization," Section 5.3, "Facility Staff Qualification," Section 5.4, "Training," and Section 5.8, "Procedures"

The licensee proposed revisions to or deletion of these administrative controls sections in a letter dated September 28, 2016 (ADAMS Accession No. ML16273A502), in which the licensee proposed to remove portions of the TSs that are no longer applicable to the facility in its permanently defueled condition. Subsequently, the NRC approved the changes in Section 5.1, "Responsibility," Section 5.2, "Organization," Section 5.3, "Facility Staff Qualification," Section 5.4, "Training," and Section 5.8, "Procedures," on July 28, 2017, by License Amendment No. 292 to Renewed Facility Operating License No. DPR-40 (ADAMS Accession No. ML17165A465).

Based on NRC approval of the requested changes by License Amendment No. 292 as stated above, the proposed changes in this application are of a clarifying nature and do not change any technical requirements; therefore, these changes are acceptable.

3.8.36 Section 5.9, "Reporting Requirements"

Section 5.9.3, "Special Reports," establishes the requirements for reporting to the NRC regional office items in addition to those required per 10 CFR. With the deletion of requirements in the associated TS sections, and the removal of license and testing requirements associated with the equipment in the listed reports including; inservice inspection (TS Section 3.3), tendons (TS Section 5.21), materials (TS Section 3.3), post-accident monitoring (TS Section 2.21) with the exception of the FHA, postulated accidents analyzed in the facility's safety analysis report are no longer credible in the permanently defueled condition and the FHA analysis does not take credit for or require the use of any post-accident sampling and monitoring for mitigation of the accident. Since the FCS license no longer authorizes 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 program associated with this section is no longer applicable and there will no longer be reporting items remaining in this section. Therefore, the deletion of TS Section 5.9.3 in its entirety is acceptable.

Section 5.9.4, "Unique Report Requirements," contains the requirements for the annual radiological effluent and radiological environmental reports. This section is being revised to provide a more accurate description of the station's condition. The FCS license no longer authorizes power operations or emplacement or retention of fuel in the reactor vessel, so clarifying the description of the station's conditions in this section is appropriate. The proposed changes to TS Section 5.9.4 do not change the reporting requirements for the annual radiological effluent and radiological environmental reports. Therefore, the NRC staff finds the changes to TS Section 5.9.4 to be acceptable.

Section 5.9.5, "Core Operating Limits Report (COLR)," is generated prior to each reload cycle and contains cycle specific core operating limits and coefficients. Since the FCS 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 information associated the COLR is no longer applicable and the proposed deletion of TS Section 5.9.5 in its entirety is acceptable.

Section 5.9.6, "Reactor Coolant System (RCS) Pressure-Temperature Limits Report (PTLR)," contains RCS pressure and temperature limits for heatup, cooldown, low temperature

overpressure protection, criticality, and hydrostatic testing as well as heatup and cooldown rates utilized in TS Sections 2.1.1 and 2.1.2. Since the FCS license no longer authorizes 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 information associated with the PTLR will no longer be applicable. Therefore, the proposed deletion of TS Section 5.9.6 in its entirety is acceptable.

3.8.37 Section 5.11, "Radiation Protection Program"

Section 5.11.1 establishes the requirements for adequately maintaining high radiation areas. The licensee has proposed to relocate the note associated with Section 5.11.1 to be located on the same page as the requirements. The revision the licensee has proposed is administrative and will not affect the technical requirement or the overall content of this section. Therefore, the proposed revision of TS Section 5.11.1 is acceptable.

3.8.38 Section 5.12, "Environmental Qualification"

Section 5.12 was deleted in a previous amendment (Amendment No. 93 dated December 6, 1985, available in the NRC Legacy Library at Accession No. 851230460). This change removes the title of this section to maintain format consistency. This is an administrative change and does not affect the technical requirements or content of this section. Therefore, the revision to TS Section 5.12 is acceptable.

3.8.39 Section 5.13, "Secondary Water Chemistry"

In Section 5.13, the secondary water chemistry monitoring program contains procedures, sampling points, and sampling frequencies associated with critical parameters of secondary water chemistry to inhibit SG tube degradation. Since the FCS license no longer authorizes 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 program associated with this section is no longer applicable. Therefore, the proposed deletion of TS Section 5.13 in its entirety is acceptable.

3.8.40 Section 5.14, "Systems Integrity"

Section 5.14, "Systems Integrity," establishes a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a transient or accident to as low as practical levels shall be implemented. This program includes provisions for preventive maintenance, periodic visual inspections, and integrated leak testing. With the exception of the FHA, postulated accidents analyzed in the facility's safety analysis report are no longer credible in the permanently defueled condition and the FHA does not credit or require the use of containment isolation or limiting RCS leakage for mitigation. Since the FCS license no longer authorizes 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 program associated with this section is no longer applicable. Therefore, the proposed deletion of TS Section 5.14 in its entirety is acceptable.

3.8.41 Section 5.15, "Post-Accident Radiological Sampling and Monitoring"

Section 5.15, "Post-Accident Radiological Sampling and Monitoring," ensures that the FCS has programs in place to accurately monitor and/or sample and analyze radiological effluents and concentrations in a post-accident condition. With the exception of the FHA, postulated accidents analyzed in the facility's safety analysis report are no longer credible in the

permanently defueled condition and the FHA analysis does not take credit for or require the use of any post-accident sampling and monitoring for mitigation of the accident. Since the FCS license no longer authorizes 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 program associated with this section is no longer applicable. Therefore, the proposed deletion of TS Section 5.15 in its entirety is acceptable.

3.8.42 Section 5.16.1, "Radioactive Effluent Controls Program"

Section 5.16.1, "Radioactive Effluent Controls Program," establishes a program to conform to the regulations in 10 CFR 50.36a. The licensee has proposed revision to this section of the TS to replace the use of the term "operability" with "functionality." The use of the term functionality would correctly describe the status of equipment in the Offsite Dose Calculation Manual since operability is a term used to describe the status of equipment contained in TS. In addition, the licensee has proposed removing the use of Iodine-131 as an isotope for offsite monitoring. Based on the time since FCS has shut down, the use of iodine as a dose limitation at the site boundary is no longer necessary. Therefore, the change to TS Section 5.16.1 is acceptable.

3.8.43 Section 5.19, "Containment Leakage Rate Testing Program"

Section 5.19, "Containment Leakage Rate Testing Program," establishes the administrative program for implementation of the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J. These leakage tests are no longer necessary to demonstrate the performance of containment because the FCS license no longer authorizes 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), which renders inapplicable the requirements of 10 CFR 50.54(o). Therefore, the requirements of 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, no longer apply, and deletion of TS Section 5.19 in its entirety is acceptable.

3.8.44 Section 5.20, "Technical Specification (TS) Bases Control Program"

Section 5.20, "Technical Specification (TS) Bases Control Program," establishes the requirements to update and maintain plant basis. The acronym "USAR" is being replaced by an equivalent description "FSAR as updated" to more accurately describe the plant basis requirements and to allow for future decommissioning changes. This is an administrative change and does not affect the technical content or requirements of the TS. Therefore, the proposed change to TS Section 5.20 is acceptable.

3.8.45 Section 5.21, "Containment Tendon Testing Program"

Section 5.21, "Containment Tendon Testing Program," establishes controls for monitoring any tendon degradation in prestressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. Since the FCS license no longer authorizes 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 program associated with the containment tendon testing program is no longer applicable. Therefore, the proposed deletion of TS Section 5.21 in its entirety is acceptable.

3.8.46 Section 5.22, "Diesel Fuel Oil Testing Program"

Section 5.22, "Diesel Fuel Oil Testing Program," establishes the administrative program for testing of new and stored diesel fuel oil. These oil tests are no longer necessary to support safe operation of the facility because the emergency diesel generators are not required to be immediately operable. Therefore, the program is not necessary to satisfy the requirements of 10 CFR 50.36(c)(5), and deletion of TS Section 5.22 in its entirety is acceptable.

3.8.47 Section 5.23, "Steam Generator (SG) Program"

Section 5.23, "Steam Generator (SG) Program," ensures SG tube integrity is maintained. With the exception of the FHA, postulated accidents analyzed in the facility's safety analysis report are no longer credible in the permanently defueled condition and the postulated accident analysis does not credit or require the use of SG tubes to mitigate the FHA. Since the FCS license no longer authorizes 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 program associated with this section is no longer applicable. Therefore, the proposed deletion of TS Section 5.23 in its entirety is acceptable.

3.8.48 Section 5.24, "Control Room Envelope Habitability Program"

Section 5.24, "Control Room Envelope Habitability Program," establishes the administrative program for testing of the control room habitability systems to ensure operators can safely implement actions to control the reactor and mitigate accidents from within the control room envelope. These tests of control room habitability systems are no longer necessary to support safe operation of the facility because reactor accidents challenging control room habitability are not possible with the reactor permanently defueled and the FHA analysis no longer relies on the control room habitability systems to protect operators. Therefore, the program is not necessary to satisfy the requirements of 10 CFR 50.36(c)(5), and deletion of TS Section 5.24 in its entirety is acceptable.

3.8.49 Appendix B

Appendix B contained additional short-term license conditions for the operation of FCS. Currently, Appendix B does not contain any additional license conditions. Since the FCS license no longer authorizes 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), applying additional short-term license conditions is no longer necessary. Therefore, the proposed deletion of Appendix B in its entirety is acceptable.

3.8.50 Section 6.0, "Interim Special Technical Specifications"

Section 6.0 contained short-term interim TSs for FCS. Currently, Section 6.0 does not contain any TS requirements. Since the FCS license no longer authorizes 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), applying additional interim special TSs is no longer necessary. Therefore, the proposed deletion of TS Section 6.0 in its entirety is acceptable.

3.9 Changes to Renewed Facility Operating License

In OPPD's March 31, 2017, license amendment request, as supplemented by its letter dated September 26, 2017, the licensee proposed to remove, modify, and add several facility operating license conditions, based on the permanently shutdown and defueled status of FCS.

3.9.1 Changes to License Condition 1.C

Currently, License Condition 1.C reads:

> C. 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 is proposing to revise this license condition to read as follows:

> 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;

The proposed change to the description "the facility will operate" to "the facility will be maintained" provides a more accurate description of the future requirements. Since the FCS license no longer authorizes 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 removal of the operating description provides accuracy in the 10 CFR Part 50 license description. Therefore, the proposed change is consistent with 10 CFR 50.82(a)(2) and is acceptable.

3.9.2 Changes to License Conditions 1.H and 2

Currently, License Condition 1.H reads:

> H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the Commission concludes that the issuance of Renewed Operating License No. DPR-40 is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

The licensee is proposing to revise this license condition to read as follows:

> H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs and considering available alternatives, the Commission concludes that the issuance of Renewed License No. DPR-40 is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

Currently, License Condition 2 reads:

> 2. On the basis of the forgoing findings regarding this facility, Facility Operating License No. DPR-40, issued August 9, 1973, is superseded by

Renewed Facility Operating License No. DPR-40, which is hereby issued to the Omaha Public Power District, to read as follows:

The licensee is proposing to revise this license condition to read as follows:

2. On the basis of the forgoing findings regarding this facility, Facility Operating License No. DPR-40, issued August 9, 1973, is superseded by Renewed Facility License No. DPR-40, which is hereby issued to the Omaha Public Power District, to read as follows:

Pursuant to 10 CFR 50.82(a)(2), the FCS license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel. As such, references to operation of the facility in License Conditions 1.H and 2 are inconsistent with the provisions of 10 CFR 50.82(a)(2). Therefore, the proposed changes are consistent with 10 CFR 50.82(a)(2) and are acceptable.

3.9.3 Changes to License Condition 2.A

Currently, License Condition 2.A reads:

A. This renewed license applies to the Fort Calhoun Station, Unit 1, a pressurized water nuclear reactor and associated equipment (the facility), which is owned by the Omaha Public Power District. The facility is located in Washington County, Nebraska, and is described in the Updated Safety Analysis Report as supplemented, and amended, and the Environmental Report as supplemented and amended.

The licensee is proposing to revise this license condition to read, as follows:

A. This renewed license applies to the Fort Calhoun Station, Unit 1, a pressurized water nuclear reactor and associated equipment (the facility), which is owned by the Omaha Public Power District. The facility is located in Washington County, Nebraska, and is described in the Final Safety Analysis Report as supplemented, amended, and updated and the Environmental Report as supplemented and amended.

The proposed acronym change associated with the "USAR" to "Final Safety Analysis Report" more accurately reflects the use of the design and license basis for FCS. This change is administrative in nature and does not change the technical content of the license condition. Therefore, the change is consistent with the requirements associated with the decommissioning plant and is acceptable.

3.9.4 Changes to License Condition 2.B.(1)

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

(1) 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 at the designated location in Washington County, Nebraska in accordance with the procedures and limitations set forth in this renewed license;

The licensee is proposing to revise this license condition to read as follows:

(1) 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 fuel storage at the designated location in Washington County, Nebraska in accordance with the procedures and limitations set forth in this renewed license;

Pursuant to 10 CFR 50.82(a)(2), the FCS license no longer authorizes use of the facility for power operation or emplacement or retention of fuel into the reactor vessel. As such, reference to operation of the facility in License Condition 2.B.(1) is inconsistent with the provisions of 10 CFR 50.82(a)(2). Therefore, the proposed changes are consistent with 10 CFR 50.82(a)(2) and are acceptable.

3.9.5 Changes to License Condition 2.B.(2)

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

(2) Pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess, and use at any time source and special nuclear material that was used as reactor fuel, in accordance with the limitations for storage, and amounts required for reactor operation, as described in the Updated Safety Analysis Report, as supplemented and amended;

The licensee is proposing to revise this license condition to read as follows:

(2) Pursuant to the Act and 10 CFR Parts 40 and 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 Final Safety Analysis Report, as supplemented, amended, and updated;

The licensee stated the proposed revision to this license condition is consistent with the restrictions of 10 CFR 50.82(a)(2) that no longer authorizes operation or emplacement of fuel in the reactor vessel at FCS.

The proposed change removes the authorization for receipt and use of special nuclear material (SNM) as reactor fuel and eliminates the reference to use of the SNM for reactor operations and limits the possession of SNM to SNM "that was used" as reactor fuel at FCS. Pursuant to 10 CFR 50.82(a)(2), the 10 CFR Part 50 license for FCS no longer authorizes operation of the reactor. As such, the licensee has no need to receive SNM in the form of reactor fuel and cannot use SNM as reactor fuel for reactor operations. The continued authorization to possess SNM "that was used" as reactor fuel is necessary as the licensee currently possesses the reactor fuel that was used for the past operations of the FCS reactor. Based on the above, the proposed change to License Condition 2.B.(2) is consistent with the permanently shutdown status of FCS and is, therefore, acceptable.

3.9.6 Changes to License Condition 2.B.(3)

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

(3) 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 is proposing to revise this license condition to read as follows:

(3) 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 and reactor instrumentation; and fission detectors;

The licensee proposed this license condition for revision to be consistent with the restriction of 10 CFR 50.82(a)(2) that FCS is no longer authorized to operate. The proposed changes remove the authorization for receipt and use of byproduct, source, and SNM as sealed neutron sources for reactor startup but retains authorization to possess such sources previously used for reactor startup. The deletion of the authorization to receive and use sources for reactor startup is consistent with the fact that FCS is no longer authorized to operate and the continued authorization to possess neutron sources that were used for reactor startup is consistent with the safe storage of byproduct, source, and SNM. As such, the proposed changes to the FCS License Condition 2.B.(3) is consistent with the permanently shutdown status of the facility and is, therefore, acceptable.

3.9.7 Deletion of License Condition 3.A

Currently, License Condition 3.A states:

A. Maximum Power Level

Omaha Public Power District is authorized to operate the Fort Calhoun Station, Unit 1, at steady state reactor core power levels not to exceed 1500 megawatts thermal (rated power).

The revised License Condition 3.A would state:

A. DELETED

The licensee stated that this license condition can be deleted because FCS is permanently shutdown and defueled in accordance with 10 CFR 50.82(a)(2) and therefore power operation is no longer authorized. The NRC staff reviewed the proposed deletion of License Condition 3.A and determined that power operation is no longer authorized at FCS based on the licensee's 10 CFR 50.82(a)(2) certifications of being permanently shutdown and defueled. The licensee is not authorized to operate the FCS at any power. Therefore, deletion of License Condition 3.A is appropriate and acceptable.

3.9.8 Changes to License Condition 3.B

Currently, License Condition 3.B states:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 296, are hereby incorporated in the license. Omaha Public Power District shall operate the facility in accordance with the Technical Specifications.

The licensee is proposing to revise this license condition to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 297, are hereby replaced with the Permanently Defueled Technical Specifications (PDTS). Omaha Public Power District shall maintain the facility in accordance with the Permanently Defueled Technical Specifications.

The proposed change included in this amendment incorporates the Permanently Defueled Technical Specifications. The paragraph is changed to reflect the nomenclature change to more accurately describe the document. Also changed is the designation from operating to maintaining the facility, which describes the defueled condition in which the FCS license no longer allows the use of the facility for power operation as provided in 10 CFR 50.82(a)(2). Therefore, the proposed changes are consistent with 10 CFR 50.82(a)(2) and are acceptable.

3.9.9 Deletion of License Condition 3.E

Currently, License Condition 3.E states:

E. Updated Final Safety Analysis Report

The Omaha Public Power District Updated Final Safety Analysis Report supplement, submitted pursuant to 10 CFR 54.21(d), describes certain future activities to be completed prior to the period of extended operation. The Omaha Public Power District shall complete these activities no later than August 9, 2013, and shall notify the NRC In writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement, as revised, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71(e)(4) following issuance of this renewed license. Until that update is complete, the Omaha Public Power District may make changes to the programs and activities described in the supplement without prior Commission approval, provided that the Omaha Public Power District evaluates each such change pursuant to the criteria

set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

The revised License Condition 3.E would state:

E. DELETED

This is a one-time requirement to update the USAR to include the USAR supplement required by 10 CFR 54.21(d) in the next USAR update as required by 10 CFR 50.71(e). This is duplicative of the existing requirements in 10 CFR 54.21(d) and 10 CFR 50.71(e)(4). Since the USAR update required by this license condition has been previously completed, this license condition has been satisfied and is therefore no longer needed. License Condition 3.E also states that FCS may make changes to the programs and activities described in the supplement without prior NRC approval provided that the changes are made pursuant to 10 CFR 50.59 requirements. The requirements of 10 CFR 50.59 will continue to apply to such changes after the license condition is deleted. After deletion of this license condition, changes to these programs and activities may be made without prior NRC approval provided that FCS evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59. Therefore, deletion of License Condition 3.E is acceptable.

3.9.10 Deletion of License Condition 3.F

Currently, License Condition 3.F states:

F. Appendix B

The Additional Conditions contained in Appendix B, as revised through Amendment No. 286, are hereby incorporated into this license. Omaha Public Power District shall operate the facility in accordance with the Appendix B Additional Conditions.

The revised License Condition 4 would state:

F. DELETED

This license condition is related to additional conditions contained in TS Appendix B. There presently are no conditions in Appendix B and this section is no longer required to support the Permanently Defueled Technical Specifications. Therefore, the deletion of License Condition 3.F is acceptable.

3.9.11 Changes to License Condition 4

Currently, License Condition 4 states:

4. This renewed license is effective as of the date of issuance and shall expire at midnight on August 9, 2033.

The licensee is proposing to revise this license condition to read as follows:

4. This license is effective as of the date of issuance and authorizes ownership and possession of Fort Calhoun Station until the Commission notifies the licensee in writing that the license is terminated.

The licensee stated that this license condition is revised to reflect the permanently defueled condition at the facility in accordance with 10 CFR 50.82(a)(2). The licensee has proposed that this license condition be revised to conform to 10 CFR 50.51, "Continuation of license," in that the license authorizes ownership and possession of FCS until the Commission notifies the licensee in writing that the license is terminated.

The NRC staff reviewed the proposed License Condition 4. The current License Condition 4, which documents the date of the expiration of the renewed license, is no longer necessary for a permanently shutdown condition of the plant in the process of decommissioning. The revised License Condition 4 documents the current condition of the plant and is consistent with language applicable to the facility as stated in 10 CFR 50.51. The revised License Condition 4 is consistent with the regulatory requirements applicable to the facility in the permanently shutdown and defueled condition. Therefore, the revised License Condition 4 is appropriate and acceptable.

3.9.12 TS Appendix A Title

The TS Appendix A title is proposed to be changed to "Permanently Defueled Technical Specification" to be consistent with the current condition of FCS. Since the FCS license no longer authorizes 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 proposed title change of Appendix A is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment on January 9, 2018. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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, published in the *Federal Register* on June 6, 2017 (82 FR 26135), and there has been no public comment on such finding. 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.

6.0 CONCLUSION

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

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Date: March 6, 2018

SUBJECT: FORT CALHOUN STATION, UNIT 1 – ISSUANCE OF AMENDMENT RE: REVISED TECHNICAL SPECIFICATIONS TO ALIGN TO THOSE REQUIREMENTS FOR DECOMMISSIONING (CAC NO. MF9567; EPID L-2017-LLA-0192) DATED MARCH 6, 2018

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