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

June 8, 2016

Mr. Shane M. Marik
Site Vice President and Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station
9610 Power Lane, Mail Stop FC-2-4
Blair, NE 68008

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-426, "REVISE OR ADD ACTIONS TO PRECLUDE ENTRY INTO LCO 3.0.3 - RITSTF INITIATIVE 6B & 6C" (CAC NO. MF6723)

Dear Mr. Marik:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 288 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 11, 2015.

The amendment revises the TSs to provide a short Completion Time to restore an inoperable system for conditions under which the existing TSs require a plant shutdown. The amendment is consistent with NRC-approved TS Task Force (TSTF) traveler TSTF-426, Revision 5, "Revise or Add Actions to Preclude Entry into LCO [Limiting Condition for Operation] 3.0.3 - RITSTF [Risk-Informed TSTF] Initiatives 6b & 6c."

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,

CFLyon

Carl F. Lyon, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures:
1. Amendment No. 288 to DPR-40
2. Safety Evaluation

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

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 288
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 September 11, 2015, 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 amended by changes as indicated in the attachment to this license amendment, and paragraph 3.B. of Renewed Facility Operating License No. DPR-40 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 288, are hereby incorporated in the license. Omaha Public Power District shall operate the facility in accordance with the Technical Specifications.

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

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: June 8, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 288

RENEWED FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

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

License Page

REMOVE	INSERT
-3-	-3-

Technical Specifications

REMOVE	INSERT
2.1 - Page 21	2.1 - Page 21
2.12 - Page 1	2.12 - Page 1
2.12 - Page 2	2.12 - Page 2

(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. 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 in excess of 1500 megawatts thermal (rate power).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 288 are hereby incorporated in the license. Omaha Public Power District shall operate the facility in accordance with the 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.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (continued)
2.1.6 Pressurizer and Main Steam Safety Valves (continued)

d. With both PORVs inoperable in Modes 4 or 5, depressurize and vent the RCS through at least a 0.94 square inch or larger vent within the next 36 hours.

(5) Two power-operated relief valves (PORVs) and their associated block valves shall be operable in Modes 1, 2, and 3.

a. With one or both PORV(s) inoperable and capable of being manually cycled, within 1 hour either restore the PORV(s) to operable status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

b. With one PORV inoperable and not capable of being manually cycled, within 1 hour either restore the PORV to operable status or close its associated block valve and remove power from the block valve; restore the PORV to operable status within the following 72 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

---------------------------NOTE--------------------------------
Not applicable when second PORV intentionally made inoperable.
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c. With both PORVs inoperable and not capable of being manually cycled, within 1 hour close both block valves, remove power from the block valves, and verify LCO 2.5(1), auxiliary feedwater, is met, and restore at least one PORV to OPERABLE status within 8 hours. Otherwise be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

d. With one block valve inoperable, within 1 hour restore the block valve to operable status or place the associated PORV in the closed position. Restore the block valve to operable status within the next 72 hours. Otherwise, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

---------------------------NOTE--------------------------------
Not applicable when second block valve intentionally made inoperable.
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e. With both block valves inoperable, within 1 hour verify LCO 2.5(1), auxiliary feedwater, is met and restore at least one block valve to OPERABLE status within 8 hours. Otherwise, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 36 hours.

Basis

The purpose of the two spring-loaded Pressurizer Safety Valves (PSV's) is to provide Reactor Coolant System (RCS) overpressure protection and thereby ensure that the Safety Limit for RCS pressure (i.e., 2750 psia) is not exceeded for analyzed accidents. The maximum RCS pressure transient for an analyzed accident is associated with a Loss of Load event[(2)].

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.12 Control Room Ventilation System
2.12.1 Control Room Air Filtration System - Operating

Applicability

Applies to the operational status of the control room air filtration system when the reactor coolant temperature T_{cold} ≥ 210°F.

Objective

To assure operability of equipment required to filter control room air following a Design Basis Accident.

Specification

Two control room air filtration trains shall be OPERABLE.

---Note---

The control room envelope (CRE) boundary may be opened intermittently under administrative control.

Required Actions

(1) With one control room air filtration train inoperable for reasons other than (2), restore the inoperable train to OPERABLE status within 7 days.

(2) With one or more control room air filtration trains inoperable due to inoperable CRE boundary:
  a. initiate mitigating actions immediately, AND
  b. verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits, within 24 hours, AND
  c. restore CRE boundary to OPERABLE status within 90 days.

---Note---

Not applicable when second control room air filtration train intentionally made inoperable.

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(3) With two control room air filtration trains inoperable for reasons other than (2),
  a. initiate mitigating actions immediately, AND
  b. verify LCO 2.1.3, “Reactor Coolant Activity,” is met within 1 hour, AND
  c. restore at least one control room filtration train to OPERABLE status within 24 hours.

(4) With the required actions of (1), (2), or (3) not met, be in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 36 hours.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.12 Control Room Ventilation System
2.12.2 Control Room Air Conditioning System

Applicability

Applies to the operational status of the control room air conditioning system when the reactor coolant temperature $T_{cold} \geq 210°F$.

Objective

To assure operability of equipment required to maintain air temperature within the control room following a Design Basis Accident.

Specification

Two control room air conditioning trains shall be OPERABLE.

Required Actions

(1) With one control room air conditioning train inoperable, restore the inoperable train to OPERABLE status within 30 days.

-------------------------NOTE------------------------------
Not applicable when second control room air conditioning train intentionally made inoperable.
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(2) With two control room air conditioning trains inoperable, restore at least one train to OPERABLE status within 24 hours.

(3) With the required actions of (1) or (2) not met, be in HOT SHUTDOWN within 6 hours, and COLD SHUTDOWN within the following 36 hours.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 288 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By application dated September 11, 2015 (Reference 1), Omaha Public Power District (OPPD) requested changes to the Technical Specifications (TSs, Appendix A to Renewed Facility Operating License No. DPR-40) for the Fort Calhoun Station, Unit No. 1 (FCS).

The proposed amendment revises the TSs to provide a short Completion Time (CT) to restore an inoperable system for conditions under which the existing TSs require a plant shutdown. The proposed amendment is consistent with U.S. Nuclear Regulatory Commission (NRC)-approved Technical Specifications Task Force (TSTF) Traveler TSTF-426, Revision 5, "Revise or Add Actions to Preclude Entry into LCO [Limiting Condition For Operation] 3.0.3 - RITSTF [Risk-Informed TSTF] Initiatives 6b & 6c" (TSTF-426) (Reference 2), as published in the *Federal Register* on May 30, 2013 (78 FR 32476), with certain plant-specific administrative variations.

Traveler TSTF-426 incorporated the approved Topical Report (TR) WCAP-16125-NP-A, "Justification for Risk-Informed Modifications to Selected TSs for Conditions Leading to Exigent Plant Shutdown" (TR WCAP-16125) (Reference 3), into NUREG-1432, Revision 4, "Standard Technical Specifications - Combustion Engineering [(CE)] Plants," April 2012 (STS) (Reference 4). TR WCAP-16125 provided the justification for risk-informed TS (RITS) Initiative 6 for nuclear plants with CE-designed nuclear steam supply systems. RITS Initiative 6 modifies selected exigent shutdown actions to allow a risk-informed operating time prior to shutdown.

Specifically, the licensee's proposed changes would revise TSs 2.1.6(5), "Pressurizer and Main Steam Safety Valves," 2.12.1, "Control Room Air Filtration System - Operating," and 2.12.2, "Control Room Air Conditioning System."

2.0 REGULATORY EVALUATION

The FCS TSs utilize different numbering and titles than the STS on which TSTF-426 was based. References to STS numbers in the model safety evaluation (SE) for TSTF-426 have been changed to the FCS-specific numbers and titles. These differences are administrative and do not affect the applicability of TSTF-426 or the model SE for TSTF-426 to the FCS TSs.

TR WCAP-16125 justified modifications to various TSs to add a Condition for loss of redundant features representing a loss of safety function for a system or component included within the scope of the plant TSs. It would replace Required Actions requiring either a default shutdown or explicit LCO 2.0.1 entry with a Required Action based on the risk significance for the system's degraded condition. The Condition being added is for redundant trains discovered to be inoperable. The Condition only applies to discovery of an emergent condition resulting in redundant trains being inoperable, not from the second train intentionally made inoperable. The CTs associated with the proposed actions are specified. The CTs are intentionally of short duration to allow for restoring the system to an operable condition, thereby avoiding the risk associated with an immediate controlled shutdown. In all but the TS change for pressurizer power-operated relief valves (PORVs), a 24-hour CT is justified. The CT for the pressurizer PORV specification is 8 hours. Table 1 summarizes the TS changes.

Table 1

TS	System/Component	Condition	Current CT	Proposed CT
2.1.6(5)	Power Operated Relief Valves (PORV)	Two PORVs inoperable and not capable of being cycled manually; or Two block valves inoperable	1 hour	8 hours†
2.12.1	Control Room Air Filtration System - Operating (CREACS)	Two trains inoperable (reactor coolant temperature $T_{cold} \geq 210°F$) for reasons other than an inoperable boundary	Explicit LCO 2.0.1	24 hours **
2.12.2	Control Room Air Conditioning System	Two trains inoperable	Explicit LCO 2.0.1	24 hours

† Must include verification that the LCO for auxiliary feedwater (AFW) is met, which requires both trains to be operable. In addition, the new 8-hour CT does not apply in the STS to PORVs which are leaking and unisolable.

** Must include verification that LCO 3.4.16, "RCS Specific Activity," is met.

The Commission's regulatory requirements related to the content of the TS are contained in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, "Technical specifications." Pursuant to 10 CFR 50.36(c), the TS are required to include items in the following specific categories: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) LCOs; (3) surveillance requirements; (4) design features; and (5) administrative controls. The regulation at 10 CFR 50.36(c)(2) states, in part, that "[w]hen a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

NRC Regulatory Guide (RG) 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment (PRA) in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," May 2011 (Reference 5), describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. RG 1.174 also provides risk acceptance guidelines for evaluating the results of such evaluations.

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007, of NUREG-0800, "Standard Review Plan Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) (Reference 6). SRP Section 19.2 states that a risk-informed application should be evaluated to ensure that the proposed change meets the following key principles:

- The proposed change meets the current regulations, unless it is explicitly related to a requested exemption, i.e., a "specific exemption" under 10 CFR 50.12.
- The proposed change is consistent with the defense-in-depth philosophy.
- The proposed change maintains sufficient safety margins.
- When proposed changes increase core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement (60 FR 42622).
- The impact of the proposed change should be monitored using performance measurement strategies.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the licensee's proposed change against the following:

- the requirements of 10 CFR 50.36,
- the STS changes approved for adoption in the Notice of Availability of TSTF-426 as part of the consolidated line item improvement process (CLIIP) announced in the *Federal Register* on May 30, 2013 (78 FR 32476), and
- the methodology approved in TR WCAP-16125, as documented in an SE dated May 24, 2010 (Reference 7). TR WCAP-16125 was reviewed against RG 1.174 and SRP Section 19.2.

3.1 Conformance with the Five Key Principles of SRP Section 19.2 as Summarized in the SE of TR WCAP-16125

The changes proposed in TSTF-426 are consistent with NRC-approved TR WCAP-16125. In its SE (Reference 7), the NRC staff evaluated TR WCAP-16125 for conformance with the five key principles of SRP Section 19.2.

3.1.1 Compliance with Current Regulations

Regulations in 10 CFR 50.36 permit either a plant shutdown or other remedial actions specified by TSs when an LCO is not met. The proposed change provides new action requirements for conditions of equipment inoperability which currently require an immediate plant shutdown. Since such remedial actions are permitted per 10 CFR 50.36, the proposed change continues to comply with current regulations and, therefore, satisfy this key principle.

3.1.2 Defense-in-Depth

The proposed change addresses conditions where both trains of a system are inoperable, resulting in a loss of that system's function and a temporary reduction in the defense-in-depth capabilities of the plant. Each proposed change addresses the remaining available alternative system(s) capable of providing mitigation of events, and, where applicable, includes requirements to assure these required backup systems are operable. The reduced level of defense-in-depth is justified by verification that both trains (if applicable) of the backup system are operable. Therefore, this key principle is satisfied by the unique requirements identified for each proposed TS change.

3.1.3 Safety Margins

The proposed change does not have any impact on the use of NRC-approved codes and standards, nor do the changes impact any acceptance criteria used in a plant's licensing basis. Under the current TSs, if an accident occurs during the 6-hour controlled shutdown time of LCO 2.0.1 caused by two trains of these systems being unavailable, it could potentially result in offsite dose limits that do not meet NRC regulatory limits. Since the changes proposed do not modify the design basis of the systems evaluated, extending the allowed outage time to 24 hours would have no quantitative effect on the dose consequence as compared to the existing condition. As such, the proposed changes would not significantly reduce the plant's available safety margin; therefore, this key principle is satisfied.

3.1.4 Performance Monitoring

The proposed change would permit continued plant operation for short periods to address emergent equipment failures. Degradation of equipment performance could lead to excessive use of the new action requirements. This is adequately addressed by equipment performance monitoring required by 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," and, therefore, this key principle is satisfied.

3.1.5 Risk Assessment

The risk of each of the TS LCOs for which action requirements are proposed is evaluated in TR WCAP-16125 by three methods, as described below.

Method 1:

For those TSs governing systems or components which provide mitigation of core damage and large early releases, changes in the core damage frequency and large early release frequency (ΔCDF and ΔLERF, respectively) metrics are calculated using a simplified generic method, and the results are compared to the acceptance guidelines of RG 1.174. This applies to TS 2.1.6(5), PORVs.

For calculations of ΔCDF, a bounding approach was applied to evaluate loss of function of a system by identifying the initiating events for which the system provides mitigation, and assuming that the event goes directly to core damage. No credit was taken for alternate mitigation strategies, and the baseline CDF was effectively assumed to be zero. The initiating event frequencies were taken from NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 – 1995," February 1999 (Reference 8). The licensee verified that initiating frequencies in NUREG/CR-5750 are bounding for FCS.

For ΔLERF, a simplified approach using an event tree was developed to calculate the fraction of core damage events which result in large early releases. The event tree assessed containment isolation status, Reactor Coolant System (RCS) pressure, secondary-side depressurization via the steam generators, thermally induced Steam Generator Tube Rupture (SGTR), and Reactor Pressure Vessel (RPV) lower head failure. Assumptions related to the potential impact on LERF for each of these events, and the associated basis for probabilities used in the analysis, are discussed below:

Containment Isolated – This event defines containment integrity prior to the core damage event. If containment is not isolated, then a large early release will result concurrent with core damage. A probability of 3.0E-3 was applied for an unisolated containment, which is identified as the upper end of the range used in the CE PRA models in TR WCAP-16125.

RCS Pressure – High – This event defines the RCS pressure at the time of core damage. If the pressure is low, then large early releases are assumed not to occur (except via an unisolated containment); otherwise, thermally induced SGTR and high-pressure melt ejection events are further evaluated. All core damage events involving loss-of-coolant accidents (LOCAs) are assumed to result in low or intermediate RCS pressure, and all other events result in high RCS pressure.

Steam Generator Depressurization – This event defines the status of the secondary side, and affects the next event which is the potential for induced SGTR. Depressurization of the secondary side occurs either due to prior operator response or due to failure of a safety relief valve. Based on NUREG-1570, "Risk Assessment of Severe Accident Induced Steam Generator Tube Rupture," March 1998 (Reference 9), a probability of 0.9 is assigned for secondary depressurization.

Thermally Induced SGTR Occurs – This event represents a loss of steam generator tube integrity due to thermal stresses during a severe accident, which is assumed to result in a large early release. Two values are used, based on the status of the prior event, for steam generator depressurization. A probability of 0.5 is assigned when the steam generators are depressurized, and 0.01 otherwise. These values are conservative, based on the assumptions regarding tube age and integrity and based on neglecting operator actions to depressurize the RCS after core damage.

RPV Lower Head Failure Results in Containment Failure – This event represents a high-pressure failure of the lower head, with an energetic discharge of the molten fuel and direct containment heating, leading to failure of containment. Based on NUREG/CR-6338, "Resolution of Direct Containment Heating Issue for all Westinghouse Plants with Large Dry Containments or Subatmospheric Containments," February 1996 (Reference 10), the conditional containment failure probability given the event for CE-designed plants is 0.01, which is considered to be a bounding value.

None of the assessed initiating events include either SGTRs or other containment bypass events because the systems being evaluated do not mitigate these events. The NRC staff concludes that the simplified LERF event tree is reasonable and acceptable to support the evaluation of LERF for the scope of TR WCAP-16125.

Method 2:

The remaining systems (and associated TS) associated with mitigation of radiological releases with magnitudes less than those associated with LERF are: Control Room Air Filtration System - Operating (TS 2.12.1); and Control Room Air Conditioning System (TS 2.12.2). There is no impact to either CDF or LERF, as the systems are provided to meet design basis dose limits. As described in TR WCAP-16125, an evaluation of the frequency of events which challenge the systems was made and compared to the acceptance guidelines of RG 1.174 applicable to ΔLERF in order to characterize the risk of these lesser releases. TR WCAP-16125 provided additional justification based on the availability of other systems which provide a degree of defense-in-depth for prevention of these releases.

To assess the impact of the unavailability of these systems, TR WCAP-16125 examined the expected iodine releases for three categories of events:

- Beyond design basis scenarios that lead to large early releases,
- Maximum hypothetical accident, and
- LOCA and non-LOCA design-basis accidents.

To reduce the impact of an increased CT for the Control Room Air Filtration System, TR WCAP-16125 added conditions to verify that RCS specific activity is within limits and to verify that dose mitigating actions are available in the control room. For limited durations, such as the short-term operational conditions proposed by the increased CT for the Control Room Air Filtration System, the NRC staff has accepted credit for the use of respirators and potassium iodide on an interim basis to demonstrate that control room dose limits can be met.

Similarly, TR WCAP-16125 added pre-planned actions to ensure that the impact of loss of post-accident temperature control associated with an increased CT for the Control Room Air Conditioning System is mitigated. Actions can include use of portable fans, temporary opening of doors, or use of normal heating, ventilation, and air conditioning systems. To support this change, administrative controls will be provided to monitor the control room temperature to ensure control room habitability and operability of TS equipment. If compensatory measures impact the control room envelope, the operability of containment and auxiliary building post-accident air cleanup systems will be verified. The 24-hour CT proposed in TR WCAP-16125 for the Control Room Air Filtration System and the Control Room Air Conditioning System is consistent with the allowed 24-hour period for the evaluation of a breach of the control room envelope provided in Traveler TSTF-448 (Reference 12).

The NRC staff has reviewed the bases for the increased CT for the Control Room Air Filtration System and the Control Room Air Conditioning System and has determined that the proposed conditions and compensatory measures provide reasonable assurance that control room habitability will be adequately maintained during the proposed 24-hour CT.

External events, including internal fires and floods, were not evaluated in TR WCAP-16125. None of the systems being evaluated provide a primary mitigating function for external events, and therefore these events are not significant to the risk-informed decision.

TR WCAP-16125 also evaluated sensitivity studies for key areas of uncertainty in the analyses. Specifically, TR WCAP-16125 considered uncertainties in the initiating event frequencies which are the input to the CDF calculations and showed that even assuming a 95 percent upper bound frequency would not result in excessive risk. These were also propagated into the LERF calculations with similar results. TR WCAP-16125 also addressed uncertainties in the thermally induced SGTR assumptions and steam generator depressurization assumptions, and demonstrated that the LERF results are not significantly impacted. These sensitivity studies performed to evaluate the key sources of uncertainty in the risk analyses adequately demonstrate the robustness of the results to support the proposed TS changes.

3.2 TS Changes

This section provides a description of each TS change and the NRC staff evaluation of each proposed TS change. The NRC staff's evaluation approves only the proposed changes to the TSs as described below.

TS 2.1.6(5) – Pressurizer Power Operated Relief Valves

The pressurizer PORVs and associated block valves are required to be operable to minimize the potential for a small-break LOCA through a PORV pathway. The PORVs automatically open for RCS pressure control to avoid challenging the primary safety relief valves, and may be manually opened by the operator to control pressure. In the event of a total loss-of-feedwater to the steam generators, one or more PORVs may be opened manually to provide for feed-and-bleed cooling of the reactor using once-through cooling from high-pressure injection to the RCS. The PORVs may also be used for low-temperature overpressure protection during heatup and cooldown. The PORV may be manually operated to depressurize the RCS in

response to normal or abnormal transients. The PORV may be used for depressurization when pressurizer spray is not available, a condition that may be encountered during a loss-of-offsite power. The PORVs can be manually operated to reduce RCS pressure in the event of an SGTR with a loss-of-offsite power.

The license amendment request provides an 8-hour CT to restore at least one PORV or one block valve to operable status before shutdown. This action may only be applied provided the PORV is isolable by its block valve.

The current TS requirement is to close the associated block valve, and if this cannot be accomplished, an immediate plant shutdown is required. The risk result of unavailable PORVs or block valves is primarily attributable to the non-design basis function of providing for feed-and-bleed cooling:

ΔCDF	RG 1.174 Guidance	ΔLERF	RG 1.174 Guidance
1.5E-7/yr	<1.0E-6/yr	1.1E-8/yr	<1.0E-7/yr

The ΔCDF and ΔLERF were assessed based on a bounding once-per-3-year entry into the proposed action requirement from TR WCAP-16125 and assumed that the entire 8-hour duration of the CT is used. The risk results are well below the acceptance guidelines of RG 1.174 as noted in the table.

The primary safety relief valves provide the design-basis pressure control function, controlled by TS requirements. The non-design basis feed-and-bleed function is considered to be risk significant, and the proposed change includes a TS requirement to confirm that the LCO for auxiliary feedwater (AFW) is met, which requires both trains to be operable. A reduced level of defense-in-depth is retained by verification of the operability of the both AFW pumps. These requirements assure that mitigation capability is available for those design-basis accidents or anticipated operational occurrences requiring the pressure control and heat removal functions of the PORVs.

In addition, the new 8-hour CT does not apply to PORVs which are leaking and unisolable.

TS 2.1.6, Condition 5c, states that with two PORVs inoperable and not capable of being manually cycled, close and remove power from the associated block valves within 1 hour and be in HOT SHUTDOWN in 6 hours and COLD SHUTDOWN in 36 hours. Condition 5c is modified to add new Required Actions to verify that LCO 2.5(1), "Auxiliary Feedwater," is met (i.e., all AFW trains are operable) within 1 hour and to restore at least one PORV to operable status within 8 hours. Condition 5c is modified by a note stating it is not applicable when the second PORV is intentionally made inoperable.

Proposed new Condition 5e, applies when two block valves are inoperable. Condition 5e requires verification that LCO 2.5(1), "Auxiliary Feedwater," is met (i.e., all AFW trains are operable) within 1 hour and to restoration of at least one block valve to operable status within 8 hours. Otherwise, the plant is required to be in HOT SHUTDOWN in 6 hours and COLD SHUTDOWN in 36 hours. Condition 5e is modified by a note stating it is not applicable when the second block valve is intentionally made inoperable.

The licensee modified the pressurizer PORV TS to contain requirements equivalent to NUREG-1432 with regard to leaking and unisolable PORVs. The NRC staff reviewed the modified TS and finds it to be equivalent and therefore acceptable.

The risk result is within the acceptance guidelines of RG 1.174, and there is an additional restriction on operability of both AFW trains in the TS action. Therefore, the NRC staff concludes that the proposed new action requirement and the 8-hour CT are acceptable.

TS 2.12.1 – Control Room Air Filtration System - Operating

The Control Room Air Filtration System provides for filtration of outside air delivered to the control room by the ventilation system in the event of radioactive releases of particulates or iodine from containment following an accident involving fuel failures. This is to assure that control room personnel are protected from potential radiation exposures in excess of regulatory limits. The system may also provide protection of control room personnel from chemical or toxic gas releases by isolating the control room air intakes.

The current TS 2.12.1 does not address the condition of two inoperable trains of these systems; therefore, a default LCO 2.0.1 entry is required, resulting in an immediate plant shutdown. The proposed change would provide a 24-hour CT to restore at least one train of the Control Room Air Filtration System to operable status, to permit continued operation under an existing action requirement. The current TS already provide a 24-hour CT when both trains are inoperable due specifically to control room pressure boundary inoperability.

In the event of an accident involving radioactive releases without the availability of the Control Room Air Filtration System, there would be no direct impact on the capability of the control room staff to perform any actions required to mitigate severe core damage or large early releases, because alternative protective measures would be implemented to reduce the dose impacts. If the accident did not involve severe core damage, control room doses even without the Control Room Air Filtration System would be minimal, and therefore the Control Room Air Filtration System has no direct role in preventing core damage (i.e., ΔCDF = 0). If a core damage accident did occur with Control Room Air Filtration System unavailable, then the bounding impact would be to simply assume the event proceeded to a large early release based on the unavailability of the control room personnel to perform any mitigating actions. This assumption would be very conservative, since large releases occur primarily due to containment bypass accidents, and control room actions following core damage do not prevent the release from occurring.

A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event, which with the Control Room Air Filtration System unavailable is assumed to proceed to a large early release, would be:

(1E-4/year) x (24 hours) x (year/8760 hours) = 2.7E-7

Assuming a once-per-3-year entry into the new TS, and assuming the entire 24-hour duration of the CT is used, the conservatively calculated ΔLERF is about 9.0E-8/year. This ΔLERF, and the zero ΔCDF, are below the acceptance guidelines of RG 1.174.

A significant contributor to control room radiological hazards was identified in TR WCAP-16125 from the release of radioactive RCS fluid from an SGTR event. A required TS action to verify LCO 3.4.16, "RCS Specific Activity," is met will be included in the new proposed action to provide additional defense-in-depth.

TR WCAP-16125 also addressed a TS action to require initiation of mitigating actions to lessen the effects of potential hazards of smoke, chemical, radiological, or toxic gas releases. The NRC staff considers the specific hazards and compensatory measures to be plant-specific, and did not find sufficient information to conclude that the proposed changes are acceptable for these events without a plant-specific evaluation. The request for additional information response dated July 8, 2009 (Reference 13), identifies that these mitigating actions were previously reviewed and approved by the NRC staff for Traveler TSTF-448 (Reference 12). TSTF-448 authorizes a generic TS change to permit a 24-hour CT when the control room boundary is inoperable, and includes the same mitigating actions to assure protection of the control room staff from non-radiological hazards.

TS 2.12.1, Required Action 4 applies when two Control Room Air Filtration System trains are inoperable due to any reason other than an inoperable control room boundary and requires entering LCO 2.0.1 immediately. The TR WCAP-161254 justifies a 24-hour CT for two Control Room Air Filtration System trains inoperable for any reason provided that mitigating actions are implemented immediately and it is verified that LCO 2.1.3, "Reactor Coolant Activity," is met within 1 hour. Required Action 4 is revised to require restoration of at least one Control Room Air Filtration System train to operable status within 24 hours and renumbered as Required Action 3. Proposed Required Action 3 is modified by a note stating it is not applicable when the second Control Room Air Filtration System train is intentionally made inoperable. Existing Required Action 3 requires entering HOT SHUTDOWN in 6 hours and COLD SHUTDOWN in 36 hours when Required Actions 1 or 2 are not met. Existing Required Action 3 is moved to Required Action 4 and is modified to also apply when new Required Action 3 is not met.

The requirement to immediately "initiate action to implement mitigating actions" in Required Action 3 is the same as in existing Action 2.a. Action 2.a was added by approved TSTF-448. Required Action 4 is equivalent to the action to take mitigating actions in Required Action 2.

Based on the risk result being below the acceptance guidelines of RG 1.174 and the additional restriction on meeting RCS specific activity limits in the TSs, the NRC staff concludes that the proposed new action requirement and 24-hour CT are acceptable.

<u>TS 2.12.2 – Control Room Air Conditioning System</u>

The Control Room Air Conditioning System provides for temperature control of the control room when it is isolated during accident conditions. This assures control room temperature will not exceed equipment operability requirements.

The current TS 2.12.2 requires entry into LCO 2.0.1 when both Control Room Air Conditioning System trains are inoperable. The proposed change would provide a 24-hour CT to restore at least one Control Room Air Conditioning System train to operable status, to permit continued operation under an existing action requirement.

TR WCAP-16125 stated that the unavailability of the Control Room Air Conditioning System has a negligible impact on severe accident risk, based on long room heatup times, availability of alternate cooling strategies, and alternate means to control emergency systems locally. The NRC staff reviewed the basis for this conclusion and considered the potential plant impacts if an accident occurred which isolated the control room while the Control Room Air Conditioning System was inoperable.

If an accident occurred which isolated the control room without cooling, and core cooling was being maintained by the Emergency Core Cooling System, then there would be negligible radiological consequences and the operators could simply unisolate and realign the normal control room ventilation system to provide continued cooling of the control room. Therefore, there would be no impact on CDF (i.e., ΔCDF = 0).

If core damage occurred after the accident and the control room needed to remain isolated without cooling, the bounding impact would be to simply assume the event proceeded to a large early release based on the unavailability of the control room personnel to perform any mitigating actions. This assumption would be very conservative, since large releases occur primarily due to containment bypass accidents, and control room actions following core damage do not prevent the release from occurring.

A bounding estimate for CDF of CE plants was identified as 1E-4/year, so that over a 24-hour period the probability of a significant core damage event, which with the Control Room Air Conditioning System unavailable is assumed to proceed to a large early release, would be:

$$(1E\text{-}4/\text{year}) \times (24 \text{ hours}) \times (\text{year}/8760 \text{ hours}) = 2.7E\text{-}7$$

Assuming a once-per-3-year entry into the new TS, and assuming the entire 24-hour duration of the CT is used, the conservatively calculated ΔLERF is about 9.0E-8/year. This ΔLERF, and the zero ΔCDF, are below the acceptance guidelines of RG 1.174.

Defense-in-depth is provided by alternative control room cooling actions and by the capability for local operation of equipment, if necessary. These actions are typically found in plant procedures, and are not required to be implemented by TS controls. The licensee confirmed in the license amendment request that plant procedures can establish temporary alternate means of control room cooling.

TS 2.12.2, Required Action 3, applies when two Control Room Air Conditioning System trains are inoperable and requires entering LCO 2.0.1 immediately. TR WCAP-16125 justifies a 24-hour CT for two Control Room Air Conditioning System trains inoperable. New Required Action 2 requires restoration of at least one Control Room Air Conditioning System train to operable status within 24 hours. New Required Action 2 is modified by a note stating it is not applicable when the second Control Room Air Conditioning System train is intentionally made inoperable. The existing Required Action 2 is renumbered to Required Action 3 and requires entering HOT SHUTDOWN in 6 hours and COLD SHUTDOWN in 36 hours when Required Actions 1 or 2 are not met.

Based on the risk result being below the acceptance guidelines of RG 1.174, the NRC staff concludes that the proposed new action requirement and 24-hour CT are acceptable.

3.3 TS Bases Changes

TSTF-426 included and the licensee submitted the following TS Bases changes:

- A reference to the NRC-approved TR WCAP-16125 has been added to the reference section of the TS Bases for each TS affected in TSTF-426.
- Revisions to reflect the changes to the TS.
- For all affected TS, a Note on each applicable condition was added that states: "Not applicable when second [system or component name] intentionally made inoperable." The Bases are revised to provide additional explanation of the Note: "The Condition is modified by a Note stating it is not applicable if the second [system or component name] is intentionally declared inoperable. The Condition does not apply to voluntary removal of redundant systems or components from service. The Condition is only applicable if one [system or component name] is inoperable for any reason and the second [system or component name] is discovered to be inoperable, or if both [system or component name] are discovered to be inoperable at the same time."

The NRC staff determined that TS Bases changes are consistent with the proposed TS changes and provide the purpose for each requirement in the specification consistent with the Commission's Final Policy Statement on TSs Improvements for Nuclear Power Reactors, dated July 2, 1993 (58 FR 39132).

3.4 Summary

The NRC staff has reviewed the proposed change against approved Traveler TSTF-426, which was based on approved TR WCAP-16125 (using the five key principles of risk-informed decision making) and concludes that the proposed change is acceptable. Appropriate TS notes are provided which assure that the loss of safety function action requirements are not applicable for operational convenience and that voluntary entry into these action requirements in lieu of other alternatives that would not result in redundant systems or components being inoperable are prohibited.

The NRC staff further notes that the proposed change does not alter the regulations for notifications and reports required by 10 CFR Part 50 involving the loss of safety function, and that any plant-specific license amendment which provides a condition to address a loss of safety function would not obviate the requirement for a licensee to provide such notifications and reports.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official, Ms. Julia Schmitt, was notified on May 18, 2016, of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to 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 and there has been no public comment on such finding published in the *Federal Register* on November 24, 2015 (80 FR 73239). 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.

7.0 REFERENCES

1. Cortopassi, L. P., Omaha Public Power District, letter to U.S. Nuclear Regulatory Commission, "License Amendment Request (LAR) 15-05; Application to Revise Technical Specification to Adopt TSTF-426, 'Revise or Add Actions to Preclude Entry into LCO 3.0.3 – RITSTF Initiative 6B & 6C,' Using the Consolidated Line Item Improvement Process," dated September 11, 2015 (ADAMS Accession No. ML15254A445).

2. Technical Specifications Task Force, letter to U.S. Nuclear Regulatory Commission, "Transmittal of TSTF-426, Revision 5, 'Revise or Add Actions to Preclude Entry into LCO 3.0.3 – RITSTF Initiatives 6b and 6c,' dated November 22, 2011 (ADAMS Accession No. ML113260461).

3. PWR Owners Group, letter to U.S. Nuclear Regulatory Commission, "PWR Owners Group Submittal of WCAP-16125-NP-A, Revision 2, 'Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown,' dated August 2010 (PA-LSC-0354)," dated January 1, 2011 (ADAMS Package Accession No. ML110070498).

4. U.S. Nuclear Regulatory Commission, NUREG-1432, "Standard Technical Specifications - Combustion Engineering Plants," April 2012 (ADAMS Accession No. ML12102A165).

5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," May 2011 (ADAMS Accession No. ML100910006).

6. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007 (ADAMS Accession No. ML071700658).

7. Blount, T. B., U.S. Nuclear Regulatory Commission, letter to Anthony Nowinowski, Westinghouse Electric Company, "Final Safety Evaluation of Pressurized Water Reactor Owners Group Topical Report WCAP-16125-NP, Revision 2, 'Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown,' dated May 24, 2010 (ADAMS Package Accession No. ML093560466).

8. U.S. Nuclear Regulatory Commission, NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 – 1995," February 1999 (ADAMS Accession No. ML070580080).

9. U.S. Nuclear Regulatory Commission, NUREG-1570, "Risk Assessment of Severe Accident Induced Steam Generator Tube Rupture," March 1998 (ADAMS Accession No. ML070570094).

10. U.S. Nuclear Regulatory Commission, NUREG/CR-6338, "Resolution of Direct Containment Heating Issue for all Westinghouse Plants with Large Dry Containments or Subatmospheric Containments," February 1996 (ADAMS Accession No. ML081920672).

11. Buschbaum, D., PWR Owners Group, letter to U.S. Nuclear Regulatory Commission, "Responses to the NRC Request for Additional Information (RAI) on Topical Report (TR) WCAP-16125-NP, Revision 1, 'Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown,' (PA-LSC-0364 Revision 2)," dated August 10, 2009 (ADAMS Accession No. ML092260399).

12. Technical Specifications Task Force, letters to U.S. Nuclear Regulatory Commission, "TSTF-448-A, Revision 3, 'Control Room Habitability,' dated August 8, 2006, and "Corrected Pages for TSTF-448, Revision 3, 'Control Room Habitability,'" dated December 29, 2006 (ADAMS Accession Nos. ML062210095 and ML063630467, respectively).

13. PWR Owners Group, letter to U.S. Nuclear Regulatory Commission, "Responses to the NRC Request #2 for Additional Information (RAI) on Topical Report (TR) WCAP-16125-NP, Revision 1, 'Justification for Risk-Informed Modifications to Selected Technical Specifications for Conditions Leading to Exigent Plant Shutdown,' (PA-LSC-0364 Revision 2)," dated July 8, 2009 (ADAMS Accession No. ML091940063).

Principal Contributor: M. Hamm, NRR/DSS/STSB

Date: June 8, 2016

June 8, 2016

Mr. Shane M. Marik
Site Vice President and Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station
9610 Power Lane, Mail Stop FC-2-4
Blair, NE 68008

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-426, "REVISE OR ADD ACTIONS TO PRECLUDE ENTRY INTO LCO 3.0.3 - RITSTF INITIATIVE 6B & 6C" (CAC NO. MF6723)

Dear Mr. Marik:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 288 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 11, 2015.

The amendment revises the TSs to provide a short Completion Time to restore an inoperable system for conditions under which the existing TSs require a plant shutdown. The amendment is consistent with NRC-approved TS Task Force (TSTF) traveler TSTF-426, Revision 5, "Revise or Add Actions to Preclude Entry into LCO [Limiting Condition for Operation] 3.0.3 - RITSTF [Risk-Informed TSTF] Initiatives 6b & 6c."

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,
/RA/

Carl F. Lyon, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures:
1. Amendment No. 288 to DPR-40
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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MHamm, NRR/DSS/STSB

ADAMS Accession No. ML16139A804 ***memo dated**

OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA	NRR/DSS/STSB/BC
NAME	FLyon	JBurkhardt	AKlein*
DATE	5/20/16	5/20/16	5/11/16
OFFICE	OGC NLO	NRR/DORL/LPL4-1/BC	NRR/DORL/LPL4-1/PM
NAME	STurk	RPascarelli	FLyon
DATE	6/1/16	6/8/16	6/8/16

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