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

October 9, 2009

Mr. David J. Bannister
Vice President and CNO
Omaha Public Power District
Fort Calhoun Station
444 South 16th St. Mall
Omaha, NE 68102-2247

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: REQUEST TO ADD STEAM GENERATOR BLOWDOWN ISOLATION REQUIREMENTS TO TECHNICAL SPECIFICATIONS (TAC NO. ME0596)

Dear Mr. Bannister:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 263 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 January 30, 2009, as supplemented by letters dated June 30 and August 28, 2009.

The amendment adds steam generator blowdown isolation operability and surveillance requirements to the TSs. These requirements will be needed as a result of a plant modification to add an interlock to automatically isolate steam generator blowdown following a reactor trip. In addition, the amendment modifies the remedial actions for channels that currently contain a key-operated bypass switch by deleting the term "key-operated" so that channels with a bypass switch that are not key-operated may be bypassed as well.

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,

Nicholas D. Kramer for

Lynnea E. Wilkins, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures:
1. Amendment No. 263 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. 263
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 January 30, 2009, as supplemented by letters dated June 30 and August 28, 2009, 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. 263, are hereby incorporated in the license. The licensee 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 prior to startup from the 2009 refueling outage, which is scheduled to commence on November 1, 2009.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
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: October 9, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 263

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
2.15 – Page 1	2.15 – Page 1
2.15 – Page 12	2.15 – Page 12
2.15 – Page 13	2.15 – Page 13
3.1 – Page 13	3.1 – Page 13

(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. 263 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.

2.0 LIMITING CONDITIONS FOR OPERATION

2.15 Instrumentation and Control Systems

Applicability

Applies to plant instrumentation systems.

Objective

To delineate the conditions of the plant instrumentation and control systems necessary to assure reactor safety.

Specifications

The operability, permissible bypass, and Test Maintenance and Inoperable bypass specifications of the plant instrument and control systems shall be in accordance with Tables 2-2 through 2-5.

(1) In the event the number of channels of a particular system in service falls one below the total number of installed channels, the inoperable channel shall be placed in either the bypassed or tripped condition within one hour if the channel is equipped with a bypass switch, and eight hours if jumpers or blocks must be installed in the control circuitry. The inoperable channel may be bypassed for up to 48 hours from time of discovering loss of operability; however, if the inoperability is determined to be the result of malfunctioning RTDs or nuclear detectors supplying signals to the high power level, thermal margin/low pressurizer pressure, and axial power distribution channels, these channels may be bypassed for up to 7 days from time of discovering loss of operability. If the inoperable channel is not restored to OPERABLE status after the allowable time for bypass, it shall be placed in the tripped position or, in the case of malfunctioning RTDs or linear power nuclear detectors, the reactor shall be placed in hot shutdown within 12 hours. If active maintenance and/or surveillance testing is being performed to return a channel to active service or to establish operability, the channel may be bypassed during the period of active maintenance and/or surveillance testing. This specification applies to the high rate trip-wide range log channel when the plant is at or above 10^{-4}% power and is operating below 15% of rated power.

(2) In the event the number of channels of a particular system in service falls to the limits given in the column entitled "Minimum Operable Channels," one of the inoperable channels must be placed in the tripped position or low level actuation permissive position for the auxiliary feedwater system within one hour, if the channel is equipped with a bypass switch, and within eight hours if jumpers or blocks are required; however, if minimum operable channel conditions for SIRW tank low signal are reached, both inoperable channels must be placed in the bypassed condition within eight hours from time of discovery of loss of operability. If at least one inoperable channel has not been restored to OPERABLE status after 48 hours from time of discovering loss of operability, the reactor shall be placed in a hot shutdown condition within the following 12 hours; however, operation can continue without containment ventilation isolation signals available if the containment ventilation isolation valves are closed.

TABLE 2-4

Instrument Operating Conditions for Isolation Functions

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
1	Containment Isolation				
A	Manual	1	None	None	N/A
B	Containment High Pressure				
	Logic Subsystem A	2(a)(e)(g)	1	During Leak Test	(f)
	Logic Subsystem B	2(a)(e)(g)	1		
C	Pressurizer Low/Low Pressure				
	Logic Subsystem A	2(a)(e)(g)	1	Reactor Coolant Pressure Less Than 1700 psia(b)	(f)
	Logic Subsystem B	2(a)(e)(g)	1		
2	Steam Generator Isolation				
A	Manual	1	None	None	N/A
B	Steam Generator Isolation	1	None	None	N/A
	(i) Steam Generator Low Pressure				
	Logic Subsystem A	2/Steam Gen(a)(e)(g)	1/Steam Gen	Steam Generator Pressure Less Than 600 psia(c)	(f)
	Logic Subsystem B	2/Steam Gen(a)(e)(g)	1/Steam Gen		
	(ii) Containment High Pressure				
	Logic Subsystem A	2(a)(e)(g)	1	During Leak Test	(f)
	Logic Subsystem B	2(a)(e)(g)	1		
3	Ventilation Isolation				
A	Manual	1	None	None	N/A
B	Containment High Radiation				
	Logic Subsystem A	1(d)(g)	None	If Containment Relief and Purge Valves are Closed	(f)
	Logic Subsystem B	1(d)(g)	None		
4	Steam Generator Blowdown Isolation				
A	Manual	1(h)	None	Operating Modes 3, 4, & 5	N/A
B	Reactor Trip				
	Trains A and B	2(h)(i)	None	Operating Modes 3, 4, & 5 OR if at least one valve for each steam generator is closed	(j)

TABLE 2-4
(Continued)

a Circuits on ESF Logic Subsystems A and B each have 4 channels.

b Auto removal of bypass prior to exceeding 1700 psia.

c Auto removal of bypass prior to exceeding 600 psia.

d A and B trains are both actuated by either the Containment or Auxiliary Building Exhaust Stack initiating channels. The number of installed channels for Containment Radiation High Signal is two for purposes of Specification 2.15(1).

e If minimum operable channel conditions are reached, one inoperable channel must be placed in the tripped condition within eight hours from the time of discovery of loss of operability. Specification 2.15(2) is applicable.

f If one channel becomes inoperable, that channel must be placed in the tripped or bypassed condition within eight hours from the time of discovery of loss of operability. Specification 2.15(1) is applicable.

g Specification 2.15(3) is applicable. If ESF Logic Subsystems A and B are inoperable, enter Specification 2.0.1.

h "Minimum Operable Channels" for steam generator blowdown isolation refers to the minimum number of trains (logic subsystems) which are required to be operable to provide manual or automatic SG blowdown isolation.

i If both trains become inoperable, power operation may continue provided at least one SG blowdown isolation valve for each steam generator is closed OR be in MODE 2 within 6 hours, and in MODE 3 in the next 6 hours. Specifications 2.15(1), (2), (3) and (4) are <u>not</u> applicable; TS LCO 2.0.1 is <u>not</u> applicable.

j If one train becomes inoperable, that train may be placed in the bypassed condition. If the train is not returned to OPERABLE status within 24 hours from time of discovery of loss of operability, operation may continue as long as one SG blowdown isolation valve to each steam generator is closed. If the train is not returned to OPERABLE status within 24 hours from time of discovery, with blowdown not isolated to both SGs, be in MODE 2 within 6 hours, and in MODE 3 in the next 6 hours. Specifications 2.15(1), (2), (3) and (4) are <u>not</u> applicable; TS LCO 2.0.1 is <u>not</u> applicable.

TABLE 3-2 (continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF ENGINEERED SAFETY FEATURES, INSTRUMENTATION AND CONTROLS

Channel Description		Surveillance Function	Frequency		Surveillance Method
23. Auxiliary Feedwater	a.	Check: 1) Steam Generator Water Level Low (Wide Range) 2) Steam Generator Pressure Low	S	a.	1) CHANNEL CHECK 2) CHANNEL CHECK
	b.	Test: 1) Actuation Logic	QR$^{(7)}$	b.	1) CHANNEL FUNCTIONAL TEST
	c.	Calibrate: 1) Steam Generator Water Level Low (Wide Range) 2) Steam Generator Pressure Low 3) Steam Generator Differential Pressure High	R	c.	1) CHANNEL CALIBRATION 2) CHANNEL CALIBRATION 3) CHANNEL CALIBRATION
24. Manual Auxiliary Feedwater Actuation	a.	Test	R	a.	CHANNEL FUNCTIONAL TEST
25. Manual Steam Generator Blowdown Isolation	a.	Test	R	a.	CHANNEL FUNCTIONAL TEST
26. Automatic Steam Generator Blowdown Isolation	a.	Test	R	a.	CHANNEL FUNCTIONAL TEST

NOTES: (1) Not required unless pressurizer pressure is above 1700 psia.
(2) CRHS monitors are the containment atmosphere gaseous radiation monitor and the Auxiliary Building Exhaust Stack gaseous radiation monitor.
(3) Not required unless steam generator pressure is above 600 psia.
(4) QP - Quarterly during designated modes and prior to taking the reactor critical if not completed within the previous 92 days (not applicable to a fast trip recovery).
(5) Not required to be done on a SIT with inoperable level and/or pressure instrumentation.
(6) Not required when outside ambient air temperature is greater than 50°F and less than 105°F.
(7) Tests backup channels such as derived circuits and equipment that cannot be tested when the plant is at power.
(8) SGLS is required for containment spray pump actuation only. SGLS lockout relays are not actuated for this test.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 263 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 letter dated January 30, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090340536), as supplemented by letters dated June 30 and August 28, 2009 (ADAMS Accession Nos. ML091830041 and ML092440295, respectively), Omaha Public Power District, (OPPD, the licensee) submitted a license amendment request (LAR) regarding the Fort Calhoun Station (FCS), Unit No. 1, to add steam generator (SG) blowdown isolation requirements to the Technical Specifications (TSs). The supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 7, 2009 (74 FR 15774).

The current Updated Safety Analysis Report (USAR) accident analysis for the loss of main feedwater (LMFW event) credits manual action for SG blowdown isolation within 15 minutes of an LMFW event. OPPD plans to install a modification to the SG blowdown isolation valves so that they will automatically close during an LMFW event. Automatic isolation will ensure that the 15-minute requirement is met without the risk that an unanticipated distraction could prevent manual action from occurring at the proper time.

In addition, the licensee proposed to modify the remedial actions for channels that currently contain a key-operated bypass switch. The current remedial actions allow for bypassing of these channels under certain conditions. The proposed amendment would allow for a bypass as well if the channel bypass switch was not key-operated.

2.0 REGULATORY EVALUATION

In Section 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR), "Technical specifications," the NRC established its regulatory requirements related to the content of TSs.

Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. The rule does not specify the particular requirements to be included in a plant's TS.

As stated in 10 CFR 50.36(c)(2)(i), the "[l]imiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When 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."

Criterion 3 of 10 CFR 50.36(c)(2)(ii) states that a TS LCO of a nuclear reactor must be established for "[a] structure, system, or component 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 regulations in 10 CFR 50.36(c)(3) state that "[s]urveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

In a memorandum dated September 18, 1992, the Commission approved the NRC staff's proposal in SECY-92-223, "Resolution of Deviations Identified During the Systematic Evaluation Program," not to apply 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," to plants with construction permits prior to May 21, 1971 (ADAMS Legacy Library Accession No. 9210060362). FCS was licensed for construction prior to May 21, 1971, and at that time committed to the draft General Design Criteria (GDC). The draft GDC, which are similar to Appendix A, "General Design Criteria for Nuclear Power Plants," in 10 CFR Part 50, are contained in Appendix G, "Response to 70 Criteria," of the FCS Updated Safety Analysis Report (USAR).

In its letter dated January 30, 2009, the licensee identified the following draft GDC as specified in Appendix G to the FCS USAR applied to this review:

- CRITERION 14 – CORE PROTECTION SYSTEMS states: *Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.*

- CRITERION 15 – ENGINEERED SAFETY FEATURES PROTECTION SYSTEM states: *Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.*

- CRITERION 19 – PROTECTION SYSTEMS RELIABILITY states: *Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.*

- CRITERION 20 – PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE states: *Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include as a minimum, two channels of protection for each protection function to be served. Different principles shall be used where necessary to achieve true independence of redundant instrumentation components.*

- CRITERION 21 – SINGLE FAILURE DEFINITION states: *Multiple failures resulting from a single event shall be treated as single failure.*

- CRITERION 22 – SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS states: *Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation and protection circuitry leaves intact a system satisfying all requirements for the protection channels.*

- CRITERION 23 – PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS states: *The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of protection function.*

- CRITERION 25 – DEMONSTRATION OF FUNCTION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEM states: *Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.*

- CRITERION 26 – PROTECTION SYSTEMS FAIL-SAFE DESIGN states: *The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.*

- CRITERION 53 – CONTAINMENT ISOLATION VALVES states: *Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.*

3.0 TECHNICAL EVALUATION

3.1 System Description

In its letter dated January 30, 2009, the licensee states, in part, that:

> The current USAR, Chapter 14, accident analysis for the LMFW credits manual action for isolation within 15 minutes of an LMFW event. This 15-minute manual action was requested by OPPD [by letter September 16, 2005] and subsequently approved by the Nuclear Regulatory Commission [by Amendment No. 242, ADAMS No. ML062280476]. Automatic isolation will ensure that the 15-minute requirement is met without the risk that an unanticipated distraction could prevent manual action from occurring at the proper time. The installation of this automatic feature eliminates the need for manual isolation of SG blowdown and thus will eliminate the associated operator challenge. The manual isolation capability is being maintained and the operator's ability to manually isolate SG blowdown remains unchanged by this activity.

In its supplement letter dated June 30, 2009, the licensee states, in part, that:

(a) The steam generator (SG) blowdown system performs no safety-related function. The SG blowdown system is used to:

- Maintain SG water chemistry by removal of impurities through bottom blowdown.
- Provides a sample to monitor SG chemistry and potential primary-to-secondary leakage.
- Recirculate SG water or transfer SG water from one generator to the other.
- Drain the SGs for dry layup.

...the SG blowdown isolation valves operate as follows: There are two signals that initiate a SG blowdown isolation. The first is a SG blowdown high radiation level (indicative of primary-to-secondary SG tube leakage). The second is a containment isolation actuation signal (CIAS). It is important to note that the signal being added (reactor trip) by this proposed modification is to provide an automatic means to isolate SG blowdown on a Loss of Main Feedwater (LMFW) event. The other two isolation features of the valves remain unchanged by this modification.

(b) Isolation of SG blowdown (including spurious isolation) has no safety consequences nor impacts on other TS. While not desirable from an operations standpoint (due to SG chemistry constraints), there is no safety significance to operating with SG blowdown isolated. SG blowdown has, at times, been isolated while the plant is at full power to facilitate performance of certain maintenance activities. FCS follows Nuclear Energy Institute (NEI) 97-06 R2, "Steam Generator Program Guidelines," and Electric Power Research Institute (EPRI) TR-10234, "PWR Secondary Water Chemistry Guidelines." Extended periods of SG

> blowdown isolation may result in exceeding action levels of EPRI TR 10234, which could result in a plant down-power. Sampling lines for SG blowdown lines are on the SG side of the "A" (i.e., inboard) blowdown isolation valves. This ensures the ability to perform required TS surveillances for TS 3.2, Table 3-4 and TS 5.13 with the isolation valves maintained closed. There are no other TS affected by this change.

In its letter dated January 30, 2009, the licensee states that it plans to install a modification to the SG blowdown isolation valves to automatically close the valves during an LMFW event. This modification will add automatic SG blowdown isolation following an LMFW by adding an interlock, which isolates SG blowdown following a reactor trip, to the control circuits for SG blowdown isolation valves HCV-1387A, HCV-1388A, HCV-1387B, and HCV-1388B. Because this is a new automatic feature, the current TS contain no provision for testing of the interlock nor do the TS contain operability requirements for the interlock.

3.2 Proposed Technical Specification Changes

In its letter dated January 30, 2009, OPPD requested the following TS changes:

- TS 2.15(1) is being administratively changed to remove the words "key operated." This is an administrative change as the "key" associated with bypass switches is not a critical element in controlling the use of bypass switches. The position of bypass switches is controlled through procedural guidance and the key is only necessary to position the switch. The key operation of the bypass switches does not impact the actual TS action for this LCO which remains the same with this proposed change.

- TS 2.15, Table 2-4 - Items 4A and 4B are being added to include the operability requirements for SG blowdown isolation on a reactor trip.

- TS 2.15, Table 2-4 - Footnote (h) is added to document that the minimum operable channels for steam generator blowdown isolation refers to the minimum number of trains (logic subsystems) which are installed to provide automatic SG blowdown isolation. This is being done to provide clarification since TS 2.15, for the most part, refers to initiation channels, whereas SG blowdown isolation refers to actuation logic trains.

- TS 2.15, Table 2-4 - Footnote (i) is added to document that with both trains A and B inoperable, power operation may continue provided at least one SG blowdown isolation valve for each SG is closed OR be in MODE 2 within 6 hours, and in MODE 3 in the next 6 hours. TS LCOs 2.15(3), 2.15(4), and 2.0.1 are not applicable. (This proposed change is aligned with the current TS LCO 2.5(1)C. for AFW operability.)

- TS 2.15, Table 2-4 - Footnote (j) is added to document that if one train becomes inoperable, that train may be placed in the bypassed condition. If the train is not returned to OPERABLE status within 24 hours from time of discovery of loss of operability, power operation may continue as long as one SG blowdown isolation valve to each SG is closed. If the train is not returned to OPERABLE status within 24 hours from time of discovery, with blowdown not isolated to both SGs, be in MODE 2 within 6 hours, and in MODE 3 in the next 6 hours. TS 2.15(1) and 2.0.1 are not applicable. (This proposed change is aligned with the current TS LCO 2.5(1)C. for AFW operability.)

- TS 3.1, Table 3-2 - Items 25 and 26 are added to include surveillance testing requirements for SG blowdown isolation on a reactor trip. Testing (CHANNEL FUNCTIONAL TEST) to ensure operability of the interlock will be performed on a refueling outage frequency, consistent with the current testing frequency for other interlocks that perform similar type functions.

In a teleconference on August 13, 2009, the NRC staff expressed concern with the proposed wording of footnote (h). Proposed footnote (h) provides information on the logic subsystem is designated as associated with TS 2.15, Table 2-4, Item 4B, *Reactor Trip,* which is the automatic portion of the actuation circuit, but was not designated as associated with TS 2.15, Table 2-4, Item 4A, which is the manual portion of the actuation circuit. In addition, the NRC staff noted that since TS 2.15, Table 2-4, footnotes (i) and (j) do not state that TS 2.15(1) through 2.15(4) do not apply, in some situations it could be concluded that both the TS 2.15 guidance and the footnotes would apply and contain different actions for the same condition. Therefore, by letter dated August 28, 2009, the licensee revised footnote (h) to apply to the manual actuation train in addition to the automatic actuation train, and revised footnotes (i) and (j) to specifically state that TS 2.15(1) through 2.15(4) do not apply.

3.3 NRC Staff Evaluation

The proposed changes will allow FCS to credit automatic SG blowdown isolation and will eliminate the need for manual operator action to isolate SG blowdown. Currently, manual operator action is credited for this isolation feature. The licensee plans to add a new interlock feature during the 2009 refueling outage that will provide automatic SG blowdown following a reactor trip and eliminate the need for manual operator action.

The proposed amendment adds Function 4, "Steam Generator Blowdown Isolation," to TS Table 2-4, "Instrument Operating Conditions for Isolation Functions." Function 4 will isolate on either a reactor trip or manual isolation. For the LMFW event, automatic isolation of the SG blowdown isolation is assumed to occur within 15 minutes. The analysis assumes a reactor trip will cause a SG blowdown isolation in Modes 1 and 2. Therefore, TS Table 2-4, Function 4 is required to be retained in the TSs per Criterion 3 of 10 CFR 50.36(c)(2)(ii). A manual isolation switch is included as a backup.

SG water level will lower during a LMFW event. Since a reactor trip is initiated by a SG low level signal logic, the NRC staff concludes that a reactor trip signal is an acceptable actuation signal for SG blowdown isolation. In addition, the SG blowdown isolation is assumed to occur within 15 minutes of a LMFW event. Due to the large time period involved, a SG blowdown isolation that occurs directly from an SG low level signal is not needed. As a backup, operators can manually isolate the SG blowdown line if needed. The NRC staff concludes that this procedure is acceptable.

The LMFW event assumes that the initial power level is at 100 percent rated power in order to bound the worst case scenario and be applicable at any power level. As a result, TS Table 2-4 Function 4 is only required to be operable in Mode 1 (Power Operating Condition) and Mode 2 (Hot Standby Condition).

While in Mode 3 (Hot Shutdown Condition), the SGs are still used to remove decay heat from the core. Although the main feedwater pumps are not utilized, a loss-of-feed event can still occur if the auxiliary feedwater pumps are lost. As a result, the need for SG blowdown isolation in Mode 3 was considered by the licensee. If auxiliary feedwater is lost in while in Mode 3, operators have a minimum of 1 hour to restore auxiliary feedwater. The licensee's analysis assumes that the SG blowdown line is not isolated. Due to the time period involved, and the analysis that assumes that the SG blowdown line is not isolated, TS Table 2-4, Function 4 is not required to be operable in Mode 3. In all other Modes, such as Mode 4 (Cold Shutdown Condition) and Mode 5 (Refueling Shutdown Condition), the SG is not utilized for decay heat removal. The NRC concludes that TS Table 2-4, Function 4 is not required to be operable in Modes 4 and Mode 5 and, therefore, the proposed change is acceptable.

Events other than an LMFW event could cause a reactor trip. As a result, the licensee states that the SG blowdown line may be isolated even when it may not need to be isolated. The SG blowdown line can be unisolated in such conditions via the use of an override switch. The override switch only overrides the reactor trip signal. While in an applicable Mode, TS Table 2-4, Function 4 is declared inoperable any time that the SG blowdown line isolation is in override.

SG blowdown line isolation is required if high radiation levels are detected in the line or if there is a signal to isolate containment. Function 1, "Containment Isolation," of TS Table 2-4, currently contains the TS requirements to isolate the SG blowdown line on a containment isolation signal. The high radiation isolation signal is not directly cited in the TSs, but it is controlled via the Offsite Dose Calculation Manual (ODCM) of TS 5.17. The proposed amendment is not seeking to remove the containment isolation function or the high radiation isolation function of the SG blowdown line or change the current TS requirements associated with these two isolation methods. The NRC staff agrees that the proposed change adds a third isolation method under a new TS function and therefore concludes it is acceptable.

Two trains for isolating the SG blowdown line on a reactor trip are required for redundancy. One train can isolate the blowdown lines on both SGs. If one train becomes inoperable, that train may be placed in the bypassed condition. Since one train can carry out the safety function, operation may continue for up to 24 hours to facilitate repairs. The time period is limited to 24 hours since redundancy is lost. After 24 hours, if the train is still inoperable, operation may continue as long as both SG blowdown lines are isolated. Continued operation in this state is acceptable since the safety function of TS Table 2-4, Function 4 is to isolate the SG blowdown lines. If after 24 hours, the train is still inoperable and both SG blowdown lines are not isolated, operators must exit the Mode of applicability.

If both trains become inoperable, power operation may continue as long as both SG blowdown lines are isolated. Continued operation in this state is acceptable since the safety function of TS Table 2-4, Function 4 is to isolate the SG blowdown lines. If both SG blowdown lines are not isolated, operators must exit the Mode of applicability.

While not desirable from an operations standpoint, due to SG chemistry constraints, there is no safety significance to operating with SG blowdown lines isolated. In its letter dated June 30, 2009, the licensee states, in part, that:

> FCS follows Nuclear Energy Institute (NEI) 97-06, Revision 2, "Steam Generator Program Guidelines," and Electric Power Research Institute (EPRI) Topical Report (TR)-10234, "PWR Secondary Water Chemistry Guidelines." Extended periods of SG blowdown isolation may result in exceeding action levels of EPRI TR 10234, which could result in a plant down-power.

As a result, the NRC staff concludes that there would be an incentive to repair the SG blowdown line isolations prior to requiring a plant down-power. SG sampling can still continue with the SG blowdown lines isolated since the sampling point is upstream of the isolation.

In its letters dated June 30, 2009 and August 29, 2009, the licensee states that TS 2.15(1), TS 2.15(2), TS 2.15(3), and 2.15(4) are not applicable to the reactor trip trains of TS Table 2-4, Function 4. The actions of TS 2.15(1), TS 2.15(2), TS 2.15(3), and 2.15(4) include reactor shutdown and cooldown. The NRC staff concludes that these actions do need to apply since the actions for the reactor trip trains of TS Table 2-4, Function 4 require carrying out the safety function or exiting the Mode of applicability for any discovered inoperabilities and, therefore, is acceptable.

TS 2.0.1(1), 2.0.1(2), and 2.0.1(3) are not applicable to the reactor trip trains of TS Table 2-4, Function 4. TS 2.0.1(1) contains actions in the event an LCO and/or associated action requirement cannot be satisfied because of circumstances in excess of those addressed in the specification. Actions of TS 2.0.1(1) include reactor shutdown and cooldown. The actions of TS 2.0.1(1) do need to apply since the actions for the reactor trip trains of TS Table 2-4, Function 4 require carrying out the safety function or exiting the Mode of applicability for any discovered inoperabilities. TS 2.0.1(2) contains provisions on operability associated with equipment that requires power to carry out the safety function. The actions of TS 2.0.1(2) do not need to apply because power failure results in SG blowdown circuitry going to the fail-safe isolation position, thereby maintaining the valves closed and fulfilling the safety function. TS 2.0.1(3) applies to snubbers and is not applicable to the SG blowdown isolation function. Based on the above, the NRC staff concludes the proposed changes are acceptable.

As a backup, there are two trains for manual isolation of both SG blowdown lines. TS 2.15(1), TS 2.15(2), TS 2.15(3), and 2.15(4) contain actions if one or both manual channels are inoperable. The NRC staff concludes that the actions are acceptable since the actions contain requirements that can include reactor shutdown and cooldown, and are more conservative than simply carrying out the safety function or exiting the Mode of applicability.

TS Table 3-2 contains a proposed SR for the new SG blowdown isolation function. The licensee proposed to conduct a channel functional test every 18 months for the manual and automatic isolation trains. Since the fail-safe position of the SG blowdown isolation function is to isolate, the NRC staff concludes that 18 months is an acceptable testing frequency. In addition, the proposed SRs assure that the necessary quality of TS Table 2-4, Function 4 is maintained, that facility operation will be within safety limits, and that the LCO will be met. Therefore, 10 CFR 50.36(c)(3) continues to be met.

Testing of the reactor protection system (RPS) portion of the blowdown isolation circuitry is addressed in existing TS Table 3-1, which contains numerous requirements for channel calibrations, channel checks, and channel functional tests. The proposed SR in TS Table 3-2 is focused on testing only the portion of the isolation function from the RPS output relays to the isolation valve control circuits. Since there are no instrument sensors associated with this portion of the isolation circuitry, there is no need for channel checks or calibrations. Since the USAR analysis requires that SG blowdown isolation occur within 15 minutes for a LMFW event, there is no need to conduct a response time test. Response time tests are typically used when an analysis requires an actuation within a very short time frame. The channel functional test will provide enough information as to ascertain whether or not the trip will occur within the assumed timeframe; therefore, the NRC staff concludes that it is acceptable.

In addition, TS 2.15(1) contains remedial actions when a channel in TS Tables 2-2 through 2-5 is inoperable. If the number of channels in a particular system falls one below the total number of installed channels, the inoperable channel shall be placed in either the bypassed or tripped condition within 1 hour if the channel is equipped with a key-operated bypass switch, and 8 hours if jumpers or blocks must be installed in the control circuitry. The proposed amendment would delete the term "key-operated" so that channels with a bypass switch that are not key-operated may be bypassed as well. The NRC staff concludes this change is acceptable since an installed bypass switch can be properly controlled via the use of plant procedures.

Based on the above, the NRC staff concludes that the stated changes continue to meet the requirements of 10 CFR 50.36 as well as the requirements of draft GDC as specified in Appendix G to the FCS USAR and, therefore, concludes that the proposed changes are acceptable. The NRC has no objection to the conforming changes to the TS Basis.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official was notified 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 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 and there has been no public comment on such finding published in the *Federal Register* on April 7, 2009 (74 FR 15774). 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) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. Lewin

Date: October 9, 2009

October 9, 2009

Mr. David J. Bannister
Vice President and CNO
Omaha Public Power District
Fort Calhoun Station
444 South 16th St. Mall
Omaha, NE 68102-2247

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: REQUEST TO ADD STEAM GENERATOR BLOWDOWN ISOLATION REQUIREMENTS TO TECHNICAL SPECIFICATIONS (TAC NO. ME0596)

Dear Mr. Bannister:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 263 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 January 30, 2009, as supplemented by letters dated June 30 and August 28, 2009.

The amendment adds steam generator blowdown isolation operability and surveillance requirements to the TSs. These requirements will be needed as a result of a plant modification to add an interlock to automatically isolate steam generator blowdown following a reactor trip. In addition, the amendment modifies the remedial actions for channels that currently contain a key-operated bypass switch by deleting the term "key-operated" so that channels with a bypass switch that are not key-operated may be bypassed as well.

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 by Nicholas J. DiFrancesco for/
Lynnea E. Wilkins, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285
Enclosures:
1. Amendment No. 263 to DPR-40
2. Safety Evaluation
cc w/encls: Distribution via Listserv

DISTRIBUTION:
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A. Lewin, NRR/DIRS/ITSB

ADAMS Accession No. ML092540591 *9/9/09 SE memo

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	NRR/DIRS/ITSB/BC	NRR/DSS/SRXB/BC	NRR/DE/EICB/BC
NAME	LWilkins	JBurkhardt	RElliott*	GCranston*	WKemper*
DATE	9/15/09	9/15/09	9/9/09	9/9/09	9/9/09

OFFICE	OGC	NRR/LPL4/BC	NRR/LPL4/PM
NAME	JSuttenberg	MMarkley	LWilkins NDiFrancesco for
DATE	9/21/09	10/9/09	10/9/09

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