July 25, 2000

Mr. S. K. Gambhir
Division Manager - Nuclear Operations
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
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Fort Calhoun, Nebraska 68023-0399

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - AMENDMENT NO. 194 TO FACILITY OPERATING LICENSE NO. DPR-40 (TAC NO. MA0610)

Dear Mr. Gambhir:

The Commission has issued the enclosed Amendment No. 194 to Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated December 11, 1997, as supplemented by letter dated May 8, 2000.

The amendment revises the TS by adding a new Limiting Condition for Operation (LCO) for an inoperable engineered safety features logic subsystem. In addition, administrative changes are made to either support the new LCO or clarify existing text.

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

Sincerely,
/RA/

L. Raynard Wharton, Project Manager, Section 4
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures: 1. Amendment No. 194 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

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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 FACILITY OPERATING LICENSE

Amendment No. 194
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 December 11, 1997, as supplemented by letter dated May 8, 2000, 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, Facility Operating License No. DPR-40 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of 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. 194, 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 within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen Dembek

Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: July 25, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 194

FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Revise Appendix "A" Technical Specifications as indicated below. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
4	4
2-65	2-65
2-66	2-66
2-66a	2-66a
2-66b	2-66b
---	2-66c
2-67	2-67
2-67a	2-67a
2-68	2-68
2-68a	2-68a
2-68b	2-68b
2-69	2-69
2-69a	2-69a

PROTECTIVE SYSTEMS (Continued)

Engineered Safety Feature Logic[2]

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

Degree of Redundancy

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

INSTRUMENTATION SURVEILLANCE

Channel Check

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

Channel Functional Test

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

Channel Calibration

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

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 key operated 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. If after 24 hours from time of

initiating a hot shutdown procedure at least one inoperable engineered safety features or isolation functions channel has not been restored to OPERABLE status, the reactor shall be placed in a cold shutdown condition within the following 24 hours. 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.

(3) In the event the number of channels on a particular engineered safety features (ESF) or isolation logic subsystem in service falls below the limits given in the columns entitled "Minimum Operable Channels" or "Minimum Degree of Redundancy," except as conditioned by the column entitled "Permissible Bypass Conditions," sufficient channels shall be restored to OPERABLE status within 48 hours so as to meet the minimum limits or 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 ventilation isolation valves are closed. If after 24 hours from time of initiating a hot shutdown procedure sufficient channels have not been restored to OPERABLE status, the reactor shall be placed in a cold shutdown condition within the following 24 hours.

(4) In the event the number of channels of those particular systems in service not described in (3) above falls below the limits given in the columns entitled "Minimum Operable Channels" or "Minimum Degree of Redundancy," except as conditioned by the column entitled "Permissible Bypass Conditions," the reactor shall be placed in a hot shutdown condition within 12 hours. If minimum conditions for engineered safety features or isolation functions are not met within 24 hours from time of discovering loss of operability, the reactor shall be placed in a cold shutdown condition within the following 24 hours. If the number of OPERABLE high rate trip-wide range log channels falls below that given in the column entitled "Minimum Operable Channels" in Table 2-2 and the reactor is at or above 10^{-4}% power and at or below 15% of rated power, reactor critical operation shall be discontinued and the plant placed in an operational mode allowing repair of the inoperable channels before startup or reactor critical operation may proceed.

If, during power operation, the rod block function of the secondary CEA position indication system and rod block circuit are inoperable for more than 24 hours, or the plant computer PDIL alarm, CEA group deviation alarm and the CEA sequencing function are inoperable for more than 48 hours, the CEAs shall be withdrawn and maintained at fully withdrawn and the control rod drive system mode switch shall be maintained in the off position except when manual motion of CEA Group 4 is required to control axial power distribution.

(5) In the event that any of the following Alternate Shutdown Panel instrumentation or control circuits become inoperable, either restore the inoperable component(s) to OPERABLE status within seven days, or be in hot shutdown within the next twelve hours. This specification is applicable in Modes 1 and 2.

Wide Range Logarithmic Power (AI-212)
Source Range Power (AI-212)
Reactor Coolant Cold Leg Temperature (AI-185)
Reactor Coolant Hot Leg Temperature (AI-185)
Pressurizer Level (AI-185)
Volume Control Tank Level (AI-185)

(6) In the event that any of the following Emergency Auxiliary Feedwater Panel instrumentation or control circuits become inoperable, either restore the inoperable component(s) to OPERABLE status within seven days, or be in hot shutdown within the next twelve hours. This specification is applicable in Modes 1 and 2.

Steam Generator Level, Wide Range (AI-179)
Steam Generator Level, Narrow Range (AI-179)
Steam Generator Pressure (AI-179)
Pressurizer Pressure (AI-179)

Basis

During plant operation, the complete instrumentation systems will normally be in service. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor protective system (RPS) and engineered safety features (ESF) system when one or more of the channels are out of service. Reactor safety is provided by the RPS, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continued operation with certain instrumentation channels out of service since provisions were made for this in the plant design.

The RPS and most engineered safety feature channels are supplied with sufficient redundancy to provide the capability for channel test at power, except for backup channels such as derived circuits in the ESF logic system.

When one of the four channels is taken out of service for maintenance, RPS logic can be changed to a two-out-of-three coincidence for a reactor trip by bypassing the removed channel. If the bypass is not effected, the out-of-service channel (Power Removed) assumes a tripped condition (except high rate-of-change of power, high power level and high pressurizer pressure),[1] which results in a one-out-of-three channel logic. If in the 2-out-of-4 logic system of the RPS one channel is bypassed and a second channel manually placed in a tripped condition, the resulting logic is 1-out-of-2. At rated power, the minimum OPERABLE high-power level channel is 3 in order to provide adequate power tilt detection. If only 2 channels are OPERABLE, the reactor power level is reduced to 70% rated power which protects the reactor from possibly exceeding design peaking factors due to undetected flux tilts and from exceeding dropped CEA peaking factors.

The ESF logic system is a Class 1 protection system designed to satisfy the criteria of IEEE 279, August 1968. Two functionally redundant ESF logic subsystems "A" and "B" are provided to ensure high reliability and effective in-service testing. These logic subsystems are designed for individual reliability and maximum attainable mutual independence both physically and electrically. Either logic subsystem acting alone can automatically actuate engineered safety features and essential supporting systems.

All engineered safety features are initiated by 2-out-of-4 logic matrices except containment high radiation which operates on a 1-out-of-2 basis. The number of installed channels for Containment Radiation High Signal (CRHS) is two. CRHS isolates the containment pressure relief, air sample and purge system valves.

Basis (Continued).

Entry into Technical Specification 2.15(3) is made when conditions have caused one logic subsystem ("A" or "B") to become inoperable but the redundant logic subsystem remains operable. The loss of a prime initiation relay (which renders all 4 channels of a logic subsystem inoperable) is the condition most likely to cause entry into Technical Specification 2.15(3). In this situation, the remaining ESF logic subsystem still has the capability to automatically actuate engineered safety features equipment and essential supporting systems. The 48-hour completion time is commensurate with the importance of avoiding the vulnerability of a single failure in the remaining ESF logic subsystem. Technical Specification 2.15(3) will not be used upon loss of the common channels that affect both "A" and "B" subsystems prime initiators operability. Upon exiting TS 2.15(3) following the restoration of a prime initiation relay to OPERABLE status, if any channel(s) remain inoperable, the appropriate LCO (TS 2.15(1) or (2)) is applicable with the length of inoperability measured from time of discovery of: 1) prime initiation relay inoperable, or 2) channel inoperability, whichever is longer.

The ESF logic system provides a 2-out-of-4 logic on the signals used to actuate the equipment connected to each of the two emergency diesel generator units.

The rod block system automatically inhibits all CEA motion in the event a Limiting Condition for Operation (LCO) on CEA insertion, CEA deviation, CEA overlap or CEA sequencing is approached. The installation of the rod block system ensures that no single failure in the control element drive control system (other than a dropped CEA) can cause the CEAs to move such that the CEA insertion, deviation, sequencing or overlap limits are exceeded. Accordingly, with the rod block system installed, only the dropped CEA event is considered an AOO and factored into the derivation of the Limiting Safety System Settings and Limiting Conditions for Operation. With the rod block function out-of-service several additional CEA deviation events must be considered as AOOs. Analysis of these incidents indicates that the single CEA withdrawal incident is the most limiting of these events. An analysis of the at-power single CEA withdrawal incident was performed for Fort Calhoun for various initial Group 4 insertions, and it has been concluded that the Limiting Conditions for Operation (LCO) and Limiting Safety System Settings (LSSS) are valid for a Group 4 insertion of less than or equal to 15%.

The operability of the Alternate Shutdown Panel (AI-185), including Wide Range Logarithmic Power and Source Range Monitors on AI-212, and Emergency Auxiliary Feedwater Panel (AI-179) instrument and control circuits ensures that sufficient capability is available to permit entry into and maintenance of the Hot Shutdown Mode from locations outside of the Control Room. This capability is required in the event that Control Room habitability is lost due to fire in the cable spreading room or Control Room.

Variances which may exist at startup between the more accurate ΔT-Power and Nuclear Instrumentation Power (NI-Power) are not significant for enabling of the trip functions. By 15% of rated power as measured by the uncalibrated NI Power, the Axial Power Distribution (APD) and Loss of Load (LOL) trip functions are enabled while the High Rate of Change of Power trip is bypassed.

Basis (Continued).

The APD trip function acts to limit the axial power shape to the range assumed in the setpoint analysis. Significant margins to local power density limits exist at 15% power, as well as power levels up to at least 30% (where NI calibration occurs).

The LOL trip function acts as an anticipatory trip for the high pressurizer pressure and high power trips in order to limit the severity of a LOL transient. This trip is not credited in the USAR Chapter 14 Safety Analyses and any variance between ΔT-Power and NI-Power has no effect on the safety analysis.

The High Rate of Change of Power trip acts to limit power excursions from low power levels and bypassing of this trip at a high power level is conservative. This trip is not credited in the USAR Chapter 14 Safety Analyses for Mode 1 operation. Any variance between ΔT-Power and NI-Power has no effect on the safety analysis.

References

(1) USAR, Section 7.2.7.1

TABLE 2-2

Instrument Operating Requirements for Reactor Protective System

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
1	Manual (Trip Buttons)	1	None	None	N/A
2	High Power Level	2[b][c]	1[c]	Thermal Power Input Bypassed below 10^{-4}% of Rated Power[a][d]	(e)
3	Thermal Margin/Low Pressurizer Pressure	2[b]	1	Below 10^{-4}% of Rated Power[a][d]	(e)
4	High Pressurizer Pressure	2[b]	1	None	(e)
5	Low R.C. Flow	2[b]	1	Below 10^{-4}% of Rated Power [a][d]	(e)
6	Low Steam Generator Water Level	2/Steam Gen[b]	1/Steam Gen	None	(e)
7	Low Steam Generator Pressure	2/Steam Gen[b]	1/Steam Gen	Below 600 psia[a][d]	(e)
8	Containment High Pressure	2[b]	1	During Leak Test	(e)
9	Axial Power Distribution	2[b][c]	1[c]	Below 15% of Rated Power[g]	(e)
10	High Rate Trip-wide Range Log Channels	2[b]	1	Below 10^{-4}% and above 15% of Rated Power[a][g]	(e)
11	Loss of Load	2[b]	1	Below 15% of Rated Power[g]	(e)
12	Steam Generator Differential Pressure	2[b]	1	None	(e)

a. Bypass automatically removed.

b. Specification 2.15(2) is applicable.

c. If two channels are inoperable, load shall be reduced to 70% or less of rated power.

d. For low power physics testing this trip may be bypassed up to 10^{-1}% of rated power.

e. Specification 2.15(1) is applicable.

f. Deleted.

g. For each channel, the same bistable automatically activates the Loss of Load and Axial Power Distribution (APD) trips and automatically bypasses the high rate trip at 15% of rated power. Only the APD trip is a Limiting Safety System Setting. Therefore, the bistable is set to actuate within the APD tolerance band.

TABLE 2-3

Instrument Operating Requirements for Engineered Safety Features

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
1	Safety Injection				
A	Manual	1	None	None	N/A
B	High Containment Pressure				
	Logic Subsystem A	2[a][d][l]	1	During Leak Test	(f)
	Logic Subsystem B	2[a][d][l]	1		
C	Pressurizer Low/Low Pressure				
	Logic Subsystem A	2[a][d][l]	1	Reactor Coolant Pressure Less Than 1700 psia[b]	(f)
	Logic Subsystem B	2[a][d][l]	1		
2	Containment Spray				
A	Manual	1	None	None	N/A
B	High Containment Pressure				
	Logic Subsystem A	2[a][c][d][l]	1	During Leak Test	(f)
	Logic Subsystem B	2[a][c][d][l]	1		
C	Pressurizer Low/Low Pressure				
	Logic Subsystem A	2[a][c][d][l]	1	Reactor Coolant Pressure Less Than 1700 psia[b]	(f)
	Logic Subsystem B	2[a][c][d][l]	1		
3	Recirculation				
A	Manual	1	None	None	N/A
B	SIRW Tank Low Level				
	Logic Subsystem A	2[a][k][l]	1	None	(j)
	Logic Subsystem B	2[a][k][l]	1		
4	Emergency Off-Site Power Trip				
A	Manual	1[e]	None	None	N/A
B	Emergency Bus Low Voltage (Each Bus)				
	-Loss of Voltage	2[d]	1	Reactor Coolant Temperature Less Than 300°F	(f)
	-Degraded Voltage	2[a][d]	1		

TABLE 2-3
(Continued)

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
5	Auxiliary Feedwater				
A	Manual	1	None	None	N/A
B	Auto Initiation Logic Subsystem A Logic Subsystem B			Operating Modes 3, 4, and 5	
	-Steam Generator Low Level	2[a][d][l]	1		(h)
	-Steam Generator Low Pressure	3[a][g][l]	1		(i)
	-Steam Generator Differential Pressure	3[a][g][l]	1		(i)

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

b Auto removal of bypass above 1700 psia.

c Coincident high containment pressure and pressurizer pressure low signals required for initiation of containment spray.

d If minimum OPERABLE channel conditions are reached, one inoperable channel must be placed in the tripped condition or low level actuation position for auxiliary feedwater system within eight hours from the time of discovery of loss of operability. Specification 2.15(2) is applicable.

e Control switch on incoming breaker.

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

g Three channels required because bypass or failure results in auxiliary feedwater actuation block in the affected channel.

h Specification 2.15(1) is applicable.

TABLE 2-3
(Continued)

i If the channel becomes inoperable, that channel must be placed in the bypassed condition within eight hours from time of discovery of loss of operability. If the channel is not returned to OPERABLE status within 48 hours from time of discovery of loss of operability, one of the eight channels may continue to be placed in the bypassed condition provided the Plant Review Committee has reviewed and documented the judgment concerning prolonged operation in bypass of the inoperable channel. The channel shall be returned to OPERABLE status no later than during the next cold shutdown. If one of the four channels on one steam generator is in prolonged bypass and a channel on the other steam generator becomes inoperable, the second inoperable channel must be placed in bypass within eight hours from time of discovery of loss of operability. If one of the inoperable channels is not returned to OPERABLE status within seven days from the time of discovery of the second loss of operability, the unit must be placed in hot shutdown within the following 12 hours.

j If one channel becomes inoperable, that channel must be placed in the bypassed condition within eight hours from time of discovery of loss of operability. If the channel is not returned to OPERABLE status within 48 hours from time of discovery of loss of operability, one of the eight channels may continue to be placed in the bypassed condition provided the Plant Review Committee has reviewed and documented the judgment concerning prolonged operation in bypass of the inoperable channel. The channel shall be returned to OPERABLE status no later than during the next cold shutdown. If a channel is in prolonged bypass and a channel on the opposite train becomes inoperable, the second inoperable channel must be placed in bypass within eight hours from time of discovery of loss of operability. If one of the inoperable channels is not returned to OPERABLE status within seven days from the time of discovery of the second loss of operability, the unit must be placed in hot shutdown within the following 12 hours.

k Specification 2.15(2) is applicable.

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

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[e][g]	1/Steam Gen	Steam Generator Pressure Less Than 600 psia[c]	(f)
	Logic Subsystem B	2/Steam Gen[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		

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.

TABLE 2-4
(Continued)

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 194 TO 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 December 11, 1997, as supplemented by letter dated May 8, 2000, Omaha Public Power District (OPPD) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. DPR-40) for the Fort Calhoun Station, Unit No. 1. The requested changes would add a new Limiting Condition for Operation (LCO) for an inoperable engineered safety features logic subsystem. In addition, administrative changes would be made to either support the new LCO or clarify existing text.

The May 8, 2000, supplemental letter provided additional clarifying information, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination published in the Federal Register on February 11, 1998 (63 FR 6987).

2.0 EVALUATION

The engineered safety features (ESF) at the Fort Calhoun Station function to cool the core, limit the magnitude and duration of the pressure transient within the containment vessel following a loss-of-coolant accident and provide long-term post-accident cooling. The ESF control and instrumentation system actuates ESF equipment automatically, based upon the values of selected unit parameters, to achieve the functions of the ESF.

The ESF instrumentation and control system contains two independent, functionally redundant logic subsystems, which act independently of each other and receive input from different sets of instrumentation. ESF instrumentation is arranged into four independent channels. The two-out-of-four logic (one-out-of-two logic for the containment radiation high signal (CRHS)) of each ESF logic subsystem receives input from the instrumentation of all four channels (two channels for the CRHS). When a measured process variable departs from its normal range, the prime initiation relay on that ESF logic subsystem trips and sends a derived signal to trip the opposite ESF logic subsystem. This design feature provides for improved system reliability as prime and derived initiation signals are each capable of independently generating the required actuation signal resulting in ESF equipment actuation from either or both ESF logic subsystems. ESF equipment actuation results from the logical combination of initiating signals, each of which is derived from a departure from the normal operating range of one of the critical parameters.

The failure of one or more prime initiation lockout relays on one of the ESF logic subsystems would prevent all of the channels associated with the failed lockout relay(s) from providing a trip signal to the prime actuation, derived initiation, or backup actuation relays. Should this occur, the current technical specifications would require the plant to be in hot shutdown within 12 hours, despite the fact that the opposite ESF logic subsystem, given the logical combination of initiating signals, would automatically actuate ESF equipment.

The proposed change to LCO 2.15(2) incorporates the discussion of Table 2-3, footnote (k) into the specification. The change clarifies a less restrictive time frame (eight hours) for an inoperable channel to be placed in the tripped position if jumpers or blocks are required. For all the functional units in Table 2-3 that footnote (d) are applicable, jumpers are necessary for the channel to be placed in the tripped position.

The new LCO 2.15(3) proposes a 48-hour allowed outage time (AOT) to restore ESF logic in the event that the number of channels associated with one of the two ESF logic subsystems falls below the limits in the TS. This addition will allow sufficient time to troubleshoot, repair, and test an inoperable ESF logic subsystem and prevent unnecessary plant shutdowns. The loss of a prime initiation relay (which renders all four channels inoperable) is the condition most likely to cause entry into TS 2.15(3). In this situation, the remaining ESF logic subsystem would still have the capability to automatically actuate ESF equipment.

The 48-hour completion time is commensurate with the importance of avoiding the vulnerability of a single failure in the remaining ESF logic subsystem. In addition, it is identical to the AOT in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants," Rev. 1, for a similar situation regarding inoperable ESF logic channels. Therefore, based on the above, the addition of the proposed LCO is acceptable.

In addition to the above, OPPD proposed administrative changes to clarify (1) the definition of ESF logic to note that the CRHS is 1-out-of-2 logic, (2) TS 2.15(2) to be consistent with footnotes in Tables 2-2, 2-3 and 2-4 regarding the requirements when more than one plant instrumentation channel is inoperable, (3) TS 2.15(4) (formerly TS 2.15(3) so that it does not apply to ESF logic, which would be covered by the new TS 2.15(3), and (4) the Basis for TS 2.15 to provide additional detail concerning ESF logic. The proposed administrative clarifications were reviewed and found to be acceptable.

3.0 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.

4.0 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 (63 FR 6987). 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.

5.0 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.

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