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

August 31, 2011

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: REVISION OF TECHNICAL SPECIFICATIONS RELATED TO CONTROL ELEMENT ASSEMBLY POSITION INDICATION (TAC NO. ME4230)

Dear Mr. Bannister:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 267 to Renewed Facility Operating License No. DPR-40 for the Fort Calhoun Station (FCS), Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated July 12, 2010, as supplemented by letters dated February 4 and August 19, 2011.

The amendment revises language in TS 2.10.2, "Reactivity Control Systems and Core Physics Parameters Limits," and revises table items related to TS 2.15, "Instrumentation and Control Systems," which pertain to operability of the primary and secondary control element assembly (CEA) position indication system (CEAPIS) channels. A new surveillance requirement is added to TS 3.1, "Instrumentation and Control," to verify the position of the CEAs each shift. The surveillance requirement for TS 3.1, Table 3-3, Item 1.a is modified to specify performance of a CHANNEL CHECK of the primary CEAPIS.

By letter dated August 19, 2011, the licensee withdrew its proposed changes to Item 2.a of TS 3.1, Table 3-3 regarding the secondary CEAPIS.

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,

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. 267 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. 267
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 July 12, 2010, as supplemented by letters dated February 4 and August 19, 2011, 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. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 267, 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 180 from the date of issuance.

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: August 31, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 267

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
2.10 – Page 7	2.10 – Page 7
2.15 – Page 14	2.15 – Page 14
3.1 – Page 15	3.1 – Page 15
3.1 – Page 16	3.1 – Page 16

(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. 267 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.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.2 Reactivity Control Systems and Core Physics Parameters Limits (Continued)

b. When the regulating CEA groups are inserted beyond the Long Term Insertion Limit for a time interval in excess of 4 hours per 24 hour interval, operation may proceed deleting this daily restriction, provided either:

(i) Regulating CEA groups are not inserted below the Short Term Insertion Limit, or

(ii) Regulating CEA groups are not inserted below the Transient Insertion Limit and rates of power increases initiated when the regulating CEA's are inserted below the Short Term Insertion Limit are less than 5%/hour.

c. When the regulating CEA groups are inserted below the Long Term Insertion Limit for time intervals in excess of 4 EFPD per 30 EFPD interval or in excess of 14 EFPD per fuel cycle, either:

(i) Restore the regulating groups to within the Long Term Insertion Limit within two hours, or

(ii) Be in hot shutdown within 6 hours.

(8) CEA Drop Time

The individual full length (shutdown and regulating) CEA drop time, from a fully withdrawn position, shall be ≤ 2.5 seconds from the time the clutch coil is de-energized until the CEA reaches its 90 percent insertion position with:

a. $T_{cold} \geq$ 515°F, and
b. All reactor coolant pumps operating.

With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to hot standby or power operation.

(9) Test Exemption

a. CEA Insertion Limits and Misalignment

(i) The insertion limits of Specification 2.10.2 may be suspended during the performance of physics tests provided:

TABLE 2-5

Instrumentation Operating Requirements for Other Safety Feature Functions

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition
1	CEA Position Indication Systems	1[(e)]	None	None
2	Pressurizer Level	1	None	Not Applicable
3	PORV Acoustic Position Indication-Direct	1[(a)(c)]	None	Not Applicable
4	Safety Valve Acoustic Position Indication	1[(a)(c)]	None	Not Applicable
5	PORV/Safety Valve Tail Pipe Temperature	1[(d)(b)]	None	Not Applicable

NOTES:

a One channel per valve.
b One RTD for both PORV's; two RTD's, one for each code safety.
c If item 5 is operable, requirements of specification 2.15 are modified for items 3 and 4[i] to "Restore inoperable channels to operability within 7 days or be in hot shutdown within 12 hours."
d If items 3 and 4 are operable, requirements of specification 2.15 are modified for item 5 to "Restore inoperable channels to operability within 7 days or be in hot shutdown within 12 hours."
e If one channel of CEA position indication is inoperable for one or more CEAs, requirements of specification 2.15 are modified for item 1 to "Perform TS 3.1, Table 3-3, Item 4 within 15 minutes following any CEA motion in that group." Specifications 2.15(1), (2), and (3) are not applicable.

i The requirement of Table 2-5, Note c to restore Safety Valve Acoustic Position Indication in 7 days is extended on a one-time basis. This allows the instrumentation for Functional Unit 4 for pressurizer safety valve RC-142 to be inoperable from June 1, 2010 until the next entry into Mode 3 from Mode 4.

TABLE 3-3

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

Channel Description	Surveillance Function	Frequency	Surveillance Method
1. Primary CEA Position Indication System	a. Check	S	a. CHANNEL CHECK
	b. Test	M	b. Test of power dependent insertion limits, deviation, and sequence monitoring systems.
	c. Calibrate	R	c. Physically measured CEDM position used to verify system accuracy. Calibrate CEA position interlocks.
2. Secondary CEA Position Indication System	a. Check	S	a. Comparison of output data with primary CEAPIS.
	b. Test	M	b. Test of power dependent insertion limit, deviation, out-of-sequence, and overlap monitoring systems.
	c. Calibrate	R	c. Calibrate secondary CEA position indication system and CEA interlock alarms.
3. Area and Post-Accident Radiation Monitors[(1)]	a. Check	D	a. CHANNEL CHECK
	b. Test	Q	b. CHANNEL FUNCTIONAL TEST
	c. Calibrate	R	c. Secondary and Electronic calibration performed at refueling frequency. Primary calibration with exposure to radioactive sources only when required by the secondary and electronic calibration. RM-091 A/B - Calibration by electronic signal substitution is acceptable for all range decades above 10 R/hr. Calibration for at least one decade below 1-R/hr. shall be by means of calibrated radiation source.

[(1)]Post Accident Radiation Monitors are: RM-063, RM-064, and RM-091A/B. Area Radiation Monitors are: RM-070 thru RM-082, RM-084 thru RM-089, and RM-095 thru RM-098.

TABLE 3-3 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

Channel Description	Surveillance Function	Frequency	Surveillance Method
4. CEA Position	a. Check	S	a. Verify the position of each CEA to be within 12 inches of all other CEAs in the group.
5. Primary to Secondary Leak-Rate Detection Radiation Monitors (RM-054A/B, RM-057)	a. Check	D	a. CHANNEL CHECK
	b. Test	Q	b. CHANNEL FUNCTIONAL TEST
	c. Calibrate	R	c. Secondary and Electronic calibration performed at refueling frequency. Primary Calibration performed with exposure to radioactive sources only when required by the secondary and electronic calibration.
6. Pressurizer Level	a. Check	S	a. Verify that level is within limits.
	b. Check	M	b. CHANNEL CHECK
	c. Calibrate	R	c. CHANNEL CALIBRATION
7. CEA Drive System Interlocks	a. Test	R	a. Verify proper operation of all CEDM system interlocks, using simulated signals where necessary.
	b. Test	P	b. If haven't been checked for three months and plant is shutdown.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 267 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 July 12, 2010, as supplemented by letters dated February 4 and August 19, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML101930443, ML110380127, and ML112311424, respectively), Omaha Public Power District (OPPD, the licensee) requested changes to the Technical Specifications (TSs) (Appendix A to Renewed Facility Operating License No. DPR-40) for the Fort Calhoun Station (FCS), Unit No. 1.

The proposed amendment would revise language in TS 2.10.2, "Reactivity Control Systems and Core Physics Parameters Limits," and revise table items related to TS 2.15, "Instrumentation and Control Systems," which pertain to operability of the primary and secondary control element assembly (CEA) position indication system (CEAPIS) channels. A new surveillance requirement (SR) would be added to TS 3.1, "Instrumentation and Control," to verify the position of the CEAs each shift. The SR for TS 3.1, Table 3-3, Item 1.a would be modified to specify performance of a CHANNEL CHECK of the primary CEAPIS.

The supplemental letters dated February 4 and August 19, 2011, 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 January 25, 2011 (76 FR 4387).

By letter dated August 19, 2011, OPPD withdrew its proposed changes to Item 2.a of TS 3.1, Table 3-3 regarding the secondary CEAPIS. The withdrawal reduces the scope of the application as originally noticed and, therefore, did not change the NRC staff's original proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

2.1 System Description

The FCS CEAs consist of five Inconel tubes containing boron carbide as a poison material. The tubes are joined by a spider at the upper end, and the hub of the spider couples the CEA to the drive mechanism rack through the drive shaft extension. FCS uses a rack and pinion drive system to insert and withdraw the CEAs. The CEAs can be rapidly inserted when the drive system releases them allowing them to fall into the reactor by gravity as required by a reactor trip. FCS has 45 standard and 4 full-length non-trippable CEAs. The standard CEAs are used for shutdown and regulation, or reactivity control. The non-trippable CEAs are required by the TS to remain in an essentially fully withdrawn position during power operation and also provide additional shutdown margin, as needed, after a reactor shutdown. There are two shutdown groups and four regulating groups. The shutdown groups are fully withdrawn when starting the reactor and stay fully withdrawn until a time the reactor is shut down. The regulating groups are adjusted to meet reactivity and power distribution requirements. Both groups are inserted fully by a reactor trip.

The primary CEAPIS channel utilizes the output of a Synchro transmitter geared to the drive shaft of the control element drive mechanism (CEDM). When the sending unit of the rotor is moved, the current shift moves the receiver unit a corresponding amount. CEA position, accurate to within ±0.5 inches is supplied to the plant computer for indication and control. Synchro indication is also provided to the secondary CEAPIS distributed control system (DCS). CEA position is displayed visually at the main control panel as groups A, B, 1, 2, 3, 4, and N.

The secondary CEAPIS channel measures CEA position by use of magnetic reed switches actuated by a permanent magnet attached to the rack assembly. There are 64 reed switches in a 128-inch-long string provided for each CEDM. An assembly containing a number of series resistors to form a voltage divider network with reed switches connected at each junction is attached to the CEA extension housing. A voltage is applied to the network. Output voltage depends on which reed switches are closed in the voltage divider. A magnet on top of the CEA extension actuates the reed switches as the CEA moves. Overlap between adjacent reed switches is provided. The output is a voltage directly proportional to CEA position. The information from the secondary CEAPIS is displayed on flat-panel touch monitors

CEA full-in and full-out indication is provided from limit switches on the regulating CEAs and reed switches on the shutdown CEAs and is displayed on the DCS flat-panel touch monitors.

2.2 General Requirements

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TS as part of the license. The Commission's regulatory requirements related to the content of TSs are contained in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.36, "Technical specifications. The TS requirements in 10 CFR 50.36 include 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) SRs; (4) design features; and (5) administrative controls. The regulations in 10 CFR 50.36(c)(2)(i) state, in part, that

> Limiting 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.

The regulations in 10 CFR 50.36(c)(2)(ii) state, in part, that

> A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:
>
> (B) *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The regulations in 10 CFR 50.36(c)(3), "Surveillance requirements," state that

> Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

In a memorandum dated September 18, 1992 (ADAMS Legacy Library Accession No. 9210060362) (ADAMS Accession No. ML003763736), 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. 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).

By letter dated July 12, 2010, the licensee appropriately identified the following draft GDC as specified in Appendix G to the FCS USAR:

> <u>FCS Design Criterion 6 – Reactor Core Design</u> states:
>
> The reactor core shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to

recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all off-site power.

FCS Design Criterion 27 – Redundancy of Reactivity Control states:

At least two independent reactivity control systems, preferably of different principles, shall be provided.

FCS Design Criterion 29 – Reactivity Shutdown Capability states:

At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

FCS Design Criterion 31 – Reactivity Control Systems Malfunction states:

The reactivity control systems shall be capable of sustaining any single malfunction, such as unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

The NRC staff also used FCS Design Criterion 13 – Fission Process Monitors and Controls in its review, which states:

Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Changes

The licensee's proposed changes are as follows:

- TS 2.15, Table 2-5, Item 1 specifies the minimum number of CEAPIS channels required to be operable (i.e., one channel). New note e is proposed which would modify the requirements of TS 2.15 and would state:

  If one channel of CEA position indication is inoperable for one or more CEAs, requirements of specification 2.15 are modified for item 1 to "Perform TS 3.1, Table 3-3, Item 4 within 15 minutes following any CEA motion in that group." Specifications 2.15(1), (2), and (3) are not applicable.

- TS 3.1, Table 3-3, Item 4 is being proposed to require operators to "Verify the position of each CEA is within 12 inches of all other CEAs in its group" each shift to ensure the alignment required by TS 2.10.2(4) is maintained. In addition, Item 1.a is being modified. The current TSs specifically require a check of primary CEAPIS data with secondary CEAPIS data and vice versa each shift. The modified Item 1.a now specifies performance of a "CHANNEL CHECK" of the primary and CEAPIS each shift.

- TS 2.10.2(7)c currently states that "When regulating CEA groups are inserted below the Long Term Insertion Limit for time intervals in excess of 4 EFPD [effective full power days] per 30 EFPD interval and 14 EFPD per fuel cycle, either" of two specified actions must be taken. The proposed change will replace the "and" condition with "or in excess of" which will require the actions if either condition (as opposed to both conditions) are met.

3.2 Precedents

In its letter dated June 12, 2010, the licensee stated, in part, that

> The portion of the proposed amendment pertaining to CEA position indication is consistent with LCO 3.1.4, "Control Rod Alignment," Action A, and associated SRs 3.1.4.1 and 3.1.4.2 of NUREG-1432; "Standard Technical Specifications - Combustion Engineering Plants," Revision 1 dated April 1995 as approved for the Palisades Plant. . . . Like Palisades, FCS is a CE Analog plant of similar vintage (e.g., Palisades was issued its Operating License in 1971 while FCS obtained its in 1973) and has similar CEA position indication systems based on primary indication from synchros and secondary indication from reed switches.

3.3 NRC Staff Evaluation

The proposed modifications do not involve any physical alterations of the FCS facility. The licensee's proposed amendment would modify three sections of its TSs. The NRC staff reviewed the licensee's application against the requirements stated in Section 2.2 of this safety evaluation.

3.3.1 TS 2.15, Table 2-5, Item 1

TS 2.15, Table 2-5, Item 1 currently specifies the minimum number of CEAPIS channels required to be operable. A new note (note e) is proposed which would modify the requirements of TS 2-15 to require the performance of a new SR (proposed TS 3.1, Table 3-3, Item 4) within 15 minutes following any CEA motion in that group in the event that one of the CEAPIS channels is inoperable. Proposed note e would also clarify that TS 2.15(1), (2), and (3) are not applicable to CEAPIS channels.

In addition to providing data on CEA positions and initiating alarms based upon specified CEA positions, the CEAPIS – specifically, the secondary CEAPIS – may initiate the rod (CEA) block system to inhibit CEA motion in the event of an LCO related to specified CEA positioning or motion. TS 2.15(4) contains requirements to address the event that the secondary CEAPIS and rod block function becomes inoperable or that the plant computer Power Dependent Insertion Limit alarm, CEA group deviation alarm, and the CEA sequencing function are inoperable. The amendment proposed by FCS would not alter TS 2.15(4) or its applicability.

TS 2.15(1) states that if one instrumentation and control system channel of a particular system covered by TS 2.15 becomes inoperable, the inoperable channel must be placed in either bypassed or tripped condition. Specific time limits are identified for various actions related to this occurrence. Since the CEAPIS does not initiate a reactor trip or safety system actuation, the NRC staff concludes it is acceptable that this provision not apply to failure of a single CEAPIS channel.

TS 2.15(2) states that if the number of operable instrumentation and control system channels of a particular system covered by TS 2.15 falls to the specified "Minimum Operable Channels" limit, one of the inoperable channels must be placed in the tripped position or low level actuation permissive position for the auxiliary feedwater system. Specific time limits are identified for various actions related to this occurrence. Since the CEAPIS does not initiate a reactor trip or safety system actuation, the NRC staff concludes it is acceptable that this provision not apply to the instance of only the minimum number of CEAPIS channels being operable.

TS 2.15(3) states that if the number of instrumentation and control system channels of an engineered safety feature or isolation logic subsystem covered by TS 2.15 falls below the "Minimum Operable Channels" limit value identified for that system, sufficient channels must be restored to operable status within a time limit. Specific time limits are identified for various actions related to this occurrence. Since the CEAPIS is not an engineered safety feature or isolation logic subsystem, the NRC staff concludes it is acceptable that this provision not apply to the instance of only the minimum CEAPIS channels being operable.

3.3.2 TS 3.1, Table 3-3, Item 4

TS 3.1, Table 3-3, Item 4 is being proposed to require operators to verify the position of each CEA is within 12 inches of all other CEAs in its group each shift to ensure the alignment required by TS 2.10.2(4) is maintained. In addition, Item 1.a is being modified from its current language which specifically requires the primary CEAPIS data to be checked against secondary CEAPIS data and vice versa each shift. The modified Item 1.a now specifies performance of a CHANNEL CHECK for the primary CEAPIS data each shift.

The proposed TS 3.1, Table 3-3, Item 4 would be a new check on CEA position (in addition to Items 1.a and 2.a of Table 3-3). This proposed check is specifically aimed at ensuring that FCS LCO 2.10.2(4) is met. The NRC staff concludes that the addition of Item 4 is a conservative change to the FCS TS since it represents a new surveillance requirement (in addition to Items 1.a and 2.a of Table 3-3) that is specifically aimed at providing assurance that an existing LCO is met. The staff concludes that the proposed TS 3.1, Table 3-3, Item 4 addition is consistent with 10 CFR 50.36(c)(3) and, therefore, is acceptable.

CHANNEL CHECK is defined in the FCS TS as:

> 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.

The modification of TS 3.1, Table 3-3, Item 1.a would change the TS table to state that a "CHANNEL CHECK" of the primary CEAPIS is required for each channel. Currently, each channel is required to have its output data compared against only the other CEAPIS channel to verify operability. Changing the language would permit FCS to use CEA limit switches and shutdown CEA reed switches, which indicate whether CEAs are fully withdrawn or fully inserted, to be used to during channel checks of the primary CEAPIS.

Per FCS Design Criteria 13, "means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons." FCS Design Criteria 13 also specifically identifies that "two independent rod position indicating systems" are provided to monitor and maintain control over the fission process.

Changing the language to indicate use of a CHANNEL CHECK would still permit the primary CEAPIS data to be verified against the secondary CEAPIS and the secondary CEAPIS data to be verified against the primary CEAPIS. Section 7.5.3.1 of the FCS USAR specifically states that "the primary and secondary CEA position sensing systems are separate and independent." The new capability that would be enabled via use of the requirement for a CHANNEL CHECK is the use of the CEA full-in/full-out indications (when the CEAs are either all in or all out) as a source of data for the CHANNEL CHECK.

By letter dated February 4, 2011, the licensee's response to the NRC staff's request for additional information (RAI) dated January 7, 2011 (ADAMS Accession No. ML103550188), regarding the independence between the primary CEAPIS and the CEA full-in/full-out indications stated that "from sensor output to control room indication, the data provided by CEA full-in and full-out indication is separate and independent from the primary CEAPIS." This statement is supported by system descriptions. Since the primary CEAPIS and the CEA full-in/full-out indication are independent, the NRC staff concludes that when the CEAs are either full-in or full-out, the CEA full-in/full-out indication may be used as a CHANNEL CHECK for the primary CEAPIS.

The NRC staff concludes that altering the language of TS 3.1, Table 3-3, Item 1.a to specify a "CHANNEL CHECK" to be acceptable, as the primary CEAPIS and CEA full-in/full-out indication are independent and consistent with FCS Design Criterion 13.

3.3.3 <u>TS 2.10.2(7)c</u>

The license's proposed modification to TS 2.10.2(7)c involves rewording the text from the existing language

> When the regulating CEA groups are inserted below the Long Term Insertion Limit for time intervals in excess of 4 EFPD per 30 EFPD interval <u>and</u> 14 EFPD per fuel cycle . . .

to

> When the regulating CEA groups are inserted below the Long Term Insertion Limit for time intervals in excess of 4 EFPD per 30 EFPD interval <u>or in excess of</u> 14 EFPD per fuel cycle . . .

The new language is a more conservative implementation of this LCO in the FCS TS, since the condition for action will be met with the new wording in the event that either condition is met, as opposed to the existing wording which required both conditions to be met.

Based upon the proposed change being more conservative than the existing language with no decrease in level of protection provided by FCS LCO 2.10.2, the NRC staff concludes the change will continue to meet the requirements of 10 CFR 50.36(c)(2)(i) and is, therefore, acceptable.

3.3.4 <u>CEA Left in the Core</u>

Startup or operation with the most reactive CEA left in the core would result in a large distortion in the radial power distribution and would cause excessive peaking. The FCS rod block circuitry prohibits the CEAs from exceeding 12 inches of deviation. If the group is being withdrawn and a CEA remains fully inserted, deviation alarms will be provided at 4 inches and 8 inches of deviation. When 12 inches of deviation is reached, the motion of the group is stopped by the rod block circuitry. The proposed amendment would allow the licensee to verify the position of the CEAs using position indication channels. Because the rod block function will stop rod motion if 12 inches of deviation occurs, the condition will be prevented.

To ensure that all of the CEAs are operational after coming out of a refueling outage, FCS measures the coupled and uncoupled weight of the CEAs to make sure that all of the CEAs are coupled. This helps ensure that when banks of rods are being withdrawn, no rods are left behind. If a rod was left fully inserted while its group was being withdrawn, alarms from the incore monitoring system would also occur due to flux peaking as well as a change in flux tilt.

The NRC concludes that these measures, combined with the rod block circuitry, help assure that CEAs will not be withdrawn with one left in the core.

3.3.5 <u>Insertion of the CEA Bank Permissible at Full Power with One CEA Left Out</u>

Insertion of the Group 4 regulating CEAs with one of the CEAs left withdrawn would also cause excessive peaking and distortion of power in the core. The FCS rod block circuitry prohibits the CEAs from exceeding 12 inches of deviation. If the group is being inserted and a CEA remains in the full out position, deviation alarms will be provided at 4 inches and 8 inches of deviation. When 12 inches of deviation is reached, the motion of the group is stopped by the rod block circuitry. The proposed amendment allows for position indication channels to verify the position of the CEAs. Because the rod block function will stop rod motion if 12 inches of deviation occurs the condition will be prevented. In the event that the rod block circuitry is inoperable, the CEAs are required to remain fully withdrawn with the mode switch off. If the Group 4 rods are required for axial power distribution control while the rod block circuitry is inoperable, the licensee will be required to use channel checks of the primary system to assure location of all of the CEAs as well as no CEAs left out and is, therefore, acceptable.

3.3.6 CEA Withdrawal Incident

The sequential CEA group withdrawal incident is assumed to occur as a result of a failure in the CEDM control system or by operator error. The licensee installed the rod block system after Cycle 1, which eliminates the possibility of an out-of-sequence bank withdrawal or a single CEA withdrawal due to a single failure. The CEA withdrawal incident results in a positive reactivity addition and causes core power to rise and decreases departure from nucleate boiling and linear heat rate margins.

The NRC staff reviewed the effect of the proposed amendment on the CEA withdrawal incident. The proposed amendment allows for position indication channels to verify the position of the CEAs. Because the rod block function will stop rod motion if 12 inches of deviation occurs, the condition will be prevented. The proposed amendment will require operators to ensure that the CEAs are positioned within 12 inches of all of the other CEAs in that group within 15 minutes of rod movement if a CEAPIS channel is inoperable. The rod block function, along with the new requirements for verification of CEA position after movement if a CEAPIS channel is inoperable, will assure safety and does not cause or contribute to a CEA withdrawal incident and is, therefore, acceptable.

3.3.7 Control Element Assembly Drop

The CEA drop accident is defined as the inadvertent release of a CEA causing it to drop into the reactor core. CEAs at FCS are driven by a rack and pinion set of gears and a motor. For a CEA to drop into the reactor core, an electrical interruption of the CEA holding coil or an electrical or mechanical failure of the mechanical break in the CEA mechanism would occur.

The NRC staff reviewed the effect of the proposed amendment on the CEA drop. The proposed amendment pertains to the CEAPIS. The CEA drop accident is not affected by the position indication system operability. The CEA drop accident is not adversely impacted by the proposed amendment.

3.3.8 CEA Ejection Accident

The CEA ejection accident pertains to a mechanical failure in the form of a circumferential rupture of a CEDM housing or nozzle on the reactor vessel head resulting in the ejection of a control rod. When a rod is ejected, a rapid reactivity insertion occurs along with a power distribution change that may result in localized fuel damage and is, therefore, acceptable.

The NRC staff reviewed the effect of the proposed amendment on the CEA ejection accident. The proposed amendment pertains to the CEAPIS. The CEA ejection accident is not initiated or affected by the operability of the CEAPIS channels. The CEA ejection accident is not adversely impacted by the proposed amendment and is, therefore, acceptable.

3.3.9 Loss-of-Coolant Flow Incident and Main Steam Line Break Accident

A loss of normal coolant flow can result from either a loss of electrical power to one or more of the four reactor coolant pumps or from a mechanical failure, such as shaft seizure of a single pump. These events are referred to as a loss-of-coolant flow event and a seized rotor event.

A main steam line break accident involves a large pipe break in the main steam system resulting in a rapid depletion of the steam generator inventory causing an increased rate of heat extraction from the primary coolant. The cooldown of the primary coolant along with a negative moderator temperature coefficient will cause an increase in nuclear reactor power prior to a reactor trip.

The NRC staff reviewed the effect of the proposed amendment on the loss-of-coolant flow incident and the main steam line break accident. In both cases, the analysis assumes that the CEA with the most worth is stuck fully withdrawn. The proposed amendment would require verification of the position of the CEAs to ensure that the groups are within 12 inches of each other and verify the position of CEAs after any rod movement. The actions that would be permitted by the proposed amendment would not contribute to or initiate either of these accidents. The actions would not contribute to a CEA being stuck in the fully withdrawn position.

Based on the above, the NRC staff concludes that the proposed changes are acceptable as it would not affect the loss-of-coolant flow incident or the main steam line break accident.

3.3.10 Malpositioning of Non-trippable CEAs

FCS has four non-trippable CEAs installed. By practice, these rods are withdrawn to the all-rods-out position. The inadvertent insertion of the non-trippable group from a withdrawn condition is still possible and would cause distortion of the axial power distribution.

The NRC staff reviewed the effect of the proposed amendment on the malpositioning of non-trippable CEAs. The proposed amendment allows for position indication channels to verify the position of the CEAs. The non-trippable CEAs are included in the monitoring and will still be required to be verified with 12 inches of each of the other CEAs in the group. While the rods are normally withdrawn to the all-rods-out position, inadvertent insertion of the non-trippable CEAs is possible. However, the incident is not caused or adversely affected by the proposed amendment, therefore, the NRC staff concludes the proposed amendment is acceptable.

3.3.11 NRC Staff Conclusion

Based on the above, the NRC staff concludes the TS changes proposed for TS 2.15, Table 2-5, Item 1 is acceptable, as it clarifies that certain LCO actions are not applicable to the CEAPIS channels and are instead intended for other systems listed in Table 2-5. The NRC staff concludes that these changes will continue to meet the requirements 10 CFR 50.36(c)(2)(i).

The NRC staff concludes that the addition of TS 3.1, Table 3-3, Item 4 is consistent with 10 CFR 50.36(c)(3) and is, therefore, acceptable. The modification of the wording in Item 1.a of

TS 3.1, Table 3-3 is also acceptable, as it will enable a primary CEAPIS check that is consistent with FCS Design Criterion 13.

The NRC staff concludes that the change to the wording in TS 2.10.2(7)c is acceptable, as it reduces the conditions required to invoke the action statement and is consistent with 10 CFR 50.36(c)(2)(i).

Furthermore, the NRC staff concludes the proposed TS changes to TS 2.15 and TS 3.13 regarding an inoperable CEAPIS channel and verification of CEA positions following any CEA movement are acceptable. The staff reviewed applicable accident analysis as well as worst-case of a CEA left in the core and insertion of the CEA bank permissible at full power with one CEA left out. The NRC staff concludes the changes are consistent with 10 CFR 50.36(c)(3) and the applicable Design Criteria.

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. 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 NRC staff 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 Contributors: T. Mossman
J. Miller

Date: August 31, 2011

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/

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. 267 to DPR-40
2. Safety Evaluation

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