Omaha Public Power District
444 South 16th Street Mall
Omaha, NE 68102-2247

LIC-15-0053

10 CFR 50.90

August 20, 2015

U. S. Nuclear Regular Commission
Document Control Desk
Washington, DC 20555

Fort Calhoun Station, Unit No. 1
Renewed Facility Operating License No. DPR-40
NRC Docket No. 50-285

Subject: License Amendment Request (LAR) 15-04; Application to Revise Technical Specification for Administrative Changes

In accordance with the provisions of 10 CFR 50.90, the Omaha Public Power District (OPPD), is submitting a request for an amendment to the Technical Specifications (TS) for Fort Calhoun Station (FCS), Unit No. 1.

The proposed amendment would modify the TS to make administrative changes. The enclosure contains a description of the proposed changes, the supporting technical analyses, and the significant hazards consideration determination. Attachment 1 of the enclosure provides the existing TS page marked-up to show the proposed changes. Attachment 2 of the enclosure provides retyped (clean) pages with the changes proposed by Attachment 1 and denoted by revision bars in the margin.

The proposed changes have been reviewed and approved by the Fort Calhoun Station Plant Operations Review Committee (PORC) and by the Nuclear Safety Review Board (NSRB).

OPPD requests approval of the proposed license amendment by August 8, 2016, with the amendment to be implemented within 90 days of issuance.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated State of Nebraska official.

There are no regulatory commitments contained within this letter.

If you should have any questions regarding this submittal or require additional information, please contact Mr. Bill R. Hansher at (402) 533-6894.

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 20, 2015.

Respectfully,

Louis P. Cortopassi
Site Vice President and CNO

LPC/BRH/brh

Enclosure: OPPD's Evaluation of the Proposed Change

c: M. L. Dapas, NRC Regional Administrator, Region IV
C. F. Lyon, NRC Senior Project Manager
S. M. Schneider, NRC Senior Resident Inspector
Director of Consumer Health Services, Department of Regulation and Licensure, Nebraska Health and Human Services, State of Nebraska

OPPD's Evaluation of the Proposed Change

License Amendment Request (LAR) 15-04, Application to Revise Technical Specifications for Administrative Changes

Attachments:
1. Mark-up of Technical Specification Page
2. Clean Technical Specification Pages
3. Mark-up of Technical Specification Bases Page for Information

1.0 SUMMARY DESCRIPTION

The Omaha Public Power District (OPPD) hereby requests an amendment to Fort Calhoun Station (FCS), Unit No. 1 Renewed Facility Operating License No. DPR-40 to implement administrative changes to update personnel and committee titles, delete outdated or completed additional actions contained in Appendix B of the License, and relocate the definition of Process Control Program.

2.0 DETAILED DESCRIPTION

In August 2012, OPPD and Exelon entered into a 20-year Operating Services Agreement by which Exelon manages day-to-day operations. The Fort Calhoun Station Unit No.1 is implementing procedures and processes from Exelon as part of this agreement and as a result is requesting certain Technical Specifications (TS) be revised to use consistent terminology.

The title for the Plant Review Committee is being revised to Plant Operations Review Committee. This proposed change revises the TS definition of Physics testing, footnotes in TS Table 2-3 footnotes and TS 5.17. This proposed change also revises the Bases to TS 2.13 which is included for information.

The Control Room Supervisor title is being revised to Unit Supervisor. This proposed change revises TS 5.2.2.e.

The term Restricted High Radiation Area is being revised to Locked High Radiation Area. This proposed change revises TS 5.11.2. In addition, allowance for a designee for the Manager Radiation Protection is being added to clarify that a designated alternate to the Manager Radiation Protection may control keys to locked high radiation areas. Allowing a designated alternate to the Manager-Radiation Protection is consistent with the Improved Standard TS (NUREG-1432).

Relocation of the Process Control Program. The Process Control Program was added to the TS by Amendment 152 (Reference 6.4). The FCS definition is not verbatim to that used in fleet procedures. It is proposed to re-locate the definition and program requirements to the USAR/plant procedures as the program is adequately controlled by regulation and its re-location is consistent with it being relocated to plant procedures as part of the development of the Improved Standard Technical Specifications (NUREG-1432 for Combustion Engineering plants). This proposed change deletes the definition and TS 5.18.

The following are additional administrative changes.

Deletion of Additional Condition (1) contained in Appendix B for Amendment 181.

> *(1) The licensee is authorized to relocate certain technical specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated November 20, 1996, as supplemented by letters dated February 20, 1997, and March 25, 1997, and evaluated in the staff's safety evaluation dated March 27, 1997.*

TS Amendment 181 (Reference 6.1) relocated controls for plant staff working hours to Updated Safety Analysis Report (USAR) Section 12.1.5. This USAR section continues to describe controls on work hour limitations; however with the issuance of 10 CFR Part 26 Subpart I and TS Amendment 262 (Reference 6.2) it is no longer necessary to contain this additional condition in the TS to ensure that the limitations will be followed as the controls are implemented through regulation; therefore it is proposed that this condition be deleted.

Deletion of Additional Condition (2) contained in Appendix B for Amendment 257.

> *(2) Upon implementation of Amendment No. 257 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by TS 3.1, Table 3-3, Item 10.b. in accordance with TS 5.24c.(i), the assessment of CRE habitability as required by Specification 5.24c.(ii), and the measurement of CRE pressure as required by Specification 5.24d, shall be considered met. Following implementation:*
>
> *(a) The first performance of TS 3.1, Table 3-3, Item 10.b., in accordance with Specification 5.24c.(i), shall be within the next 18 months as the time period since the most recent successful tracer gas test is greater than 6 years.*
>
> *(b) The first performance of the periodic assessment of CRE habitability, Specification 5.24c(ii), shall be within the next 9 months as the time period since the most recent successful tracer gas test is greater than 3 years.*
>
> *(c) The first performance of the periodic measurement of CRE pressure, Specification 5.24d., shall be within the next 138 days.*

TS Amendment 257 was issued June 30, 2008 (Reference 6.3) with an implementation date of within 270 days of its issuance. This additional condition specified when first performances of specific items were required to be complete. The requirement of Condition (2)(a) for the first performance of TS 3.1 Table 3-3, Item 10.b, tracer gas testing was performed in January 2010. The requirement of Condition (2)(b) for the first performance of the periodic assessment of CRE habitability was performed in January 2010. The requirement of Condition (2)(c) for the first performance of periodic measurement of CRE pressure in accordance with TS 5.24 d was performed in June 2009.

As the first performances of these tests have been completed, the additional condition contained in Appendix B is no longer necessary; therefore it is proposed that this condition be deleted.

3.0 TECHNICAL EVALUATION

The proposed changes are administrative in nature and have no technical implications with respect to the station organization, responsibilities or unit staffing requirements. Deletion of the additional conditions contained in Appendix B of the TS and the relocation of the Process Control Program only delete one-time first performance of activities that have been completed, and requirements that are redundant to regulation.

REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

4.1.1 Title 10 CFR Part 26 for controlling plant staff working hours

4.1.2 Title 10 CFR 50.36 for information to be included in Technical Specifications

4.2 Precedent

4.2.1 Precedent for the relocation of the Process Control Program - Amendment 146 to Joseph M. Farley Nuclear Plant, Unit No. 1, Farley Nuclear Plant ITS Conversion, issued November 30, 1999 designated the relocation as an administrative requirement redundant to regulations.

4.3 No Significant Hazards Consideration

The Omaha Public Power District (OPPD) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes are administrative in nature, involving changes to personnel and committee titles, deletion and or re-location of requirements redundant to regulations, and deletion of conditions controlling the first performance of testing that has since been completed. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because: 1) the proposed amendment does not represent a change to the system design, 2) the proposed amendment does not alter, degrade, or prevent action described or assumed in any accident in the USAR from being performed, 3) the proposed amendment does not alter any assumptions previously made in evaluating radiological consequences, and 5) the proposed amendment does not affect the integrity of any fission product barrier. No other safety related equipment is affected by the proposed change.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes do not alter the physical design, safety limits, or safety analysis assumptions associated with the operation of the plant. Hence, the proposed changes do not introduce any new accident initiators, nor do these changes reduce or adversely affect the capabilities of any plant structure or system in the performance of their safety function.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed changes do not alter the manner in which safety limits or limiting safety system settings are determined. The safety analysis acceptance criteria are not affected by these proposed changes. Further, the proposed changes do not change the design function of any equipment assumed to operate in the event of an accident.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, OPPD concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

4.4 Conclusion

In conclusion, based on the considerations discussed above, (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.

5.0 ENVIRONMENTAL CONSIDERATION

A review of the proposed amendment has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any

effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

6.1. Letter from NRC (L. R. Wharton) to OPPD (S. K. Gambhir), Fort Calhoun Station, Unit No. 1, Amendment No. 181 to Facility Operating License No. DPR-40 (TAC No. M97334) dated March 27, 1997 (NRC-97-0052)

6.2. Letter from NRC (A. B. Wang) to OPPD (D. J. Bannister), Fort Calhoun Station, Unit No. 1 – Issuance of Amendment RE: Adoption of TSTF-511, Revision 0, “Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance with 10 CFR Part 26” (TAC No. ME0595) dated July 24, 2009 (NRC-09-0053)

6.3. Letter from NRC (M. T. Markley) to OPPD (D. J. Bannister), Fort Calhoun Station, Unit No. 1 - Issuance of Amendment RE: Control Room Envelope Habitability (TAC No. MD5577) dated June 30, 2008 (NRC-08-0070)

6.4. Letter from NRC (S. Bloom) to OPPD (T. L. Patterson), Fort Calhoun Station, Unit No. 1 – Amendment No. 152 to Facility Operating License No. DPR-40 (TAC M83720) dated March 28, 1993 (NRC-93-0106)

Fort Calhoun Station, Unit No. 1
Renewed Facility Operating License No. DPR-40

Mark-up of Technical Specification Pages

[Word-processor mark-ups using "double underline/~~strikeout~~" feature for "new text/deleted text" respectively.]

TABLE OF CONTENTS (Continued)

DEFINITIONS

REACTOR OPERATING CONDITIONS (Continued)

Physics Testing

Testing performed under written procedures approved by Plant Operations Review Committee to determine CEA worths and other core nuclear parameters. Deviations from normal operating practice which are necessary to enable some of these tests to be performed are permitted in accordance with: 1) the specific provisions of these technical specifications, 2) authorization under the provisions of 10 CFR 50.59, or 3) other approval of the Commission.

PROTECTIVE SYSTEMS

Reactor Trip

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

Instrument Channel

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

Reactor Protective System Logic[1]

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

DEFINITIONS

Azimuthal Power Tilt - T_q

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

Maximum Radial Peaking Factor (F_R^T)

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

~~Process Control Program (PCP)~~

~~The document(s) that contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 61, 71, State Regulations, burial ground requirements, and other requirements governing the disposal of solid waste.~~

Dose Equivalent I-131

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

Dose Equivalent I-131 (ΦCi/gm)	=	ΦCi/gm of I-131
	+	0.0361 x ΦCi/gm of I-132
	+	0.270 x ΦCi/gm of I-133
	+	0.0169 x ΦCi/gm of I-134
	+	0.0838 x ΦCi/gm of I-135

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(m)	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		
D	Steam Generator Low/Low Pressure				
	Logic Subsystem A	2/Steam Gen(a)(c)(d)(l)	1/Steam Gen	Steam Generator Pressure Less Than 600 psia(n)	(f)
	Logic Subsystem B	2/Steam Gen(a)(c)(d)(l)	1/Steam Gen		
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 containment high pressure, pressurizer low/low pressure, and steam generator low pressure signals are 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 2.15.1(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(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(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 Operations 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 Operations 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 2.15.1(2) is applicable.

l Specification 2.15.1(3) is applicable. If ESF Logic Subsystems A and B are inoperable, enter Specification 2.0.1.

m Steam Generator Low Pressure permissive is required for actuation.

n Auto removal of bypass prior to exceeding 600 psia.

5.0 **ADMINISTRATIVE CONTROLS**

5.2 Organization (Continued)

b. An Operator or Technician qualified in Radiation Protection Procedures shall be onsite when fuel is in the reactor.

c. All core alterations shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator limited to fuel handling who has no other concurrent responsibilities during the operation.

d. Fire protection program responsibilities are assigned to those positions and/or groups designated by asterisks in USAR 12.1-1 through 12.1-4 according to the procedures specified in Section 5.8 of the Technical Specifications.

e. The Manager - Shift Operations, the Shift Managers, and the ~~Control Room~~ Unit Supervisors shall hold a senior reactor operator license. The Licensed Operators shall hold a reactor operator license.

5.3 Facility Staff Qualification

5.3.1 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, with the exception of the Manager - Radiation Protection (MRP) and the Shift Technical Advisor (STA), the senior reactor operator licensees, and the reactor operator licensees, who shall meet the requirements set forth in Regulatory Guide 1.8, Revision 3, dated May 2000, entitled "Qualification and Training of Personnel for Nuclear Power Plants."

5.0 **ADMINISTRATIVE CONTROLS**

5.10 Record Retention

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

5.11 Radiation Protection Program

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

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

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

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

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

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

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

*Radiation Protection personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.
**At 30 centimeters (12 inches) from the radiation source or from any surface penetrated by the radiation.
***At 1 meter from the radiation source or from any surface penetrated by the radiation.

5.0 **ADMINISTRATIVE CONTROLS**

5.16 Radiological Effluents and Environmental Monitoring Programs (Continued)

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

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

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

5.17 Offsite Dose Calculation Manual (ODCM)

Changes to the ODCM:

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

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

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

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

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

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

5.18 ~~Process Control Program (PCP)~~DELETED

~~Changes to the PCP:~~

5.18 ~~Process Control Program (PCP) (Continued)~~

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

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

~~2. A determination that the change will maintain the overall conformance of the solidified waste program to existing requirements of federal, state, or other applicable regulations.~~

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

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

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

5.19 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995," as modified by the following exceptions:

(1) If the Personnel Air Lock (PAL) is opened during periods when containment integrity is not required, the PAL door seals shall be tested at the end of such periods and the entire PAL shall be tested within 14 days after RCS temperature T_{cold} > 210EF.

(2) Type A tests may be deferred for penetrations of the steel pressure retaining boundary where the nominal diameter does not exceed one inch.

(3) Elapsed time between consecutive Type A tests used to determine performance shall be at least 24 months or refueling interval.

b. The containment design accident pressure (P_a) is 60 psig.

Appendix B

Additional Conditions

Renewed Facility Operating License No. DPR-40

Omaha Public Power District shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Conditions	Implementation Date
~~181~~	~~(1) The licensee is authorized to relocate certain technical specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated November 20, 1996, as supplemented by letters dated February 20, 1997, and March 25, 1997, and evaluated in the staff's safety evaluation dated March 27, 1997.~~	~~The amendment shall be implemented as of its date of issuance.~~
~~257~~	~~(2) Upon implementation of Amendment No. 257 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by TS 3.1, Table 3-3, Item 10.b. in accordance with TS 5.24c.(i), the assessment of CRE habitability as required by Specification 5.24c.(ii), and the measurement of CRE pressure as required by Specification 5.24d, shall be considered met. Following implementation:~~ ~~(a) The first performance of TS 3.1, Table 3-3, Item 10.b., in accordance with Specification 5.24c.(i), shall be within the next 18 months as the time period since the most recent successful tracer gas test is greater than 6 years.~~ ~~(b) The first performance of the periodic assessment of CRE habitability, Specification 5.24c(ii), shall be within the next 9 months as the time period since the most recent successful tracer gas test is greater than 3 years.~~ ~~(c) The first performance of the periodic measurement of CRE pressure, Specification 5.24d., shall be within the next 138 days.~~	~~The amendment is effective as of the date of its issuance and shall be implemented within 270 days of the date of issuance.~~

Fort Calhoun Station, Unit No. 1
Renewed Facility Operating License No. DPR-40

Clean Technical Specification Pages

TABLE OF CONTENTS (Continued)

DEFINITIONS

REACTOR OPERATING CONDITIONS (Continued)

Physics Testing

Testing performed under written procedures approved by Plant Operations Review Committee to determine CEA worths and other core nuclear parameters. Deviations from normal operating practice which are necessary to enable some of these tests to be performed are permitted in accordance with: 1) the specific provisions of these technical specifications, 2) authorization under the provisions of 10 CFR 50.59, or 3) other approval of the Commission.

PROTECTIVE SYSTEMS

Reactor Trip

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

Instrument Channel

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

Reactor Protective System Logic[1]

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

DEFINITIONS

Azimuthal Power Tilt - T_q

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

Maximum Radial Peaking Factor (F_R^T)

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

Dose Equivalent I-131

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

Dose Equivalent I-131 (ΦCi/gm)	=	ΦCi/gm of I-131
	+	0.0361 x ΦCi/gm of I-132
	+	0.270 x ΦCi/gm of I-133
	+	0.0169 x ΦCi/gm of I-134
	+	0.0838 x ΦCi/gm of I-135

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 Operations 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 Operations 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 2.15.1(2) is applicable.

l Specification 2.15.1(3) is applicable. If ESF Logic Subsystems A and B are inoperable, enter Specification 2.0.1.

m Steam Generator Low Pressure permissive is required for actuation.

n Auto removal of bypass prior to exceeding 600 psia.

5.0 **ADMINISTRATIVE CONTROLS**

5.2 Organization (Continued)

b. An Operator or Technician qualified in Radiation Protection Procedures shall be onsite when fuel is in the reactor.

c. All core alterations shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator limited to fuel handling who has no other concurrent responsibilities during the operation.

d. Fire protection program responsibilities are assigned to those positions and/or groups designated by asterisks in USAR 12.1-1 through 12.1-4 according to the procedures specified in Section 5.8 of the Technical Specifications.

e. The Manager - Shift Operations, the Shift Managers, and the Unit Supervisors shall hold a senior reactor operator license. The Licensed Operators shall hold a reactor operator license.

5.3 Facility Staff Qualification

5.3.1 Each member of the plant staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, with the exception of the Manager - Radiation Protection (MRP) and the Shift Technical Advisor (STA), the senior reactor operator licensees, and the reactor operator licensees, who shall meet the requirements set forth in Regulatory Guide 1.8, Revision 3, dated May 2000, entitled "Qualification and Training of Personnel for Nuclear Power Plants."

5.0 **ADMINISTRATIVE CONTROLS**

5.10 Record Retention

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

5.11 Radiation Protection Program

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

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

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

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

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

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

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

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

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

***At 1 meter from the radiation source or from any surface penetrated by the radiation.

5.0 ADMINISTRATIVE CONTROLS

5.16 Radiological Effluents and Environmental Monitoring Programs (Continued)

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

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

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

5.17 Offsite Dose Calculation Manual (ODCM)

Changes to the ODCM:

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

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

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

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

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

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

5.18 DELETED

5.0 **ADMINISTRATIVE CONTROLS**

5.19 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995," as modified by the following exceptions:

(1) If the Personnel Air Lock (PAL) is opened during periods when containment integrity is not required, the PAL door seals shall be tested at the end of such periods and the entire PAL shall be tested within 14 days after RCS temperature T_{cold} > 210EF.

(2) Type A tests may be deferred for penetrations of the steel pressure retaining boundary where the nominal diameter does not exceed one inch.

(3) Elapsed time between consecutive Type A tests used to determine performance shall be at least 24 months or refueling interval.

b. The containment design accident pressure (P_a) is 60 psig.

Appendix B

Additional Conditions

Renewed Facility Operating License No. DPR-40

Omaha Public Power District shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Conditions	Implementation Date

Fort Calhoun Station, Unit No. 1
Renewed Facility Operating License No. DPR-40

Mark-up of Technical Specification Bases Page for Information

[Word-processor mark-ups using "double underline/~~strikeout~~" feature for "new text/deleted text" respectively.]

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.13 Limiting Safety System Settings, Reactor Protective System (continued)

(9) Steam Generator Differential Pressure - The Asymmetric Steam Generator Transient Protection Trip Function (ASGTPTF) utilizes a trip on steam generator differential pressure to ensure that neither a DNBR of less than the minimum DNBR limit nor a fuel centerline temperature greater than the safety limit corresponding to FCM, as determined each fuel cycle and contained in the COLR, occurs as a result of the loss of load to one steam generator.

(10)Physics Testing at Low Power - During physics testing at power levels less than 10^{-1}% of rated power, the tests may require that the reactor be critical. For these tests only the low reactor coolant flow and thermal margin/low pressure trips may be bypassed below 10^{-1}% of rated power. Written test procedures which are approved by the Plant Operations Review Committee will be in effect during these tests. At reactor power levels less than 10^{-1}% of rated power the low reactor coolant flow and the thermal margin/low pressure trips are not required to prevent fuel element thermal limits being exceeded. Both of these trips are bypassed using the same bypass switch. The low steam generator pressure trip is not required because the low steam generator pressure will not allow a severe reactor cooldown if a steam line break were to occur during the tests.

References

(1) USAR, Section 14.1
(2) USAR, Section 7.2.3.3
(3) USAR, Section 7.2.3.2
(4) USAR, Section 3.6.6
(5) USAR, Section 14.6
(6) USAR, Section 14.7
(7) USAR, Section 7.2.3.1
(8) USAR, Section 3.6