(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. 255 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.

D. Fire Protection Program

Omaha Public Power District shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Safety Analysis Report for the facility and as approved in the NRC safety evaluation reports (SERs) dated February 14 and August 23, 1978; November 17, 1980; April 8 and August 12, 1982; July 3 and November 5, 1985; July 1, 1986; December 20, 1988; November 14,1990; March 17, 1993; and January 14, 1994, subject to the following provision:

> Omaha Public Power District may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

E. Updated Final Safety Analysis Report

The Omaha Public Power District Updated Final Safety Analysis Report supplement, submitted pursuant to 10 CFR 54.21 (d), describes certain future activities to be completed prior to the period of extended operation. The Omaha Public Power District shall complete these activities no later than August 9, 2013, and shall notify the NRC In writing when implementation of these activities is complete and can be verified by NRC inspection.

The Updated Final Safety Analysis Report supplement, as revised, shall be included in the next scheduled update to the Updated Final Safety Analysis Report required by 10 CFR 50.71 (e)(4) following issuance of this renewed license. Until that update is complete, the Omaha Public Power District may make changes to the programs and activities described in the supplement without prior Commission approval, provided that the Omaha Public Power District evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

F. Appendix B

The Additional Conditions contained in Appendix B, as revised through Amendment No. 255 , are hereby incorporated into this license. Omaha Public Power District shall operate the facility in accordance with the Appendix B Additional Conditions.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.4 Containment Cooling (Continued)

b. During power operation one of the components listed in (1)a.i. or ii. may be inoperable. If the inoperable component is not restored to operability within seven days, the reactor shall be placed in hot shutdown condition within 12 hours. If the inoperable component is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

c. For cases involving Raw Water pump inoperability, if the river water temperature is below 60 degrees Fahrenheit, one Raw Water pump may be inoperable indefinitely without applying any LCO action statement. When the river water temperature is greater than 60 degrees Fahrenheit, an inoperable Raw Water pump shall be restored to operability within 7 days or the reactor shall be placed in a hot shutdown condition within 12 hours. If the inoperable Raw Water pump is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

(2) Modification of Minimum Requirements

a. During power operation, the minimum requirements may be modified to allow a total of two of the components listed in (1)a.i. and ii. to be inoperable at any one time. (This does not include: 1) One Raw Water pump which may be inoperable as described above if the river water temperature is below 60 degrees Fahrenheit or, 2) SI-3A and SI-3B being simultaneously inoperable; or 3) VA-3A and VA-3B, or VA-7C and VA-7D, being simultaneously inoperable. Only two raw water pumps may be out of service during power operations. Either containment spray pump, SI-3A or SI-3B, must be operable during power operations. One train of the containment air cooling and filtering systems, (VA-3A and VA-7C), or (VA-3B and VA-7D), must be operable during power operations.) If the operability of one of the two components is not restored within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. LCO 2.4(1)b. shall be applied if one of the inoperable components is restored within 24 hours. If the operability of both components is not restored within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

b. During power operation one component cooling heat exchanger may be inoperable. If the operability of the heat exchanger is not restored within 14 days, the reactor shall be placed in a hot shutdown condition within 12 hours. If two component cooling heat exchangers are inoperable, the reactor shall be placed in hot shutdown condition within 12 hours. If the inoperable heat exchanger(s) is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

c. Any valves, interlocks and piping directly associated with one of the above components and required to function during accident conditions shall be deemed to be part of that component and shall meet the same requirements as for that component.

d. Any valve, interlock or piping associated with the containment cooling system which is not included in the above paragraph and which is required

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.4 Containment Cooling (Continued)

to function during accident conditions may be inoperable for a period of no more than 24 hours. If operability is not restored within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours.

Basis

A full capacity diesel-generator is connected to each of the two engineered safeguards 4.16-kV buses. Three engineered safeguards 480-Volt double-ended load centers are provided; of the six transformers, three are connected to each of the two 4.16-kV buses. Two load centers are operated as two-bus-section units; the third is provided with a center bus manually transferable to either associated end section. The center bus section supplies HPSI Pump SI-2C, CS Pump SI-3C and Charging Pump CH-1C any of which can thus be supplied from either 4.16-kV bus if required (note that CS pump SI-3C is connected to the center bus, however, this pump is available for manual operation only).

The containment spray pumps will start for a main steam line break (MSLB) but will not start during a loss-of-coolant accident (LOCA). The containment spray pumps initially take coolant from the safety injection and refueling water tank (SIRWT) before the pump suction is switched to the containment recirculation line. For an MSLB, with all safety injection pumps running, the SIRWT supply of water is not exhausted for at least 24 minutes[2]. One shutdown cooling heat exchanger is sufficient to satisfy the shutdown cooling requirements during the long-term containment cooling period.[3] In addition, in the unlikely event of the component cooling water supply being lost, raw water can be utilized for direct cooling of certain engineered safeguard components.[4]

The containment air cooling and filtering system is independent from the containment spray system for the containment cooling function.[5] For the limiting LOCA scenario, one of the two containment air cooling trains, comprised of one containment air cooling and filtering unit and one containment air cooling unit, would limit the containment pressure to below the design value without taking credit for the cooling capacity of the containment spray system.[6] [7] For the limiting MSLB scenario, a heat removal contribution is credited from the air coolers and the containment spray system in the mitigation of containment peak pressure.[7]

The cooling equipment provided to limit the containment pressure following a LOCA is divided between the independent power supply systems. The raw water and component cooling water pumps are similarly distributed on the 4.16-kV and 480 Volt buses. In the event of a LOCA, loss of normal power sources and failure of one diesel generator to operate, a minimum of one air cooler train would be connected to the available diesel generator. This would provide adequate containment cooling equipment to limit the containment pressure below the design value. The limiting MSLB event which results in the maximum containment pressure is not affected by the loss of one diesel generator.

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.4 Containment Cooling (Continued)

References

(1) Deleted
(2) USAR, Section 6.2.3.1
(3) USAR, Section 6.2.3.4
(4) USAR, Section 9.8.2
(5) USAR, Section 6.4.5
(6) USAR, Section 6.4.1.2
(7) USAR, Section 14.16.5
(8) Deleted
(9) USAR, Section 9.7
(10) USAR, Section 6.3
(11) USAR, Section 9.8

2.0 LIMITING CONDITIONS FOR OPERATION

2.14 Engineered Safety Features System Initiation Instrumentation Settings (Continued)

(3) Containment High Radiation (Air Monitoring) (Continued)

Effluent radiation monitor isolation function setpoints will be calculated in accordance with the ODCM. Process radiation monitor setpoints will be calculated in accordance with the applicable Chemistry Manual calibration procedure.

(4) Low Steam Generator Pressure

A signal is provided upon sensing a low pressure in a steam generator to close the main steam isolation valves in order to minimize the temperature reduction in the reactor coolant system with resultant loss of water level and possible addition of reactivity. The setting of 500 psia includes a ± 22 psi uncertainty and was the setting used in the safety analysis.[3]

Closure of the MSIVs (and the bypass valves, along with main feedwater isolation and bypass valves) is accomplished by the steam generator isolation signal which is a logical combination of low steam generator pressure or high containment pressure.

Initiation of containment spray is a logical combination of containment high pressure, pressurizer low/low pressure, and steam generator low pressure.

As part of the AFW actuation logic, a separate signal is provided to terminate flow to a steam generator upon sensing a low pressure in that steam generator if the other steam generator pressure is greater than the pressure setting. This is done to minimize the temperature reduction in the reactor coolant system in the event of a main steam-line break. The setting of 466.7 psia includes a +31.7 psi uncertainty; therefore, a setting of 435 psia was used in the safety analysis.

(5) SIRW Tank Low Level

Level switches are provided on the SIRW tank to actuate the valves in the safety injection pump suction lines in such a manner so as to switch the water supply from the SIRW tank to the containment sump for a recirculation mode of operation after a period of at least 45 minutes in response to a LOCA. The switch-over point of 16 inches above tank bottom is set to prevent the pumps from running dry during the time required to stroke the valves and to hold in reserve approximately 28,000 gallons of water of at least the refueling boron concentration. The USAR loss of coolant accident analysis[4] assumed the recirculation started when the minimum usable volume of 250,000 gallons had been pumped from the tank. Technical Specification 2.3(1) requires that the SIRW tank contain a minimum of 283,000 gallons of usable water. This additional volume over that assumed in the USAR analysis provides sufficient margin to account for the instrument uncertainty.

TABLE 2-1

ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENT SETTING LIMITS

	Functional Unit	Channel	Setting Limit
1.	High Containment Pressure	a. Safety Injection b. Containment Spray (3) c. Containment Isolation d. Containment Air Cooler DBA Mode e. Steam Generator Isolation	≤ 5 psig
2.	Pressurizer Low/Low Pressure	a. Safety Injection b. Containment Spray (3) c. Containment Isolation d. Containment Air Cooler DBA Mode	≥ 1600 psia (1)
3.	Containment High Radiation	Containment Ventilation Isolation	In accordance with the applicable Chemistry Manual calibration procedure
4.	Low Steam Generator Pressure	a. Steam Line Isolation b. Auxiliary Feedwater Actuation c. Containment Spray (3)	≥ 500 psia (2) ≥ 466.7 psia ≥ 500 psia(2)
5.	SIRW Low Level Switches	Recirculation Actuation	16 inches +0, -2 in. above tank bottom

TABLE 2-1 (continued)

ENGINEERED SAFETY FEATURES SYSTEM INITIATION INSTRUMENT SETTING LIMITS

	Functional Unit	Channel	Setting Limit
6.	4.16 KV Emergency Bus Low Voltage	a. Loss of Voltage	(2995.2 + 104) volts } Trip ≤ 5.9 (4)-20.8 seconds }
		b. Degraded Voltage	
		i) Bus 1A3 Side	≥ 3988.8 volts } Trip (4.8 ± .5) seconds }
		ii) Bus 1A4 Side	≥ 3990.6 volts } Trip (4.8 ± .5) seconds
7.	Low Steam Generator Water Level	Auxiliary Feedwater Actuation	≥ 28.2% of wide range tap span
8.	High Steam Generator Delta Pressure	Auxiliary Feedwater Actuation	≤ 119.7 psid

(1) May be bypassed below 1700 psia and is automatically reinstated prior to exceeding 1700 psia.
(2) May be bypassed below 600 psia and is automatically reinstated prior to exceeding 600 psia.
(3) Simultaneous containment high pressure, pressurizer low/low pressure, and steam generator low pressure.
(4) Applicable for bus voltage ≤ 2995.2 - 20.8 volts only. (For voltage ≥ (2995.2 - 20.8) volts, time delay shall be > 5.9 seconds).

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	(f)
	Logic Subsystem B	2[(a)(d)(l)]	1	Test	
C	Pressurizer Low/Low Pressure				
	Logic Subsystem A	2[(a)(d)(l)]	1	Reactor Coolant(f)	
	Logic Subsystem B	2[(a)(d)(l)]	1	Pressure Less Than 1700 psia[(b)]	
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	(f)
	Logic Subsystem B	2[(a)(c)(d)(l)]	1	Test	
C	Pressurizer Low/Low Pressure				
	Logic Subsystem A	2[(a)(c)(d)(l)]	1	Reactor Coolant(f)	
	Logic Subsystem B	2[(a)(c)(d)(l)]	1	Pressure Less Than 1700 psia[(b)]	
D	Steam Generator Low Pressure				
	Logic Subsystem A	2/Steam Gen[(a)(c)(d)(l)]	1/Steam Gen	Steam Generator Pressure Less Than	(f)
	Logic Subsystem B	2/Steam Gen[(a)(c)(d)(l)]	1/Steam Gen	600 psia [(n)]	
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(f)	
	-Degraded Voltage	2[(a)(d)]	1	Temperature Less Than 300° F	

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(2) is applicable.

e Control switch on incoming breaker.

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

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

h Specification 2.15(1) is applicable.

TABLE 2-3
(Continued)

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

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

k Specification 2.15(2) is applicable.

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

m Steam Generator Low Pressure permissive is required for actuation.

n Auto removal of bypass prior to exceeding 600 psia.

TABLE 2-4

Instrument Operating Conditions for Isolation Functions

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
1	Containment Isolation				
A	Manual	1	None	None	N/A
B	Containment High Pressure				
	Logic Subsystem A	2[(a)(e)(g)]	1	During Leak	(f)
	Logic Subsystem B	2[(a)(e)(g)]	1	Test	
C	Pressurizer Low/Low Pressure				
	Logic Subsystem A	2[(a)(e)(g)]	1	Reactor Coolant(f)	
	Logic Subsystem B	2[(a)(e)(g)]	1	Pressure Less Than 1700 psia[(b)]	
2	Steam Generator Isolation				
A	Manual	1	None	None	N/A
B	Steam Generator Isolation	1	None	None	N/A
	(i) Steam Generator Low Pressure				
	Logic Subsystem A	2/Steam Gen[(a)(e)(g)]	1/Steam Gen	Steam Generator Pressure Less Than 600 psia[(c)]	(f)
	Logic Subsystem B	2/Steam Gen[(a)(e)(g)]	1/Steam Gen		
	(ii) Containment High Pressure				
	Logic Subsystem A	2[(a)(e)(g)]	1	During Leak	(f)
	Logic Subsystem B	2[(a)(e)(g)]	1	Test	
3	Ventilation Isolation				
A	Manual	1	None	None	N/A
B	Containment High Radiation				
	Logic Subsystem A	1[(d)(g)]	None	If Containment	(f)
	Logic Subsystem B	1[(d)(g)]	None	Relief and Purge Valves are Closed	

a Circuits on ESF Logic Subsystems A and B each have 4 channels.
b Auto removal of bypass prior to exceeding 1700 psia.
c Auto removal of bypass prior to exceeding 600 psia.

TABLE 3-2 (continued)

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

Channel Description	Surveillance Function	Frequency	Surveillance Method
4. Containment Pressure High Signal	a. Test	Q	a. CHANNEL FUNCTIONAL TEST
	b. Calibrate	R	b. CHANNEL CALIBRATION
5. Containment Spray Actuation Logic	a. Test	Q	a. CHANNEL FUNCTIONAL TEST (Simulation of PPLS, CPHS, and SGLS(8) 2/4 Logic)
	b. Test	R(7)	b. CHANNEL FUNCTIONAL TEST
6. Containment Radiation High Signal (2)	a. Check	D	a. CHANNEL CHECK

TABLE 3-2 (continued)

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

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

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

3.0 **SURVEILLANCE REQUIREMENTS**
3.5 Containment Tests (Continued)

A-10	C-7	E-4	F-11
A-11	C-8	E-5	G-1
B-1	C-9	E-6	G-2
B-2	C-10	E-7	G-3
B-4	C-11	E-8	G-4
B-5	D-1	E-9	H-1
B-6	D-2	E-10	H-2
B-7	D-4	E-11	H-3
B-8	D-5	F-1	H-4

(4) Containment Isolation Valves Leak Rate Tests (Type C Tests)

Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program for the following penetrations:

M-2	M-31	M-52	IA-3092
M-7	M-38	M-53	IA-3093
M-8	M-39	M-57	IA-3094
M-11	M-40	M-58	
M-14	M-42	M-69	
M-15	M-43	M-73	
M-18	M-44	M-74	
M-19	M-45	M-79	
M-20	M-46	M-80	
M-22	M-47	M-86	
M-24	M-48	M-87	
M-25	M-50	M-88	
M-30	M-51	M-89	
		M-HCV-383-3	
		M-HCV-383-4	

3.0 **SURVEILLANCE REQUIREMENTS**

3.6 Safety Injection and Containment Cooling Systems Tests (Continued)

(3) Containment Recirculating Air Cooling and Filtering System

a. The emergency mode dampers will be verified to be in their accident positions and the automatic valve, fan, and fusible link automatic damper operation will be checked for operability on a refueling surveillance interval.

b. Each fan required to function during accident conditions will be exercised at intervals not to exceed three months.

c. Each air filtering circuit will be operated at least 10 hours every month.

d. A visual examination of the HEPA and charcoal filters will be made on a refueling surveillance interval to ensure that leak paths do not exist.

e. Measurement of pressure drop across the HEPA filter bank shall be performed on a refueling surveillance interval to verify a pressure drop of less than 2 inches of water at system design flow. Measurement of pressure drop across the combined HEPA and charcoal adsorber banks shall be performed on a refueling surveillance interval to verify a pressure drop of less than 6 inches of water at system design flow.

f. Fans shall be shown to operate within +/-10% design flow on a refueling surveillance interval.

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.
255	(2) This license shall be deemed to contain the following specified conditions and specified actions: (a) During the 2008 RFO [refueling outage], replace the CACF [containment air cleaning and filtering] unit HEPA [high-efficiency particulate air] filters not previously replaced during 2006; (b) During the 2008 RFO, ensure all CACF unit HEPA filters meet the 2” wc [water column] differential pressure limitation; (c) Implement and maintain the newly created HEPA filter replacement criteria in Procedure PE-RR-VA-0209 unless superseded by other criteria via the license amendment process; and (d) Submit a license amendment request (LAR) by October 31, 2008, that will add the HEPA filter testing and replacement criteria to the FCS [Fort Calhoun Station] Technical Specifications.	The amendment is effective as of its date of issuance and shall be implemented prior to startup from the 2008 refueling outage.
255	(3) This license shall be deemed to contain the following specified conditions and specified actions: (a) During the 2008 RFO, perform the surveillance operability testing of the containment cooling unit relief ports; and (b) Submit a LAR by October 31, 2008, that will add surveillance operability and testing of the containment cooling unit relief ports to the FCS Technical Specifications.	The amendment is effective as of its date of issuance and shall be implemented prior to startup from the 2008 refueling outage.