Template # NRR-058

February 10, 2000

Mr. Otto L. Maynard
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, KA 66839

SUBJECT: WOLF CREEK GENERATING STATION - CORRECTION TO AMENDMENT NO. 131 (TAC NO. MA7043)

Dear Mr. Maynard:

On December 16, 1999, the Commission issued Amendment No. 131 to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment was in response to your application dated November 8, 1999.

The amendment corrected 15 errors in the improved Technical Specifications that was issued in Amendment No. 123 on March 31, 1999. In addition, there were 4 corrections to Table LG, "Details Relocated from Current Technical Specifications [CTS]," that was attached to the safety evaluation dated March 31, 1999, issued with Amendment No. 123.

Due to an administrative error, the technical specifications were incorrectly issued with Amendment No. 131. Enclosed is a corrected version of the technical specifications with the errata page. We apologize for any inconvenience this may have caused.

Sincerely,

/RA/

Jack N. Donohew, Senior Project Manager, Section 2
Project Directorate IV and Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures: 1. Errata Page
2. Technical Specifications Pages

cc w/encls: See next page

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Wolf Creek Generating Station

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ATTACHMENT TO LICENSE AMENDMENT NO. 131

FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Replace the following pages of the Appendix A Improved Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
ii	ii
1.1-3	1.1-3
3.2-6	3.2-6
3.2-7	3.2-7
3.2-8	3.2-8
3.3-15	3.3-15
3.3-29	3.3-29
3.3-30	3.3-30
3.4-17	3.4-17
3.4-18	3.4-18
3.6-7	3.6-7
3.6-12	3.6-12
3.7-22	3.7-22
3.9-6	3.9-6
4.0-1	4.0-1
5.0-3	5.0-3
5.0-19	5.0-19
5.0-20	5.0-20
5.0-25	5.0-25

1.1 Definitions (continued)

Ē - AVERAGE DISINTEGRATION ENERGY (continued)	gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME	The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.
LEAKAGE	LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;

(continued)

1.1 Definitions (continued)

LEAKAGE (continued)	c. Pressure Boundary LEAKAGE LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE--OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, of the USAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued)	Once within 24 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified AND 31 EFPD thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------------NOTE------------- Required Actions A.2 and A.3 must be completed whenever Condition A is entered. --------------------------------- $F^N_{\Delta H}$ not within limit.	A.1.1 Restore $F^N_{\Delta H}$ to within limit.	4 hours
	<u>OR</u>	
	A.1.2.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
	<u>AND</u>	
	A.1.2.2 Reduce Power Range Neutron Flux - High trip setpoints to ≤ 55% RTP.	72 hours
	<u>AND</u>	
	A.2 Perform SR 3.2.2.1.	72 hours
	<u>AND</u>	
		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 --------------NOTE------------ THERMAL POWER does not have to be reduced to comply with this Required Action. ---------------------------------- Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP AND Prior to THERMAL POWER exceeding 75% RTP AND 24 hours after THERMAL POWER reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
During power escalation following shutdown, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter

Table 3.3.1-1 (page 1 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE (a)
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA
	$3^{(b)}$, $4^{(b)}$, $5^{(b)}$	2	C	SR 3.3.1.14	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 112.3% RTP
b. Low	$1^{(c)}$,2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 28.3% RTP
3. Power Range Neutron Flux Rate					
a. High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.3 % RTP with time constant ≥ 2 sec
b. High Negative Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤6.3% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	$1^{(c)}$, $2^{(d)}$	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 35.3% % RTP
5. Source Range Neutron Flux	$2^{(e)}$	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.6 E5 cps
	$3^{(b)}$, $4^{(b)}$, $5^{(b)}$	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.6 E5 cps

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
(c) Below the P-10 (Power Range Neutron Flux) interlock.
(d) Above the P-6 (Intermediate Range Neutron Flux) interlock.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlock.

Table 3.3.1-1 (page 2 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE (a)
6. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1 (Page 3.3-19)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 2 (Page 3.3-20)
8. Pressurizer Pressure					
a. Low	1(g)	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 1931 psig
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2400 psig
9. Pressurizer Water Level - High	1(g)	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.9% of instrument span
10. Reactor Coolant Flow - Low	1(g)	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 88.9%(m)
11. Not Used.					
12. Undervoltage RCPs	1(g)	2/bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 10355 Vac

(continued)

(a) The Allowable Value defines the Limiting Safety System Setting. See the Bases for the Trip Setpoints.
(g) Above the P-7 (Low Power Reactor Trips Block) interlock.
(m) % of design flow - 90,324 gpm.

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.2.3	---NOTE--- The continuity check may be excluded. --- Perform ACTUATION LOGIC TEST.	 31 days on a STAGGERED TEST BASIS
SR 3.3.2.4	Perform MASTER RELAY TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.2.5	Perform COT.	92 days
SR 3.3.2.6	---NOTE--- Not applicable to slave relays K602, K620, K622, K624, K630, K740, and K741. --- Perform SLAVE RELAY TEST.	 92 days
SR 3.3.2.7	---NOTE--- Verification of relay setpoints not required. --- Perform TADOT.	 18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.8	----------NOTE---------- Verification of setpoint not required for manual initiation functions. ------------------------------ Perform TADOT.	18 months
SR 3.3.2.9	----------NOTE---------- This Surveillance shall include verification that the time constants are adjusted to the prescribed values. ------------------------------ Perform CHANNEL CALIBRATION.	18 months
SR 3.3.2.10	----------NOTE---------- Not required to be performed for the turbine driven AFW pump until 24 hours after SG pressure is ≥ 900 psig. ------------------------------ Verify ESF RESPONSE TIMES are within limits.	18 months on a STAGGERED TEST BASIS
SR 3.3.2.11	----------NOTE---------- Verification of setpoint not required. ------------------------------ Perform TADOT.	18 months
SR 3.3.2.12	Perform COT.	31 days

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

------------------------------------NOTES------------------------------------

1. All RHR pumps may be removed from operation for ≤ 1 hour provided:

    a. The core outlet temperature is maintained at least 10°F below saturation temperature;

    b. No operations are permitted that would cause a reduction of the RCS boron concentration; and

    c. Reactor vessel water level is above the vessel flange.

2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

------------------------------------NOTE------------------------------------

While this LCO is not met, entry into MODE 5 Loops Not Filled from MODE 5 Loops Filled is not permitted.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR loop inoperable.	A.1	Initiate action to restore RHR loop to OPERABLE status.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required RHR loops inoperable. OR No RHR loop in operation	B.1	Suspend all operations involving reduction in RCS boron concentration.	Immediately
	AND		
	B.2	Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify one RHR loop is in operation.	12 hours
SR 3.4.8.2	Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	7 days

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTES-----

1. Penetration flow path(s) except for containment shutdown purge valve flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----NOTE----- Only applicable to penetration flow paths with two containment isolation valves. ----- One or more penetration flow paths with one containment isolation valve inoperable except for purge valve leakage not within limit.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	4 hours (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 -----------NOTES----------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. ------------------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	------------NOTES------------ 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. ---------------------------------	
		Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
	AND		
	D.3	Perform SR 3.6.3.6 or SR 3.6.3.7 for the resilient seal purge valves closed to comply with Required Action D.1.	Once per 92 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each containment shutdown purge valve is sealed closed or closed and blind flange installed except for one purge valve in a penetration flow path while in Condition D of this LCO.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
SR 3.6.3.2	Verify each containment mini-purge valve is closed, except when the containment mini-purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.	31 days

(continued)

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Ventilation System (CREVS)

LCO 3.7.10 Two CREVS trains shall be OPERABLE.

-----NOTE-----
The control room boundary may be opened intermittently under administrative controls.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CREVS train inoperable.	A.1	Restore CREVS train to OPERABLE status.	7 days
B. Two CREVS trains inoperable due to inoperable control room boundary in MODES 1, 2, 3, and 4.	B.1	Restore control room boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 5.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	D.1.1	Place OPERABLE CREVS train in CRVIS mode.	Immediately
	AND		
	D.1.2	Verify OPERABLE CREVS train is capable of being powered by an emergency power source.	Immediately
	OR		
	D.2.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	D.2.2	Suspend movement of irradiated fuel assemblies.	Immediately
E. Two CREVS trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	E.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	E.2	Suspend movement of irradiated fuel assemblies.	Immediately
F. Two CREVS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1	Enter LCO 3.0.3.	Immediately

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts;

b. One door in the emergency air lock closed and one door in the personnel air lock capable of being closed; and

-----------------------------------NOTE-----------------------------------
An emergency personnel escape air lock temporary closure device is an acceptable replacement for an emergency air lock door.
----

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge Isolation valve.

-----------------------------------NOTE-----------------------------------
Penetration P-63 (Service Air valves KAV-039 and KAV-118) and Penetration P-98 (Breathing Air valves KB V-001 and KB V-002) may be unisolated under administrative controls.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status except for containment penetrations P-63 and P-98 that are open under administrative controls.	7 days
SR 3.9.4.2	Verify each required containment purge isolation valve actuates to the isolation position on an actual or simulated actuation signal.	18 months

4.0 DESIGN FEATURES

4.1 Site Location

4.1.1 The WCGS site is approximately 3.5 miles east of the John Redmond Reservoir in Coffey County, Kansas and is approximately 3.5 miles northeast of the town of Burlington.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or ZIRLO clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control rod material shall be silver indium cadmium or hafnium metal as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.0 weight percent. For fuel with enrichments greater than 4.6 nominal weight percent of U-235, the combination of enrichment and integral fuel burnable absorbers shall be sufficient so that the requirements of 4.3.1.1.b are met.

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1A of the USAR;

c. A nominal 8.99 inch center to center distance between fuel assemblies placed in the fuel storage racks;

d. Partially spent fuel assemblies with a discharge burnup in the "Acceptable Burnup Domain for Region 2 and 3 Storage" of Figure 3.7.17-1 may be allowed unrestricted storage in acceptable fuel storage locations;

e. Partially spent fuel assemblies with a discharge burnup in the "Acceptable Burnup Domain for Region 3 Storage" of Figure 3.7.17-1 may be allowed unrestricted storage in acceptable fuel storage locations, except in Region 2 locations in a Mixed Zone Three Region configuration; and

f. New or partially spent fuel assemblies with a discharge burnup in the "Unacceptable Burnup Domain for Region 2 and 3 Storage" of Figure 3.7.17-1 will be stored in Region 1 locations.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the USAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the USAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

(continued)

5.2 Organization

5.2.2 Unit Staff (continued)

shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. An individual from the Health Physics Group qualified in radiation protection procedures shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. Administrative procedures shall be developed and implemented to limit the working hours of personnel who perform safety related functions (e.g., licensed Senior Reactor Operator (SROs), licensed Reactor Operator (ROs), health physics technicians, nuclear station operators, and key maintenance personnel).

The controls shall include guidelines on working hours that ensure adequate shift coverage shall be maintained without routine heavy use of overtime. Any deviation from the above guidelines shall be authorized in advance by the Plant Manager or the Plant Manager's designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Routine deviation from the working hour guidelines shall not be authorized.

Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned.

e. The Superintendent Operations or Manager Operations shall hold an SRO license.

f. An individual shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This position shall be manned in MODES 1, 2, 3 or 4, unless the Shift Manager or the individual with a Senior Operator License meets the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

5.0 ADMINISTRATIVE CONTROLS

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 with the following exceptions:

5.3.1.1 Licensed Operators and Senior Operators shall meet or exceed the qualifications of ANSI/ANS 3.1-1981 as endorsed by Regulatory Guide 1.8, Revision 2, and 10 CFR Part 55.

5.3.1.2 The position of Manager Chemistry/Radiation Protection shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 for a Radiation Protection Manager.

5.3.1.3 The position of Manager Operations shall hold or have previously held a senior reactor operator license for a similar unit (PWR).

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal absorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2 at the system flowrate specified below ± 10%.

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation System – Filtration	2000 cfm
Control Room Emergency Ventilation System-Pressurization	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	6500 cfm

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal absorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and greater than or equal to the relative humidity specified below.

ESF Ventilation System	Penetration	RH
Control Room Emergency Ventilation System (Filtration/Pressurization)	2%	70%
Auxiliary/Fuel Building Emergency Exhaust	2%	70%

d. Demonstrate at least once per 18 months for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal absorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system flowrate specified below ± 10%.

ESF Ventilation System	Delta P	Flowrate
Control Room Emergency Ventilation System - Filtration	6.6 in. W.G.	2000 cfm
Control Room Emergency Ventilation System - Pressurization	3.6 in. W.G.	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	4.7 in. W.G.	6500 cfm

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

e. Demonstrate at least once per 18 months that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ANSI N510-1975.

ESF Ventilation System	Wattage
Control Room Emergency Ventilation System - Pressurization	5 ± 1 kW
Auxiliary/Fuel Building Emergency Exhaust	37 ± 3 kW

f. Demonstrate at least once per 18 months for each of the ESF systems that following the creation of an artificial Delta P across the combined HEPA filters, the prefilters, and the charcoal absorbers of not less than the value specified below (dirty filter conditions), that the flowrate through these flow paths is with ± 10% of the value specified below when tested in accordance with ANSI N510-1980.

ESF Ventilation System	Delta P	Flowrate
Control Room Filtration System	6.6 in. W.G.	2000 cfm
Control Room Pressurization System	3.6 in. W.G.	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	4.7 in. W.G.	6500 cfm

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, Revision 0, July 1981, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Revision 2, July 1981, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."

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5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in a format similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

(continued)

5.6 Reporting Requirements (continued)

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Specification 3.1.3: Moderator Temperature Coefficient (MTC),
2. Specification 3.1.5: Shutdown Bank Insertion Limits,
3. Specification 3.1.6: Control Bank Insertion Limits,
4. Specification 3.2.3: Axial Flux Difference,
5. Specification 3.2.1: Heat Flux Hot Channel Factor, $F_Q(Z)$,
6. Specification 3.2.2: Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$),
7. Specification 3.9.1: Boron Concentration, and
8. SHUTDOWN MARGIN for Specification 3.1.1 and 3.1.4, 3.1.5, 3.1.6, and 3.1.8.

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