



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

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 131
License No. NPF-42

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment to the Wolf Creek Generating Station (the facility) Facility Operating License No. NPF-42 filed by the Wolf Creek Nuclear Operating Corporation (the Corporation), dated November 8, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-42 is hereby amended to read as follows:

2. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 131 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented December 31, 1999, the same time Amendment No. 123 issued March 31, 1999, is implemented.

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment: Changes to the Technical
Specifications

Date of Issuance: December 16, 1999

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.

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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)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.2.1.2 (continued)	<p>Once within 24 hours after achieving equilibrium conditions after exceeding, by $\geq 10\%$ RTP, the THERMAL POWER at which F₀^W(Z) was last verified</p> <p><u>AND</u></p> <p>31 EFPD thereafter</p>

ACTIONS (continued)

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

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.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.3	<p>-----NOTE----- The continuity check may be excluded. -----</p> <p>Perform ACTUATION LOGIC TEST.</p>	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	<p>-----NOTE----- Not applicable to slave relays K602, K620, K622, K624, K630, K740, and K741. -----</p> <p>Perform SLAVE RELAY TEST.</p>	92 days
SR 3.3.2.7	<p>-----NOTE----- Verification of relay setpoints not required. -----</p> <p>Perform TADOT.</p>	18 months

(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)

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
<p>A. -----NOTE----- Only applicable to penetration flow paths with two containment isolation valves.</p> <p>One or more penetration flow paths with one containment isolation valve inoperable except for purge valve leakage not within limit.</p>	<p>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.</p> <p><u>AND</u></p>	<p>4 hours</p> <p>(continued)</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)	<p>D.2</p> <p>-----NOTES-----</p> <ol style="list-style-type: none"> 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. <p>-----</p> <p>Verify the affected penetration flow path is isolated.</p> <p><u>AND</u></p> <p>D.3</p> <p>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.</p>	<p>Once per 31 days for isolation devices outside containment</p> <p><u>AND</u></p> <p>Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment</p> <p>Once per 92 days</p>

(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
	<u>AND</u> C.2 Be in MODE 5.	36 hours

(continued)

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.

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)

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.
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5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

- b. Demonstrate for each of the ESF systems that an in-place 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 \pm 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 \leq 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 \pm 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)

5.0 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)