August 12, 1994

Docket No. 50-482

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WReckley	OPA

Mr. Neil S. Carns
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, Kansas 66839

Dear Mr. Carns:

SUBJECT: WOLF CREEK GENERATING STATION - AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. M88129)

The Commission has issued the enclosed Amendment No. 76 to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated October 27, 1993.

The amendment revises Technical Specification 4.6.1.2.a, Overall Integrated Containment Leakage Rate, to provide one-time relief from the requirements to perform the surveillance at intervals of 40 months plus or minus 10 months. The schedule for the third Type A test is extended to the eighth refueling outage, approximately 54 months after the second test, in order to have it coincide with the 10-year inservice inspections. The staff has addressed your requested exemption from Appendix J of 10 CFR 50 related to the third test exceeding the first 10-year service period separately.

A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
Original Signed By

William D. Reckley, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 76 to NPF-42
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	PDIV-2/LA	PDIV-2/PM	NRR/SCSB	OGC	PDIV-2/D
NAME	EPeyton	WReckley:ye	RBarrett	EHoller	TQuay
DATE	5/27/94	6/ 1/94	6/20/94	8/ 1/94	8/10/94

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cc w/enclosures:
Jay Silberg, Esq.
Shaw, Pittman, Potts & Trowbridge
2300 N Street, NW
Washington, D.C. 20037

Mr. C. John Renken
Policy and Federal Department
Missouri Public Service Commission
P. O. Box 360
Jefferson City, Missouri 65102

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 311
Burlington, Kansas 66839

Chief Engineer
Utilities Division
Kansas Corporation Commission
1500 SW Arrowhead Road
Topeka, Kansas 66604-4027

Office of the Governor
State of Kansas
Topeka, Kansas 66612

Attorney General
1st Floor - The Statehouse
Topeka, Kansas 66612

County Clerk
Coffey County Courthouse
Burlington, Kansas 66839

Mr. Gerald Allen
Public Health Physicist
Bureau of Air & Radiation
Division of Environment
Kansas Department of Health
and Environment
Forbes Field Building 283
Topeka, Kansas 66620

Mr. Otto Maynard
Director Plant Operations
Wolf Creek Nuclear Operating Corporation
P. O. Box 411
Burlington, Kansas 66839

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Mr. Keven J. Moles
Manager Regulatory Services
Wolf Creek Nuclear Operating Corporation
P. O. Box 411
Burlington, Kansas 66839

U.S. Nuclear Regulatory Commission
Resident Inspectors Office
8201 NRC Road
Steedman, Missouri 65077-1302

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
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 October 27, 1993, 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. 76, 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 within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Director
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: August 12, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 76

FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Replace the following pages of the Appendix A 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
3/4 6-2a	3/4 6-2a
B 3/4 6-1	B 3/4 6-1

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month* intervals during shutdown at a pressure not less than either P_a, 48 psig, or P_t, 24 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

*A one time extension of the test interval is allowed for the third Type A test of the first 10-year service period, provided that unit shutdown occurs no later than March 31, 1996, and performance of the Type A test occurs prior to unit restart following the eighth refueling outage.

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or 0.75 L_t, as applicable, during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

For reduced pressure tests, the leakage characteristics yielded by measurements L_{tm} and L_{am} shall establish the maximum allowable test leakage rate L_t of not more than L_a (L_{tm}/L_{am}). In the event L_{tm}/L_{am} is greater than 0.7, L_t shall be specified as equal to L_a $(P_t/P_a)^{1/2}$

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.*

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

---

*A one time extension of the test interval is allowed for the third Type A test of the first 10-year service period, as required by Surveillance Requirement 4.6.1.2.a and by Section III.D.1.(a) of Appendix J of 10 CFR 50, provided unit shutdown occurs no later than March 31, 1996 and performance of the Type A test occurs prior to unit restart following the eighth refueling outage.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.0 psig, and (2) the containment peak pressure does not exceed the design pressure of 60 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 48.9 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 50.4 psig, which is less than design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained in accordance with safety analysis requirements for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 50.4 psig in the event of a steam line break accident. The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of proposed Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," April 1979, and proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerance on cracking, the results of the engineering evaluation and the corrective actions taken.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By application dated October 27, 1993, Wolf Creek Nuclear Operating Corporation (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-42) for the Wolf Creek Generating Station. The proposed changes would revise Technical Specification 4.6.1.2.a, Overall Integrated Containment Leakage Rate, to provide one-time relief from the requirements to perform the surveillance at intervals of 40 months plus or minus 10 months. The schedule for the third Type A test is extended to the eighth refueling outage, approximately 54 months after the second test, in order to have it coincide with the 10-year inservice inspections.

2.0 BACKGROUND

Section III.D.1(a) of Appendix J to 10 CFR Part 50 establishes the required retest schedule for Type A, Overall Integrated Containment Leakage Rate tests. The rule states that after the preoperational leakage rate tests, a set of three Type A tests shall be performed, at approximately equal intervals during each 10-year service period. The third test of each set is required to be performed when the plant is shutdown for the 10-year plant inservice inspections.

The requirements of Appendix J are reflected in the test schedule included in Technical Specification 4.6.1.2.a. The technical specification requires that three Type A tests shall be conducted at 40 +/- 10 month intervals during each 10-year service period. The third test of each set is required to be conducted during the shutdown for the 10-year plant inservice inspection.

The licensee is proposing to extend the interval between the second and third Type A tests to approximately 54 months. The current refueling outage schedules for Wolf Creek Generating Station cannot support the schedule established in Appendix J and technical specifications. In order to perform three tests in the first 10-year service period, the licensee would need to perform the third test during the seventh refueling outage. However, the 10-year inservice inspections mentioned in Appendix J and technical specifications are scheduled for the eighth refueling outage. In order to meet all requirements of the rule and technical specifications, the licensee would need to perform a fourth test during the eighth refueling outage. The proposed technical specification change and related exemption request being addressed by separate correspondence, are intended to resolve the discrepancies between the requirements and refueling outage schedules.

3.0 EVALUATION

The intent of the established test interval is that three approximately equally spaced Type A tests be conducted within a given 10-year service period. The Appendix J and technical specification retest requirements do not coincide with the anticipated refueling outage schedules such that Type A tests would need to be performed during both the seventh and eighth refueling outages resulting in a total of four tests for the first 10-year inservice inspection period. This additional testing, resulting solely from the circumstances of the refueling outage schedules, is contrary to the intent of the regulations and existing technical specifications.

The licensee has stated that the results of previous Type A tests indicate that an extension of the maximum test interval by approximately four months would not jeopardize the ability of the containment to maintain leakage at or below the applicable limit. The preoperational ILRT was performed during January 1985 and the first two tests during this service period were performed October 1988 and September 1991. The as-left leakage rates measured during the two tests were 0.112 and 0.070 wt.%/day respectively. These are well below the acceptance limit of 0.15 wt.%/day. The licensee has found that the majority of the leakage detected during the ILRTs was from the containment penetrations and not from the containment barrier itself. Local leak rate testing of penetrations will continue to be performed as required by technical specifications and can be relied upon to detect the most probable sources of containment leakage. The licensee would be required by Appendix J, Section IV.A to perform additional testing to demonstrate containment integrity if any major modifications affecting containment are performed prior to the proposed test during the eighth refueling outage.

Based on the past Type A test results, the continued performance of local leak rate testing, and intent of the Appendix J and technical specification requirements, the staff finds that the one-time extension of the required test interval for Type A tests would not adversely affect plant safety and is, therefore, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Kansas State Official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 64616). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: William Reckley, PDIV-2

Date: August 12, 1994