August 22, 1991

Docket No. 50-482

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Mr. Bart D. Withers
President and Chief Executive Officer
Wolf Creek Nuclear Operating Corporation
Post Office Box 411
Burlington, Kansas 66839

Dear Mr. Withers:

SUBJECT: WOLF CREEK GENERATING STATION - AMENDMENT NO. 47 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. 80431)

The Commission has issued the enclosed Amendment No. 47 to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station (WCGS). The amendment consists of changes to the Technical Specifications in response to your application dated May 14, 1991 (ET 91-0074).

The amendment modifies Technical Specification 3.1.3.4 to increase the maximum allowed control rod drop time from 2.2 to 2.7 seconds. This increase is requested in anticipation of an increase in the measured drop time because of the planned use of Westinghouse VANTAGE-5H fuel in WCGS. The VANTAGE-5H fuel design incorporates a control rod guide thimble diameter slightly smaller than the fuel design currently used.

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

Douglas V. Pickett, Project Manager
Project Directorate IV-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 47 to NPF-42
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	PDIV-2/LA	PDIV-2/PM	OGC	PDIV-2/(A)D		
NAME	EPeyton	DPickett:	J. Hull	GDick for		
DATE	8/6/91	8/7/91	8/7/91	8/20/91		

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

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 47
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 May 14, 1991, 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. 47 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Mel Fields for

George F. Dick, Jr., Acting Director
Project Directorate IV-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: August 22, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 47

FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Replace the following page of the Appendix A Technical Specifications with the enclosed page. The revised page is identified by amendment number and contains marginal lines indicating the area of change. The corresponding overleaf page is also provided to maintain document completeness.

REMOVE	INSERT
3/4 1-19	3/4 1-19

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length shutdown and control rod drop time from the physical fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the rod drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

#With K_{eff} greater than or equal to 1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 47 TO FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By letter dated May 14, 1991, the Wolf Creek Nuclear Operating Corporation, the licensee for the Wolf Creek Generating Station (WCGS), submitted an application for amendment to Facility Operating License No. NPF-42 (Ref. 1). The proposed amendment increases the maximum allowable control rod drop time from 2.2 to 2.7 seconds. This increase is anticipated to bound any increase in drop time resulting from the planned use of the Westinghouse VANTAGE-5H fuel for the next refueling of Wolf Creek. The VANTAGE-5H fuel design incorporates a control rod guide thimble diameter slightly smaller, i.e., 0.442 inch in diameter than the 0.450 inch currently used. The slightly smaller diameter will increase the hydraulic resistance which will result in an increased rod drop time.

The evaluation methodology includes analyses of loss-of-coolant accident (LOCA) and non-LOCA transients to confirm acceptability of the 0.50 second increase in the rod drop time. For non-LOCA transients, the objective is to confirm that the minimum departure from nucleate boiling ratio (DNBR) meets the safety limits and there is no increase in potential fuel rod failures. For the LOCA transients, both large- and small-break LOCAs were considered, and changes were included in hydraulic forces, hot leg switchover, and long term cooling.

2.0 EVALUATION

2.1 Non-LOCA Transients

The rod drop time affects only the fast non-LOCA transients for which a reactor trip signal will be generated. The transients are: partial and complete loss of flow, rod cluster control assembly (RCCA) withdrawal from subcritical, locked, reactor coolant pump rotor, and RCCA ejection.

2.1.1 Partial and Complete Loss of Forced Coolant Flow

The partial (2 out of 4 pumps) and complete (4 out of 4 pumps) loss of flow were reanalyzed. The results showed that the minimum DNBR remains well above the safety limit.

2.1.2 Reactor Coolant Pump Shaft Seizure

In this transient, departure from nucleate boiling (DNB) and peak primary pressure are the limiting parameters and both are very sensitive to the RCCA drop time. Reanalyses with the increased drop time demonstrated that the peak primary pressure to be 2,617 psia, which is lower than 2,750 psia, i.e., 10 percent above the design pressure of 2,500 psia. The rods in DNB are lower than 5 percent which was taken into account in offsite dose calculations and found acceptable.

2.1.3 Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition

In this transient, the rapid power increase is partially retarded by the Doppler feedback but the transient is terminated by a reactor trip on neutron flux. This transient is sensitive to the RCCA drop time due to the high rate of power increase. However, because of the low initial fuel temperature and the system's thermal capacity, the peak heat flux is less than the full power nominal value. Thus, there is a large DNB margin during the transient. Similarly, there is a large subcooling during the transient.

2.1.4 Uncontrolled RCCA Bank Withdrawal at Power

In this transient, the objective is to establish that the DNB design basis is met. Assuming 100 percent power, minimum reactivity feedback and unchanged overtemperature delta-T, the transient minimum DNBR occurs immediately after the reactor trip and remains above the safety analysis limit. There is a very small decrease due to the increased drop time to 2.7 sec.

2.1.5 Other Transients

All the non-LOCA design basis transients which appear in the Updated Safety Analysis Report (USAR) were reexamined for the increased rod insertion time. Except for those discussed above, they were either insensitive to rod drop time, no reactor trip was assumed, or the reactor trip had no effect on the critical parameter or the impact was negligible. Examples of such transients include: feedwater system malfunctions, increased steam flow, steam generator relief and safety valve opening. The safety limits will therefore not be violated for these events as a result of the increase in scram time.

2.2 LOCA and LOCA Related Analyses

Regarding LOCA and LOCA related analyses, it is significant to note that the proposed VANTAGE-5H fuel is less limiting for the LOCA analysis than the standard 17x17 fuel assembly now in the core. Therefore, the only parameter which could affect the LOCA analyses outcome is the RCCA rod drop time.

2.2.1 Large Break LOCA (LBLOCA)

The LBLOCA was carried out using the Westinghouse 1981 evaluation model BART (Ref. 2). The analysis was performed for the limiting double ended cold leg guillotine break with Cd = 0.4. The result showed that the increased RCCA rod drop time did not increase the peak clad temperature which is below 2200°F.

2.2.2 Small Break LOCA (SBLOCA)

The SBLOCA was analyzed using WFLASH, the Westinghouse small break LOCA evaluation model (Ref. 3). The results showed that the increase in the rod drop time to 2.7 seconds results in an increase of the peak cladding temperature by about 2°F to 1899°F, well within the regulatory limits. In addition, the maximum cladding oxidation and maximum hydrogen generation are within the 10 CFR 50.46 limits.

2.2.3 Post-LOCA Long-Term Core Cooling

Long-term cooling is assured if the reactor remains subcritical. Because credit is not taken for inserted control rods after a LOCA, subcriticality must be assured by boration. The evaluation model for this analysis is described in Reference 4. In this particular analysis, the increase in the rod drop time does not affect the sources of borated water. Long-term cooling after a LOCA is demonstrated for each reload design on a cycle specific basis.

3.0 <u>SUMMARY</u>

We reviewed the potential effects of an increase of the RCCA rod drop time for Wolf Creek from 2.2 to 2.7 seconds. Such an increase is anticipated from a switchover from standard 17x17 Westinghouse fuel to VANTAGE-5H fuel, which has a slightly smaller control rod guide thimble diameter. The effect of such an increase was evaluated for all USAR transients. In particular, the transients which result in a reactor trip were explicitly reanalyzed. Such analyses were carried out with accepted and approved methods. The results were found to be within corresponding regulatory requirements and we find them acceptable. We note that technical specifications require that the assumed maximum rod drop time of 2.7 seconds be verified by measurement after the next cycle loading.

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. 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 (56 FR 27051). 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.

7.0 REFERENCES

1. Letter from F. T. Rhodes, Wolf Creek Nuclear Operating Corporation to USNRC, "Revision to Technical Specification 3.1.3.4, Control Rod Drop Time," dated May 14, 1991.

2. WCAP-9561-PA, "BART-1A, A Computer Code for the Best Estimate Analyzed Reflood Transients," 1984.

3. WCAP-8200-PA, Rev. 2, "WFLASH, A FORTRAN-IV, Computer Program for Simulation of Transients in a Multi-Loop PWR," July 1974.

4. WCAP-8339, "Westinghouse Emergency Core Cooling Evaluation Summary," 1974.

Principal Contributor: Lambros Lois, SRXB, NRR

Date: August 22, 1991