Docket No. 50-482

November 1_ 1993

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. 69 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. M85311)

The Commission has issued the enclosed Amendment No. 69 to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the license and Technical Specifications in response to your application dated January 5, 1993, as supplemented by letter dated October 1, 1993.

The amendment increases the maximum allowable core power from 3411 megawatts thermal (MWt) to 3565 MWt and changes the limiting safety system settings and limiting conditions for operation related to the reactor coolant system average temperature.

A copy of the related Safety Evaluation and Notice of Issuance are also enclosed.

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

Enclosures:
1. Amendment No. 69 to NPF-42
2. Safety Evaluation
3. Notice

Office	PDIV-2/LA	PDIV-2/PM	NRR:EMCB*	NRR:EMCB*	NRR:SPLB*	NRR:SRXB*
Name	EPeyton	WReckley:ye	JStrosnider	JNorberg	CMcCracken	RJones
Date	11/3/93	11/3/93	11/1/93	10/18/93	10/28/93	10/27/93
Copy	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

Office	NRR:SCSB*	NRR:HICB*	NRR:PRPB*	OGC*	PDIV-2/D	APR4/5
Name	RBarrett	JWermiel	LCunningham	MYoung	SBlack	EAdensam
Date	10/28/93	10/26/93	10/28/93	10/25/93	11/3/93	11/3/93
Copy	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

Office	DRPW:D	ADP	NRR:D
Name	JRoe	LJCallan	TMurley
Date	11/4/93	11/5/93	11/9/93
Copy	Yes/No	Yes/No	Yes/No

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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. 69
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 January 5, 1993, as supplemented by letter dated October 1, 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.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraphs 2.C.(1) and 2.C.(2) of Facility Operating License No. NPF-42 are hereby amended to read as follows:*

   (1) Maximum Power Level

   The Operating Corporation is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100% power) in accordance with the conditions specified herein and in Attachment 1 to this license. The activities identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 69, 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 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachment:
1. Page 3 of License
2. Changes to the Technical Specifications

Date of Issuance: November 10, 1993

*Page 3 is attached, for convenience, for the composite license to reflect this change. Please remove page 3 of the existing license and replace with the attached page.

(1) Pursuant to Section 103 of the Act and 10 CFR Part 50 "Domestic Licensing of Production and Utilization Facilities," the Operating Corporation, to possess, use and operate the facility at the designated located in Coffey County, Kansas, in accordance with the procedures and limitations set forth in this license;

(2) KG&E, KCPL and KEPCO to possess the facility at the designated location in Coffey County, Kansas, in accordance with the procedures and limitations set forth in this license;

(3) The Operating Corporation, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended.

(4) The Operating Corporation, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) The Operating Corporation, 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 associated with radioactive apparatus or components; and

(6) The Operating Corporation, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I 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:

(1) <u>Maximum Power Level</u>

The Operating Corporation is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100% power) in accordance with the conditions specified herein and in Attachment 1 to this license. The activities identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

ATTACHMENT TO LICENSE AMENDMENT NO. 69

FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
1-5	1-5
2-4	2-4
2-8	2-8
2-10	2-10
3/4 2-16	3/4 2-16
B 3/4 2-3	B 3/4 2-3

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

1.23 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.24 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.25 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.26 RATED THERMAL POWER shall be a total core heat transfer rate to the reactor coolant of 3565 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.27 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.28 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.30 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SLAVE RELAY TEST

1.31 A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOURCE CHECK

1.32 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.33 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.34 THERMAL POWER shall be the total core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.35 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤112.3% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤28.3% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	2.4	0.5	0	≤4% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	2.4	0.5	0	≤4% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤35.3% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.6 \times 10^5$ cps
7. Overtemperature ΔT	7.9	4.61	2.57	See Note 1	See Note 2
8. Overpower ΔT	5.0	2.15	0.15	See Note 3	See Note 4
9. Pressurizer Pressure-Low	3.7	0.71	2.49	≥1915 psig	≥1906 psig
10. Pressurizer Pressure-High	7.5	0.71	2.49	≤2385 psig	≤2400 psig
11. Pressurizer Water Level-High	8.0	2.18	1.96	≤92% of instrument span	≤93.9% of instrument span

*RTP = RATED THERMAL POWER
**Loop design flow = 93,600 gpm

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} \left[T \left(\frac{1}{1+\tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT;
$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 6$ s, $\tau_2 = 3$ s;
$\frac{1}{1+\tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, $\tau_3 = 2$ s;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.10;
K_2	=	0.0137/°F;
$\frac{1+\tau_4 S}{1+\tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 = 16$ s, $\tau_5 = 4$ s;
T	=	Average temperature, °F;
$\frac{1}{1+\tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	581.2°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.000671;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) for $q_t - q_b$ between -25% and + 7%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -25% the ΔT Trip Setpoint shall be automatically reduced by 1.8% of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT Trip Setpoint shall be automatically reduced by 1.384% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.5% of ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 [T \left(\frac{1}{1 + \tau_6 S}\right) - T''] - f_2(\Delta I)\}$$

Where:

ΔT	=	Measured ΔT;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 6 s, τ_2 = 3 s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, τ_3 = 2 s;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
K_4	=	1.10;
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature;
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation;
τ_7	=	Time constant utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 s;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.00128/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$;
T	=	Average temperature, °F;
T''	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 581.2 °F);
S	=	Laplace transform operator, s^{-1}; and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.8% of ΔT span.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION 1.b and/or 3, above; subsequent POWER OPERATION may proceed provided that the indicated RCS total flow rate is demonstrated to be within the region of acceptable operation prior to exceeding the following THERMAL POWER levels:

   a. A nominal 50% of RATED THERMAL POWER,

   b. A nominal 75% of RATED THERMAL POWER, and

   c. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.5.1 The provisions of Specification 4.0.4 are not applicable to Specification 3.2.5.c.

4.2.5.2 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.3 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.4 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months. Within 7 days prior to performing the precision heat balance, the instrumentation used for determination of steam pressure, feedwater pressure, feedwater temperature, and feedwater venturi ΔP in the calorimetric calculations shall be calibrated.

4.2.5.5 The feedwater venturi shall be inspected for fouling and cleaned as necessary at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS Four Loops in Operation
1. Indicated Reactor Coolant System T_{avg}	≤585.0°F
2. Indicated Pressurizer Pressure	≥2220 psig*
3. Reactor Coolant System Flow Rate	≥38.4 x 10^4 GPM

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

QUADRANT POWER TILT RATIO (Continued)

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such ACTION does not correct the tilt, the margin for uncertainty on $F_Q(X,Y,Z)$ is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the Reactor Coolant System T_{avg} and the pressurizer pressure assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial USAR assumptions and have been analytically demonstrated adequate to maintain a DNBR above the safety analysis limit DNBR specified in the CORE OPERATING LIMITS REPORT (COLR) throughout each analyzed transient. The indicated T_{avg} value of 585°F and the indicated pressurizer pressure value of 2220 psig correspond to analytical limits of 587.7°F and 2205 psig respectively, with allowance for measurement uncertainty.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins completely offset any rod bow penalties. This is the margin between the correlation DNBR limit and the safety analysis limit DNBR. These limits are specified in the COLR.

The applicable values of rod bow penalties are referenced in the USAR.

When RCS flow rate and $F_{\Delta H}(X,Y)$, per Specification 3.2.3, are measured, no additional allowances are necessary prior to comparison with the limits in the COLR. Measurement uncertainties of 2.5% for RCS total flow rate and 4% for $F_{\Delta H}(X,Y)$ have been allowed for in determination of the design DNBR value.

DNB PARAMETERS (Continued)

The measurement uncertainty for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a nonconservative manner. Therefore, an inspection is performed of the feedwater venturi each refueling outage.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation specified in Table 3.2-1. This surveillance also provides adequate monitoring to detect any core crud buildup.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 69 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 January 5, 1993, as supplemented by letter dated October 1, 1993, Wolf Creek Nuclear Operating Corporation (the licensee) requested an amendment to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The proposed change would revise the Wolf Creek license and Technical Specifications to increase the maximum reactor core power level (rated thermal power) from the present value of 3411 megawatts thermal (MWt) to a revised limit of 3565 MWt. The increase in allowed core power combined with the energy added by the reactor coolant pumps would result in the proposed changes allowing Wolf Creek to operate at a nuclear steam supply system (NSSS) power of 3579 MWt. In addition to the increased core power level, the licensee has also proposed changes in the allowable operating temperatures of the reactor coolant system. The reductions in reactor coolant hot-leg temperatures are being proposed in order to reduce the potential for stress corrosion cracking of steam generator tubes.

The proposed changes represent an approximate 4.5 percent increase over the current licensed power level. The proposed temperature changes include a planned hot leg temperature reduction of 5 degrees Fahrenheit and a possible 15 degree Fahrenheit reduction which may be pursued in the future. The proposed maximum temperature reduction (15°F) results in secondary conditions that would likely require modifications to the turbine in order to gain the desired increase in electrical output. In support of the proposed changes, the licensee provided the results of analyses and evaluations performed to determine the impact of the changes in power level and operating temperature on the NSSS and balance of plant (BOP). Analysis of the core thermal-hydraulic response to various potential accidents determined that the 5°F hot leg temperature reduction was necessary in order to meet safety limit design criteria. The October 1, 1993, letter provided additional information in response to a request by the staff and did not propose additional changes from those described in the Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for Hearing published in the Federal Register (58 FR 26565 dated May 4, 1993).

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2.0 EVALUATION

2.1 Nuclear Steam Supply System (NSSS)

The licensee termed the combination of the higher core power level and reduced reactor coolant hot leg temperature as the proposed rerate conditions of the Wolf Creek Generating Station. The scope of the licensee's review to support the rerate conditions encompassed all aspects of the Wolf Creek NSSS design and operations affected by the proposed changes. NSSS designs were reviewed to verify compliance at the rerated conditions with licensing criteria and standards currently specified in the Wolf Creek operating license. The structural design of the NSSS equipment was reviewed to assure that compliance with industry codes and standards had been maintained at the rerated conditions. The review encompassed the verification that the NSSS components and systems will continue to meet functional requirements specified in the Updated Safety Analysis Report (USAR). Currently approved analytical methods were used for the analyses at the rerated conditions.

In addition to evaluating the ability of the NSSS to operate at the rerated conditions during normal operation, the licensee reanalyzed the design basis transients and accidents which the staff utilizes to determine the adequacy of safety margins. The licensee submitted these analyses in support of License Amendment Number 61 which included revised limits for reactor coolant flow rate, core peaking factors, and administrative processes related to Wolf Creek's seventh operating cycle. Although Amendment 61 did not include the rerated core power level or reduced hot leg temperatures, the supporting analyses for this previously issued amendment included assumptions for the limiting rerate conditions. The staff findings related to Amendment 61 in terms of the design basis transients and accidents have been determined to be bounding for the proposed rerate conditions.

Core Design

On March 30, 1993, the NRC issued License Amendment 61 for Wolf Creek Generating Station authorizing the changes requested by the licensee for the unit's seventh operating cycle. The supporting analyses were performed assuming the limiting rerate conditions and therefore remain applicable for the proposed changes being addressed by this safety evaluation. The specific changes related to changing the plant's operating conditions to the power level and temperatures associated with the rerate include changes to the definition of rated thermal power, overtemperature and overpower delta-T setpoints, and maximum reactor coolant average temperature.

Overpressure Protection

Pressurizer safety valves are required to be designed with sufficient capacity to prevent the pressurizer pressure from exceeding 110 percent of design pressure following the worst reactor coolant system (RCS) pressure transient. For purposes of analytical justification, this event is specified to be a turbine trip. No credit is taken for operation of reactor coolant system relief valves, steam line relief valves, steam dump valves, pressurizer level or pressure control systems, or direct reactor trip on turbine trip. The

Safety Evaluation Report for Wolf Creek (NUREG-0881) was based upon the criteria provided in the Standard Review Plan (NUREG-0800) in that the safety valve capacity had been shown to be adequate assuming a reactor trip occurred on the second safety-grade signal received following the turbine trip. In the licensee's submittal, the analysis which demonstrated adequate overpressure protection credited the pressurizer high pressure reactor trip which is the first trip signal generated except for the anticipatory reactor trip on turbine trip. This assumption was consistent with the analysis presented in the original USAR. However, in order to resolve the discrepancy between the various licensing basis documents, the licensee has committed to submit information in a future update of the USAR which will document the plant specific analyses related to overpressure protection and adequacy of relief capacity assuming the rerated conditions and reactor trip upon receipt of the second safety-grade signal. The licensee has indicated that assurance that the existing overpressure protection is adequate to meet the more conservative analytical criteria is provided by sensitivity analyses performed for similar plants which have concluded that the safety valve capacity is not exceeded if the reactor trip is delayed until actuation of the second safety-grade signal. In addition, the original sizing calculation for the pressurizer safety valves was based on a reactor power equivalent to 102 percent of the proposed rerate power level. Based upon the licensee's evaluations, related analyses for similar plants, and the licensee's commitment to submit confirmatory documentation, the staff concludes that compliance with General Design Criteria (GDC) 15, Reactor Coolant System Design, and other applicable regulatory requirements is maintained during operation at the rerate conditions.

The licensee stated that the low temperature overpressure protection function was not affected by the proposed rerate changes. This conclusion is based upon the fact that the low temperature overpressure protection function is only used during shutdown conditions which are not significantly affected by the proposed changes in operating conditions. The staff finds this conclusion acceptable.

Auxiliary Feedwater and Residual Heat Removal

The staff review and approval of the Wolf Creek auxiliary feedwater system (AFS) design is given in NUREG-0881, Safety Evaluation Report (SER), Section 10.4.9 by reference to NUREG-0830, SER for Callaway. These SERs describe the limiting transients identifying worst single failure assumptions and minimum flow requirements for the Wolf Creek AFS design. The rerate conditions were found not to change the limiting scenarios for single failure and required flow. The adequacy of revised AFS flow capacity was demonstrated in the supporting analyses submitted for Amendment 61. The licensee determined that the condensate storage tank volume required by technical specifications is adequate to support AFS operation at the rerate power level for the design basis plant cooldown to RHR entry conditions and for the station blackout coping analysis. The staff concludes that the performance of the AFS remains adequate to provide the required cooling functions following the change to the rerate conditions.

The licensee indicated that the original 16-hour plant cooldown time from residual heat removal (RHR) initiation at 350°F to 140°F would be increased to 17 hours for the rerated conditions. Although the time to cool down to 200°F with only a single RHR train available increases from 12 hours to 26 hours, the system capabilities continue to comply with the guidance provided in Branch Technical Position RSB 5-1 and 10 CFR Part 50, Appendix R. Therefore, the staff concludes that the impact of the rerate conditions on RHR performance is acceptable.

Emergency Core Cooling System (ECCS)

From the licensee's study, no adverse impact to ECCS operability or vulnerability to single failure due to the rerated conditions was identified. The licensee submitted revised ECCS performance analyses in support of Amendment 61 which justified various changes associated with Cycle 7 operation. The licensee performed large and small break analyses at the limiting rerate conditions and determined that all acceptance criteria continued to be satisfied. The NRC staff has reviewed the licensee's analyses and concludes that the ECCS analyses referenced in support of the rerate conditions continues to be in compliance with 10 CFR 50.46 and Appendix K. The Wolf Creek ECCS is, therefore, acceptable for operation at the rerated conditions.

Accident Analyses

The licensee states that all events in USAR Chapter 15 were reanalyzed or reevaluated considering the rerate conditions. The details of these analyses were submitted in support of Amendment 61 which justified various parameter changes associated with Cycle 7 operation. Included in the revised accident analysis assumptions is a steam generator plugging level of 10 percent. The staff reviewed these analyses and concluded that the appropriate safety criteria continue to be met. The methodologies used by the licensee to analyze the USAR Chapter 15 accidents and transients have been previously reviewed and approved by the staff.

The licensee provided the evaluation of the effect of the rerate on accident radiological consequences with the supporting documentation for Amendment 61. The original licensing basis accident analysis source terms for Wolf Creek were conservatively based on an assumed core power level equal to the proposed rerated core power level of 3565 MWt. The original licensing basis source terms assumed a twelve month operating cycle. As part of the analytical efforts related to the rerate and cycle 7 operation (Amendment 61), the radiological source terms were determined with an assumed operating cycle of eighteen months to reflect current fuel cycle designs. Except for the changes related to cycle length, the accident doses were re-calculated using methodology and assumptions which are consistent with the original licensing basis. The calculated doses for the exclusion area, low population zone, and control room were found to change slightly from those presented in the original licensing basis but in all cases remained well below applicable regulatory limits.

2.2 Safety Related Cooling Water Systems

Service Water and Ultimate Heat Sink

The Wolf Creek station service water system consists of the service water system (SWS) and the essential service water system (ESWS). The SWS is used during normal operation and normal shutdown conditions. The ESWS removes heat from plant components which require cooling for safe shutdown of the reactor or following a design basis accident. Major components supplied by the ESWS include the component cooling water (CCW) heat exchangers, containment air coolers, diesel generator cooler, and various engineered safety features room coolers. The ESWS can also provide makeup water to the spent fuel pool and CCW system as well as serve as the backup water supply to the auxiliary feedwater system. The ESWS intake and discharge is the plant's main cooling lake which also serves as the ultimate heat sink (UHS). The UHS for Wolf Creek consists of a normally submerged seismic category I cooling pond which is formed by a dam built into the main cooling lake.

The licensee evaluated the ESWS and UHS for the rerated conditions under the most limited postulated post-accident scenarios. New analyses of ESWS heat inputs, including the rerate related changes, determined that no significant increases occurred which would result in the plant being outside original bounding conditions or analysis results. Analysis of UHS capabilities were originally performed assuming the rerated power level. Other changes associated with the rerate were determined to have an insignificant impact on UHS temperatures. Thus the analysis demonstrates that the UHS and ESWS have adequate cooling capability to satisfy design and regulatory requirements and are therefore acceptable for operation at the rerated conditions.

Component Cooling Water

The component cooling water (CCW) system provides cooling water to selected auxiliary components during normal operation and provides cooling water to several engineered safety feature systems (ESFS) during design basis accidents. The system is a closed loop system which serves as an intermediate barrier between the SWS or ESWS and potentially radioactive systems in order to eliminate the possibility of an uncontrolled release of radioactivity. Nonessential loads are isolated in the event of a safety injection signal. Essential loads of the CCW system include the residual heat removal (RHR) heat exchangers, ECCS pump coolers, spent fuel pool heat exchangers, and reactor coolant pump coolers.

The licensee evaluated the effects of the proposed rerated conditions on the CCW system. The licensee found that normal and accident heat loads following the proposed rerate would be bounded by existing analyses. Based upon the licensee's evaluations of the CCW and related systems, the staff finds that the system will continue to fulfill its functions under the rerated plant conditions.

Spent Fuel Pool Cooling

The licensee evaluated the spent fuel pool cooling (SFPC) system to determine the effects of the increased rated thermal power on the capability of the SFPC system to maintain fuel pool temperatures within acceptable limits. The evaluation included scenarios related to a normal refueling offload and a full core offload. In each scenario, only a single train of SFPC was assumed to be operating. The total heat load was determined in accordance with the current licensing basis analysis.

The licensee's analyses concluded that the bulk spent fuel pool temperature would remain below 135°F for the normal offload and below 160°F for the full core offload case. The staff concludes that the SFPC system has adequate capacity to support the additional heat loads associated with the rerated conditions and continues to satisfy applicable design and regulatory requirements.

2.3 Balance of Plant

Turbine Generator

The turbine overspeed protection system reduces the risk of generation of turbine missiles that could impact operation of safety-related structures, systems, or components. The licensee's evaluation concluded that the maximum steam generator outlet steam flow for the various possible operating conditions would not be significantly greater than the valves wide open (VWO) steam flows assumed in the original turbine overspeed analyses. The bases for the turbine overspeed protection system are not considered to be adversely impacted by the proposed rerate conditions. Since the probability of turbine missile generation is not significantly changed by operation at the rerated conditions, the staff concludes that the existing turbine overspeed protection is adequate for the rerated conditions.

It should be noted that although bounded by the above discussion regarding steam flows and turbine overspeed protection, the licensee does not anticipate that the maximum proposed temperature reduction (15°F) can be achieved with the desired electrical output without modification of the turbine generator.

Main Steam

The main steam system dissipates energy generated by the reactor core to the turbine generator and auxiliary steam loads, the main condenser via the steam dump valves, or to the atmosphere via atmospheric relief valves or main steam safety valves. Isolation of the main steam system is achieved by the main steam isolation valves and main steam bypass isolation valves.

The licensee evaluated the capability of the main steam system components to perform their design functions under the proposed rerate conditions. The licensee determined that the existing setpoints and capacity of the main steam safety valves are adequate to prevent exceeding 110 percent of design pressure of the main steam system under the most limiting transient. The setpoint and

capacity of the atmospheric relief valves were found to remain adequate to control the design load shed of 10 percent rated thermal power. In addition, the atmospheric relief valves were found to have adequate capacity to achieve a 50°F per hour cooldown if the main condenser was unavailable. The main steam isolation valves were evaluated to ensure the valves will continue to perform their isolation function under the maximum differential pressure conditions and within the time limits assumed in the safety analysis.

The staff concludes that the existing main steam system components are adequate to perform their safety functions under the rerated plant conditions.

Main Feedwater

The main feedwater system delivers feedwater, at the required pressure and temperature, to the four steam generators. The safety-related portions of the system ensure isolation capability and provide a path to permit the addition of auxiliary feedwater for reactor cooldown following design basis transients.

The licensee's evaluation shows that the existing design basis for the main feedwater isolation valves and main feedwater bypass isolation valves is not significantly affected by operation at the rerate conditions. The piping configurations associated with the feedwater and auxiliary feedwater systems do not change as a result of the rerate conditions. The ability of the auxiliary feedwater system to perform its heat removal function was addressed by the licensee. The staff finds that the safety functions of the feedwater system will continue to be satisfied during operation at the rerate conditions.

Other Power Conversion Systems

The licensee's analyses indicate that power conversion systems and components (e.g., the steam dump valves, extraction steam system, turbine generator, main condenser, condensate system, feedwater heaters, heater drain system, and circulating water system) satisfy their power generation design bases for the operation as proposed except that the turbine may require modification to allow full power operation at the minimum proposed reactor coolant temperature. Since the power generation design bases do not include safety functions, these systems and components do not require review by the staff.

2.4 Containment Analyses

The licensee has performed containment integrity analyses at the rerated conditions to ensure that the maximum pressure inside containment following a design basis accident remains below the containment building design pressure of 60 psig. The calculated peak pressure is also used as a basis for the containment leak rate test pressure to ensure that dose limits would be met in the event of a release of radioactivity to containment.

The licensee indicated that the containment functional analyses included the assumption of most limiting single active failure and the modelling of minimum

and maximum safety injection flows. Bounding initial temperatures and pressures for analyses were selected to envelope the limiting conditions of operation. The licensee indicated that although the current licensed NSSS power level is 3425 MWt, containment pressure analyses were initially performed assuming an NSSS power of 3579 MWt. However, improved analytical models, developed since the original licensing analyses, have been utilized in the evaluations performed for the proposed rerate.

Loss of Coolant Accidents (LOCA)

As in the current licensing basis, the licensee's rerate analyses determined the containment temperature and pressure response for a postulated double ended reactor coolant pump suction break (DEPSG) for both minimum and maximum safety injection cases and a postulated double ended hot leg break (DEHLG) which results in the maximum mass and energy release rates during the blowdown phase of the accident.

For the DEHLG case, the Wolf Creek rerate analyses resulted in a containment peak pressure of 42.6 psig. The existing licensing basis analyses for Wolf Creek resulted in a peak pressure of 41.7 psig. For the DEPSG cases, the rerate analyses resulted in peak containment pressures of 43.9 psig and 41.0 psig for minimum and maximum safety injection respectively. The corresponding analysis results for existing licensing basis analyses are 47.3 psig and 46.0 psig. The licensee indicated that the reduced calculated peak pressures for the rerate conditions compared to the existing analyses are due to model enhancements. The differences between the existing licensing basis methodology and the rerate analyses include mass and energy releases calculated using the Westinghouse 1979 evaluation model instead of the 1975 model and modelling of steam/water mixing in the broken loop.

The current technical specification containment leak rate pressure (Pa) is 48 psig. Although the rerate analysis results determined a reduced peak containment pressure, the licensee has not proposed to reduce the value of Pa as part of the rerate request. In addition to the review of the peak containment pressures, the licensee demonstrated that pressure decreased to less than 50 percent of the peak value within the 24 hours following the postulated LOCA. The LOCA temperature response was found to remain bounded by the main steamline break analysis results. The staff has reviewed the licensee's evaluations and determined that the licensee has adequately demonstrated that the containment will satisfy its design functions under the rerate conditions.

Main Steam Line Breaks

The licensee's rerate evaluation included determination of containment temperature and pressure response for four cases selected from the existing analysis spectrum of main steamline break sizes and initial power levels. The cases were chosen to demonstrate that the existing analyses bound the rerate conditions. The evaluation was determined to adequately demonstrate that the existing analyses conservatively bound the possible rerate operating conditions.

Subcompartment Analysis

The licensee indicated that the existing mass and energy releases remain bounding for the reactor cavity and steam generator subcompartments if leak-before-break is credited. The limiting case for the pressurizer subcompartments was and remains an assumed break of the surge line. The licensee determined that the rerate conditions would result in an increased mass and energy release to the pressurizer subcompartments. However, significant margins remain between the design limits for the pressurizer subcompartments and the calculated maximum pressures associated with the rerate analyses. Based on the licensee's evaluation, the staff concludes that the rerating is acceptable in terms of subcompartment pressure loading analyses.

Post-LOCA Hydrogen

The licensee indicated that the Wolf Creek containment post-LOCA hydrogen generation analyses were reviewed to determine any impact due to the rerate. The LOCA analysis determined that the hydrogen generation from zirconium/water reactions remains less than the hypothetical 1 percent assumed. The current hydrogen generation from radiolysis is based on a core power of 3636 MWt, which is 102 percent of the rerated power level. For corrosion sources, the hydrogen generation was determined based on corrosion rates corresponding to the temperature profiles calculated for the containment under post-LOCA conditions. Considering the various sources, the licensee's analysis determined that the current hydrogen generation analysis remains bounding for the rerate conditions. Therefore, the staff concludes that the hydrogen control systems and related hydrogen generation analysis are not affected by the rerate conditions.

2.5 Plant Structural Analyses

The licensee has performed evaluations of the effects of the proposed rerate conditions upon the structural integrity of the NSSS and BOP pressure boundary systems, including the system piping, components, related supports, the reactor vessel and internals, steam generators, control rod drive mechanisms, reactor coolant pumps, and pressurizer.

Reactor Vessel and Internals

The licensee assessed the adequacy of the reactor vessel by determining the stress and fatigue usage effects resulting from operation at the rerated conditions throughout the period of the current operating license. The calculated stresses and fatigue usage factors for the reactor vessel components were found to be within the allowable design limits.

The licensee assessed the adequacy of the reactor internal components for the rerate conditions. The assessment included analyses for a LOCA, flow induced vibration, seismic, thermal transients, and stress and fatigue analysis for reactor internal components. The licensee stated that the structural adequacy of the reactor internals is not affected by operation at the rerate

conditions. The staff has reviewed the licensee's assessment and concludes that the integrity of the reactor vessel and internal components is not affected by the rerate conditions.

The licensee has also addressed reactor vessel structural integrity with respect to fracture toughness requirements for protection against pressurized thermal shock. Recent licensee submittals have provided the analytical results of the most recently withdrawn surveillance capsule; response to Generic Letter 92-01, "Reactor Vessel Structural Integrity," and proposed changes to the reactor coolant pressure and temperature limits. The licensee assumed the rerated conditions for future operation in the determination of the revised pressure and temperature limits. The staff has determined that no immediate concerns related to reactor vessel embrittlement and compliance with regulations are introduced by the proposed rerate at Wolf Creek and will address the issue more thoroughly during reviews of the above submittals.

Control Rod Drive Mechanisms

The licensee evaluated the adequacy of the Control Rod Drive Mechanisms (CRDMs) by comparing the design bases input parameters with the operating conditions for the proposed rerate. The licensee stated that the rerate conditions would have an insignificant impact on the original design bases analyses for the CRDMs. The staff has reviewed the licensee's evaluation and concurs with the licensee's conclusion that the current design of the CRDMs would not be impacted by the rerate.

Steam Generators

The licensee performed analyses of Wolf Creek's steam generators for operation at the proposed rerate conditions. The evaluations were performed according to the requirements of the ASME Code, Section III, 1971 to 1973 Editions. The calculated stress intensities were found to be within the allowable limits for all locations. The fatigue usage factors were also found to be acceptable. However, the increased cumulative usage in the secondary manway bolts shortened their fatigue life from 20 years to 18 years for the planned rerate conditions and to 15.8 years for the maximum temperature reduction rerate conditions.

The Westinghouse Model F steam generators at Wolf Creek have thermally treated Inconel-600 tubes with an outside diameter of 0.688 inches and a wall thickness of 0.040 inches. As a result of the proposed changes in operating conditions, several design parameters of interest to steam generator tube structural performance would change. These include reactor coolant temperatures, the differential pressure across the steam generator tubes, and the flow rates through the steam generators. The licensee has analyzed these design parameter changes associated with the rerate for the effect on the structural integrity of the steam generator tubing. The evaluations performed included cases bounding the potential range of rerate conditions. The licensee performed evaluations of the effect of the rerate on the minimum required steam generator tube wall thickness, the number of steam generator tubes susceptible to anti-vibration bar (AVB) wear, and the propensity of the steam generator tubing to various forms of corrosion degradation.

The licensee determined the minimum acceptable steam generator tube wall thickness for the rerated conditions to be 0.016 inches (40 percent of the nominal wall thickness) using the criteria of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." The present minimum acceptable tube wall thickness for Wolf Creek is 0.014 inches (35 percent of the nominal wall thickness). This increase in minimum steam generator tube wall thickness is due to the increased differential pressure across the tubes for the rerate conditions. The technical specification plugging limit is an imperfection depth of 40 percent (minimum undegraded wall thickness of 60 percent) and does not need to be amended for the rerate conditions.

The licensee also performed an analysis of the number of additional steam generator tubes that may be affected by wear at the AVB supports as a result of operation at the rerate conditions. This analysis determined that the number of tubes which may require plugging due to AVB wear could increase slightly as a result of operation at the rerate conditions. The estimated increase in AVB wear for the rerated conditions was determined using a three step approach. First, the change in stability of the tubes with respect to fluid elastic vibration was determined. A probabalistic approach was then used to estimate the proportion of tubes which would become unstable as a result of possible fluid conditions at the AVB supports. Finally, the increase in the number of tubes subject to wear was estimated using field data for the distribution of wear as a function of time.

Since the analysis determined that there was a slight increase in the potential for tube wear, the licensee plans to perform increased inspections. The licensee committed to follow Westinghouse recommendations for inspection of the steam generators following the implementation of the plant rerate. The surveillances will consist of inspecting all steam generator tubes in Rows 25 or greater during the three refueling outages following the rerate. A significant fraction of these inspections will be completed during the first or second refueling outage following the rerate. The purpose of these inspections is to assess wear rates under the rerate conditions. The staff notes that the growth rates are used in the calculation of the technical specification tube plugging limit. The licensee has also committed to notify the NRC of any significant increase in the historically observed steam generator tube wear rates at Wolf Creek.

The licensee performed an evaluation of the effect of the rerating on the corrosion propensity for the steam generators. The analysis technique did not calculate absolute corrosion rates associated with the various corrosion mechanisms but did calculate the relative rate changes resulting from operation at the rerate conditions. The Wolf Creek steam generators were determined to be most susceptible to pitting and ODSCC and least susceptible to denting corrosion. For the proposed rerate conditions, the impacts on the various corrosion mechanisms were determined to be that pitting and ODSCC propensity remained basically unchanged, PWSCC is reduced, and the propensity for denting is increased. It should be noted that factors such as chemistry which are not directly affected by the rerate continue to significantly influence the corrosion rates of steam generator tubes.

Based upon the information presented by the licensee, the staff finds that the Wolf Creek steam generators are acceptable for operation under the conditions related to the rerate.

Reactor Coolant Pumps

The licensee has evaluated the impact of the conditions resulting from the proposed rerate on the design analyses of the reactor coolant pumps. The licensee determined that the proposed changes in operating temperatures have an insignificant effect on the thermal analysis performed on the reactor coolant pump design. Based on the review of the rerated transients, the licensee confirmed that the original design analyses remain bounding. The reactor coolant pump motors were evaluated and the increased loads associated with the reduced coolant temperatures were considered to be acceptable. The NRC staff has reviewed the licensee's assessment and finds that the reactor coolant pumps are acceptable for operation at the rerated conditions.

Pressurizer

The licensee has evaluated the pressurizer equipment specification and stress report relative to the proposed rerate conditions and revised NSSS transients.

The evaluation determined that the generic transients used in the original analyses remain bounding. The evaluation included consideration of changes in hot leg and pressurizer spray temperatures. Although these parameters change as a result of the rerate, the temperature differentials and related thermal stresses are enveloped by the original design analyses. The licensee's evaluations determined that the pressurizer control features such as safety valves, relief valves, spray valves, and heaters remained adequate for operation at the rerated conditions. Based on its review, the staff concludes that the pressurizer remains acceptable for operation at the rerated conditions.

NSSS Piping, Components, and Supports

The licensee evaluated the following components and supports for the rerated operating conditions: the RCS piping and supports, the primary equipment nozzles, the primary equipment supports, and the Class 1 auxiliary (branch) lines connected to the RCS piping. The evaluation compared the existing design bases with the performance requirements at the rerated conditions with respect to the design system parameters, transients, LOCA forcing functions, and the dynamic LOCA reactor vessel movements used in the original structural analyses. The licensee's adoption of leak-before-break design criteria ensures that the original design analyses, based on large guillotine breaks, remain bounding for the proposed rerate conditions. The gap conditions of the primary equipment support system have been assumed to change by a negligible amount and therefore the seismic loadings on the equipment would not be significantly affected by the lower operating temperatures. The licensee also factored the analyses and evaluations related to the "noise events" into the rerate program. The licensee's evaluations also addressed piping and equipment fatigue and surge line thermal stratification issues.

The licensee also evaluated the auxiliary NSSS components. The evaluation was performed by comparing the original design bases and qualification requirements with those for the rerated conditions. The licensee found that, for each component evaluated, the original design enveloped those for the rerate.

Based on its review, the staff agrees with the licensee's conclusion that the existing NSSS piping and supports, primary equipment nozzles, primary equipment supports, branch lines, and auxiliary NSSS components remain in compliance with the design bases criteria given in the USAR with respect to the rerate operating conditions. These components and supports are, therefore, acceptable for operation as proposed.

BOP Piping, Components, and Supports

The licensee evaluated the adequacy of the BOP piping systems by comparing the existing design basis conditions with those for the proposed rerated conditions. From this review, the licensee determined that most of the original design analyses bounded the conditions associated with the rerate. For those cases in which an increase in forces was calculated, the licensee confirmed that piping stresses and support loads remained within acceptable limits.

The licensee also reviewed the design bases pipe break analyses to evaluate the effects of the rerate conditions upon pipe break locations, jet thrust and impingement forces, and the design of pipe whip restraints. In all cases, the licensee determined that original analyses bases remained conservative with respect to the rerate operating conditions.

Based on the review of the information provided by the licensee, the staff finds that the changes associated with the rerate conditions would either remain bounded by the original design analyses or have been evaluated and the related effects have been found to be acceptable.

2.6 Miscellaneous Systems and Programs

Environmental Qualification

The licensee evaluated the effects of the proposed changes on qualified equipment. The radiological dose related to equipment qualification were evaluated by the licensee and determined to be acceptable. Conservatisms in the original equipment specifications allowed for a source term assumption of 50 percent of the total available Cesium immediately following a loss-of-coolant accident. For those cases in which the increased doses associated with the rerate were above the original analyses, the licensee confirmed that a revision to the design basis to a 1 percent Cesium source term would ensure equipment qualification. The calculated environmental temperature and pressure profiles for the limiting LOCA and MSLB cases based on the proposed rerate conditions are enveloped by the current Wolf Creek equipment qualification analyses for equipment located inside containment. The licensee's evaluation of mass and energy releases in the MSIV area, including blowdown of superheated steam under conditions associated with the rerate,

determined that the maximum calculated temperatures are bounded by the existing design basis analyses.

Since the equipment qualification parameters affected by the proposed changes remain bounded by the values determined by the licensee's current analyses, the staff concludes that the effect of the proposed rerate on equipment qualification is minimal and therefore operation of the plant at the rerated conditions is acceptable.

Electrical Distribution System

The staff has reviewed the information provided by the licensee regarding the main generator, transformers, onsite and offsite electrical distribution systems, and miscellaneous power systems. The electrical system changes described included only modifications to high voltage transformers to handle the increased electrical output. The staff would not expect any significant system load changes as a result of the proposed rerate conditions.

Accordingly, the staff has concluded that the licensee's discussions in these areas are acceptable.

Radioactive Waste Systems

The Wolf Creek radwaste systems were originally evaluated and accepted by the NRC staff based upon an assumed core power level of 3565 MWt. This power level was used to determine the source term for gaseous and liquid effluents and the waste volume. Therefore, operation at the rerate conditions would not change the analyses results or the staff's conclusions in the Wolf Creek SER (NUREG-0881). Accordingly, the radwaste systems continue to be acceptable for control of radioactive wastes for operation as proposed.

Heating, Ventilation, and Air Conditioning (HVAC) Systems

The control room emergency ventilation system is equipped with isolation, pressurization and filtering components to limit the dose to control room personnel following a design basis accident and to maintain a habitable environment in the control room. The original design basis analysis was based upon a source term for a core power level of 3565 MWt which is the same as the proposed rerate conditions. Therefore, the staff's conclusion in the Wolf Creek SER (NUREG-0881) remain valid for the proposed changes.

The licensee has also addressed other requirements on HVAC systems including the required heat removal for the support of engineered safety features equipment. The proposed rerate conditions were found to have negligible impact on the requirements for or performance of HVAC systems. The staff finds the licensee's conclusions acceptable.

Internal Flooding

The licensee evaluated the potential impact of the rerate conditions on the threat of internal flooding. The evaluation of piping systems considered the changes in operating conditions including pressures and flow rates. Based

upon a review of the pipe break outflows for primary and secondary systems, the evaluation determined that the results from the existing flooding analyses bound the analyses for the rerate conditions.

Therefore, the staff concludes that the present flooding analyses continue to be valid and that flood levels would not increase as a result of operation at the rerate conditions.

2.7 Proposed License and Technical Specification Changes

In order to allow the operation of Wolf Creek Generating Station at the proposed rerate conditions, the licensee proposed several changes to the Facility Operating License and associated Technical Specifications. The proposed changes consist of:

1. A change in the Facility Operating License maximum power level from "3411 megawatts thermal (100% power)" to "3565 megawatts thermal (100% power)".

2. Changes to the uncertainty allowances and nominal T_{avg} associated with the overpower and overtemperature delta-temperature trip function setpoints in Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints.

3. Changes the maximum indicated reactor coolant system T_{avg} from 592.5°F to 585.0°F in Table 3.2-1, DNB Parameters.

The staff's review of the proposed changes to the Operating License and Technical Specifications determined that the changes are consistent with the design analyses discussed previously. The staff finds the proposed changes to be acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on October 25, 1993 (58 FR 55086)

In this finding, the Commission determined that issuance of this amendment would not have a significant effect on the quality of the human environment.

5.0 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: W. Reckley

Date: November 10, 1993

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

WOLF CREEK NUCLEAR OPERATING CORPORATION

DOCKET NO. 50-482

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 69 to Facility Operating License No. NPF-42, issued to Wolf Creek Nuclear Operating Corporation (the licensee), which revised the License and Technical Specifications for operation of the Wolf Creek Generating Station located in Coffey County, Kansas.

The amendment revises the License and the Technical Specifications to allow an increase in the maximum reactor core power level (rated thermal power) from the present value of 3411 megawatts thermal (MWt) to a revised limit of 3565 MWt. The increase in allowed core power combined with the energy added by the reactor coolant pumps would result in the proposed changes allowing Wolf Creek to operate at a nuclear steam supply system power of 3579 MWt. In addition to the increased core power level, changes were made in the allowable operating temperatures of the reactor coolant system. The reductions in reactor coolant hot-leg temperatures were made in order to reduce the potential for stress corrosion cracking of steam generator tubes.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate

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findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on June 9, 1993 (58 FR 32392). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment. The environmental assessment and finding of no significant impact was published in the FEDERAL REGISTER on October 25, 1993 (58 FR 55086).

For further details with respect to the action see (1) the application for amendment dated January 5, 1993, as supplemented by letter dated October 1, 1993, (2) Amendment No. 69 to License No. NPF-42, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, D.C. 20555, and at the Local Public Document Rooms, Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621. A copy of items (2), (3), and (4) may be obtained upon request

addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects, III/IV/V.

Dated at Rockville, Maryland, this 10th day of November 1993.

FOR THE NUCLEAR REGULATORY COMMISSION

William D. Reckley

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