March 3, 1994

Docket No. 50-482

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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. 72 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. M88734)

The Commission has issued the enclosed Amendment No. 72 to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications in response to your application dated February 7, 1994.

The amendment allows an increase in reactor coolant temperature in order to support operation at the rated thermal power of 3565 megawatts thermal (MWt). The amendment changes reactor protection system overtemperature/overpower delta-temperature setpoints by increasing the nominal reactor coolant average temperature from 581.2°F to 586.5°F, changing the axial flux difference penalties, and changing the setpoint uncertainty allowances. The amendment also increases Technical Specification 3/4.2.5, DNB Parameters, maximum indicated reactor coolant system average temperature from 585.0°F to 590.5°F.

This license amendment has been handled on an exigent basis in accordance with 10 CFR 50.91(a)(6). 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/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 72 to NPF-42
2. Safety Evaluation

cc w/enclosures:
See next page

*For previous concurrences, see attached ORC

CP-1

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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.72
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 February 7, 1994, 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 Specifi cations 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. 72 , 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 60 days of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

William D Beckley for

Suzanne C. Black, 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: March 3, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 72

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
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

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	≤10^5 cps	≤1.6×10^5 cps
7. Overtemperature ΔT	7.0	4.86	1.67	See Note 1	See Note 2
8. Overpower ΔT	4.6	2.02	0.14	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

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

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_o	=	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'	≤	586.5°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 + 5%, $f_1(\Delta I) = 0$, 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 +5%, the ΔT Trip Setpoint shall be automatically reduced by 1.56% 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 1.8% of ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

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_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}, $\tau_7 = 10$ 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, $\tau_6 = 0$ 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 ≤ T";
T	=	Average temperature, °F;
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 586.5°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 2.6% 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}	$\leq$590.5°F
2. Indicated Pressurizer Pressure	$\geq$2220 psig*
3. Reactor Coolant System Flow Rate	$\geq 38.4 \times 10^4$ 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 590.5°F and the indicated pressurizer pressure value of 2220 psig correspond to analytical limits of 593.0°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.

BASES

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. 72 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 February 7, 1994, 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 allow an increase in reactor coolant temperature in order to support operation at the rated thermal power of 3565 megawatts thermal (MWt). The proposed amendment changes reactor protection system overtemperature and overpower delta-temperature setpoints by increasing the nominal reactor coolant temperature from 581.2°F to 586.5°F, changing the axial flux difference penalties, and changing the setpoint uncertainty allowances. The proposed amendment also increases the maximum indicated reactor coolant system average temperature of Technical Specification 3/4.2.5, DNB Parameters, from 585.0°F to 590.5°F.

2.0 BACKGROUND

The NRC issued Amendment No. 69 to the Wolf Creek Generating Station Facility Operating License on November 10, 1993. The amendment increased the rated thermal power for Wolf Creek from 3411 megawatts thermal (MWt) to 3565 MWt. The amendment also included changes in reactor coolant temperature specifications to reflect the planned operation of Wolf Creek at the higher power level and reduced operating temperatures. The desire to operate at reduced reactor coolant temperatures is related to minimizing the propensity for some forms of steam generator tube corrosion mechanisms. Upon attempting to implement the power increase, the licensee discovered that the unit was unable to achieve 3565 MWt at the reduced operating temperatures and associated steam pressures. The proposed amendment would allow operation at increased operating temperatures in order to allow the plant to reach its licensed power level. The licensee plans to implement modifications during the next refueling outage which will allow operation at the licensed power level and reduced operating temperatures.

3.0 EXIGENT CIRCUMSTANCES

The Commission's regulations, 10 CFR 50.91, contain provisions for issuance of amendments with less than a 30 day comment period if either emergency or exigent circumstances are determined to exist.

Emergency situations involve those cases in which failure to act in a timely way results in the derating or shutdown of a nuclear power plant, or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level. Under emergency circumstances, the Commission may issue a license amendment involving no significant hazards consideration without prior notice and opportunity for a hearing or for public comment. In such a situation, the Commission publishes a notice of issuance under 10 CFR 2.106, providing for opportunity for a hearing and for public comment after issuance.

The processing of an amendment under exigent circumstances usually applies to those cases in which the licensee and Commission must act promptly, but failure to act promptly does not involve a plant shutdown, derating, or delay in startup. For both emergency and exigent circumstances, the licensee is required to explain the reason for the condition and why it could not be avoided. This requirement is intended to prevent the abuse of the special provisions of 10 CFR 50.91(a)(6). Under exigent circumstances, the Commission notifies the public in one of two ways: by issuing a Federal Register notice providing notice of an opportunity for hearing and allowing at least two weeks from the date of the notice for prior public comment; or by using local media to provide reasonable notice to the public in the area surrounding a licensee's facility and providing special instructions for providing comment. For this amendment request, the Commission employed the first approach with a Federal Register notice published on February 15, 1994 (59 FR 7269) which presented the staff's proposed no significant hazards consideration determination and requested public comment within 15 days after the date of publication of the notice.

The Commission issued Amendment 69 to the Wolf Creek Generating Station Facility Operating License on November 10, 1993. The amendment increased the rated thermal power for Wolf Creek from 3411 MWt to 3565 MWt. The amendment also included changes in reactor coolant temperature specifications in order to reduce the propensity for some forms of steam generator tube corrosion. The licensee's implementation of the power rerate and temperature reductions were performed during the period from November 17, 1993 to December 21, 1993. During the implementation, the licensee discovered that the unit was unable to achieve 3565 MWt at the reduced operating temperatures. The reduced operating temperature specifications had therefore resulted in an effective derating of the unit.

Following the completion of various safety and nuclear design analyses, the licensee submitted revisions to the temperature specifications on February 7, 1994, in order to allow the unit to reach its licensed power level. The licensee has determined that this is the only feasible method to increase power output until design changes can be implemented during the next refueling outage. The staff has determined that the licensee cannot avoid the current condition limiting the power output of Wolf Creek Generating Station and has filed a timely application to allow operation at increased operating temperatures until design modifications can be implemented during the next refueling outage. Therefore, the special provisions of 10 CFR 50.91(a)(6) are applicable to this proposed amendment.

The amendment may have satisfied the criterion for issuance under emergency circumstances because the licensee was unable to increase power output to the plant's licensed power level. However, the plant has been able to continue power production at a level above the initial licensed power of 3411 MWt. In an effort to balance the desire to provide an opportunity for prior public comment whenever possible and the economic impact of the derating of the Wolf Creek Generating Station, the staff is issuing this amendment on an exigent basis following a 15-day comment period as permitted by 10 CFR 50.91(a)(6).

4.0 EVALUATION

Amendment No. 69 to the Wolf Creek Generating Station Facility Operating License involved an increase in the unit's maximum licensed power level from 3411 MWt to 3565 MWt. The changes also reflected a planned hot leg temperature reduction of 5 degrees Fahrenheit (5°F) and a possible 15°F reduction which may be pursued in the future. In support of the amendment, 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 nuclear steam supply system (NSSS) and balance of plant (BOP). Many of the supporting analyses for the rerate associated with Amendment 69 were performed with an assumed hot leg temperature of 620°F which represented an increase of approximately 1.8°F compared to the operating conditions prior to the rerate. As stated in the staff's safety evaluation related to Amendment 69, the 5°F hot leg temperature reduction which was associated with the rerate was proposed in order to meet safety limit design criteria (Departure from Nucleate Boiling (DNB)). A comparison of the operating conditions associated with the rerate and proposed amendment are provided below:

Parameter	Prior to Amendment 69	Amendment 69 Upper Bound	Amendment 69 Lower Bound	Proposed Amendment
Core Power	3411 MWt	3565 MWt	3565 MWt	3565 MWt
Thermal Design Flow	374,400 gpm	374,400 gpm	374,400 gpm	374,400 gpm
Vessel Outlet Temperature	618.2°F	620.0°F*	603.2°F	618.2°F
Vessel Average Temperature	588.5°F	588.4°F*	570.7°F	586.5°F

* - Upper Bound value was used for most analyses. For selected analyses, including the loss of flow transient and core design, values of 613.2°F and 581.2°F respectively, were used for the assumed vessel outlet and average temperature

The licensee's evaluation determined that the only Updated Safety Analysis Report (USAR) Chapter 15 transient which required re-analysis to support the proposed increase in reactor coolant temperature was the complete loss of forced reactor coolant flow DNB evaluation. The remaining transient analyses had been performed assuming the limiting vessel average temperature of either 588.4°F or 570.7°F. The methodology and assumptions, other than reactor coolant temperatures, used in the analysis of the loss of flow transient were the same as those submitted in support of Amendments Nos. 61 and 69. Amendment 61 supported operation of Wolf Creek following the sixth refueling outage and represented a transfer of many of the safety analysis and nuclear design functions from the fuel vendor to Wolf Creek Nuclear Operating Corporation. As listed in Technical Specification 6.9.1.9, Core Operating Limits Report, the methodologies utilized by the licensee have been reviewed and approved by the staff. The reanalysis of the complete loss of forced reactor coolant flow transient at the higher reactor coolant temperature, 586.5°F average temperature, demonstrated that the departure from nucleate boiling ratio (DNBR) remained above the safety analysis limit.

The licensee examined the nuclear design operational and transient limits necessary for the remainder of Cycle 7 operation at a core power of 3565 MWt and proposed increase in reactor coolant average temperature to 586.5°F. The core power distribution limits were determined as described in the NRC approved topical report NSAG-007, Rev 0, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station." The maneuvering analyses determined that more restrictive axial flux difference limits were required to support operation at the increased reactor coolant temperatures. The more restrictive axial flux difference penalty associated with the overtemperature delta-temperature protection function is part of the proposed amendment. The licensee evaluated other nuclear design parameters for operation for the remainder of Cycle 7 at increased reactor coolant temperatures and determined all were bounded by the values assumed in the safety analyses.

Amendment 69 supporting analyses related to piping and component integrity were reviewed and determined to remain bounding of the proposed operating temperatures. As stated above, these analyses were performed at the more limiting hot leg temperature of 620°F or 603.2°F. This bounds the proposed operating condition for hot leg temperatures of 618.2°F and average coolant temperature of 586.5°F.

The licensee also reperformed the uncertainty analyses which determined the total allowance (TA), sensor error (S), and "Z" terms in Table 2.2.1, Reactor Protection System Instrumentation Trip Setpoints. Discussions with the licensee determined that the changes were a result of minor changes to the calculations and that the overall uncertainty methodology remained similar to that used since the initial licensing of the facility.

The staff has reviewed the licensee's proposed Technical Specification changes and supporting evaluations and finds the changes acceptable.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations if operation of the facility in accordance with the amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. The probability of occurrence and the consequences of an accident evaluated previously in the Updated Safety Analysis Report (USAR) are not increased due to the proposed technical specification change. Plant operation at 3565 MWt with the revised temperatures does not affect any of the mechanisms postulated in the USAR to cause loss-of-coolant accident (LOCA) or non-LOCA design basis events. Analyses, evaluations, and minimum departure from nucleate boiling ratio (DNBR) calculations confirm that the USAR conclusions remain valid for the proposed changes. On these bases it is concluded that the probability and consequences of the accidents previously evaluated in the USAR are not increased.

Operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed technical specification changes do not increase the probability of occurrence of a malfunction of equipment important to safety or increase the consequences of a malfunction of equipment evaluated in the USAR. The technical specification changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because the change in operating T_{hot} will not pose a new operating configuration that would create a new failure scenario. The proposed changes do not change the plant configuration in a way that introduces a new potential hazard to the plant and do not involve a significant reduction in the margin of safety. No new failure modes will be created by the proposed changes for any plant equipment. Operation with a 0° - 5°F T_{hot} reduction is bounded by the analyses performed previously for the power rerate and approved by the NRC in Amendment No. 69 to the Wolf Creek Generating Station (WCGS) Technical Specifications on November 10, 1993, and does not create a new or unanalyzed condition. For these reasons, the possibility of a new accident which is different from any already evaluated in the USAR is not created.

Operation of the facility in accordance with the amendment will not involve a significant reduction in a margin of safety. The analyses and evaluations discussed in the safety evaluation demonstrate that all applicable safety analysis acceptance criteria continue to be met for the proposed operating conditions. The change in operating T_{hot} does not involve a significant reduction in a margin of safety because the operating temperature is one of the inherent assumptions that determines the safe operating range defined by the accident analyses, which are in turn protected by the technical specifications. The acceptance criteria for the accident analyses are conservative with respect to the operating conditions defined by the technical specifications. The analyses performed for the power rerate and this proposed change confirm that the accident analyses criteria are met at the revised configuration. Therefore, it is concluded that the proposed change does not involve a reduction in a margin of safety described in the bases to any technical specification.

Based upon the above considerations, the staff concludes that the amendment meets the three criteria of 10 CFR 50.92. Therefore, the staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

6.0 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.

7.0 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 (59 FR 7269). 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.

8.0 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.

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