November 6, 1991

Docket No. 50-482

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

Dear Mr. Withers:

SUBJECT: WOLF CREEK GENERATING STATION - AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. M79923)

The Commission has issued the enclosed Amendment No. 51 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 March 5, 1991.

The amendment changes the Reactor Coolant System (RCS) thermal design flow from the current Technical Specification value of 95,700 gpm/loop to a new value of 93,750 gpm/loop. Other Technical Specification changes related to the reduced thermal design flow include an increase in the setpoint of the Pressurizer Pressure Low Reactor Trip System Instrumentation, revision of Figure 2.1-1, Reactor Core Safety Limits, and various changes to the Bases.

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. 51 to NPF-42
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	PDIV-2/LA	PDIV-2/PM	OGC MAY wrote provisions	PDIV-2/D			
NAME	EPeyton	WReckley	MYoung	SBlack			
DATE	10/28/91	10/29/91	11/1/91	11/5/91			

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cp

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

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 51
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 March 5, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 51, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Suzanne C. Black

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: November 6, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 51

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
2-2	2-2
2-4	2-4
3/4 2-9	3/4 2-9
B 2-1	B 2-1
B 2-2	B 2-2
B 3/4 2-4	B 3/4 2-4
B 3/4 2-6	B 3/4 2-6
B 3/4 4-1	B 3/4 4-1

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1 for four loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

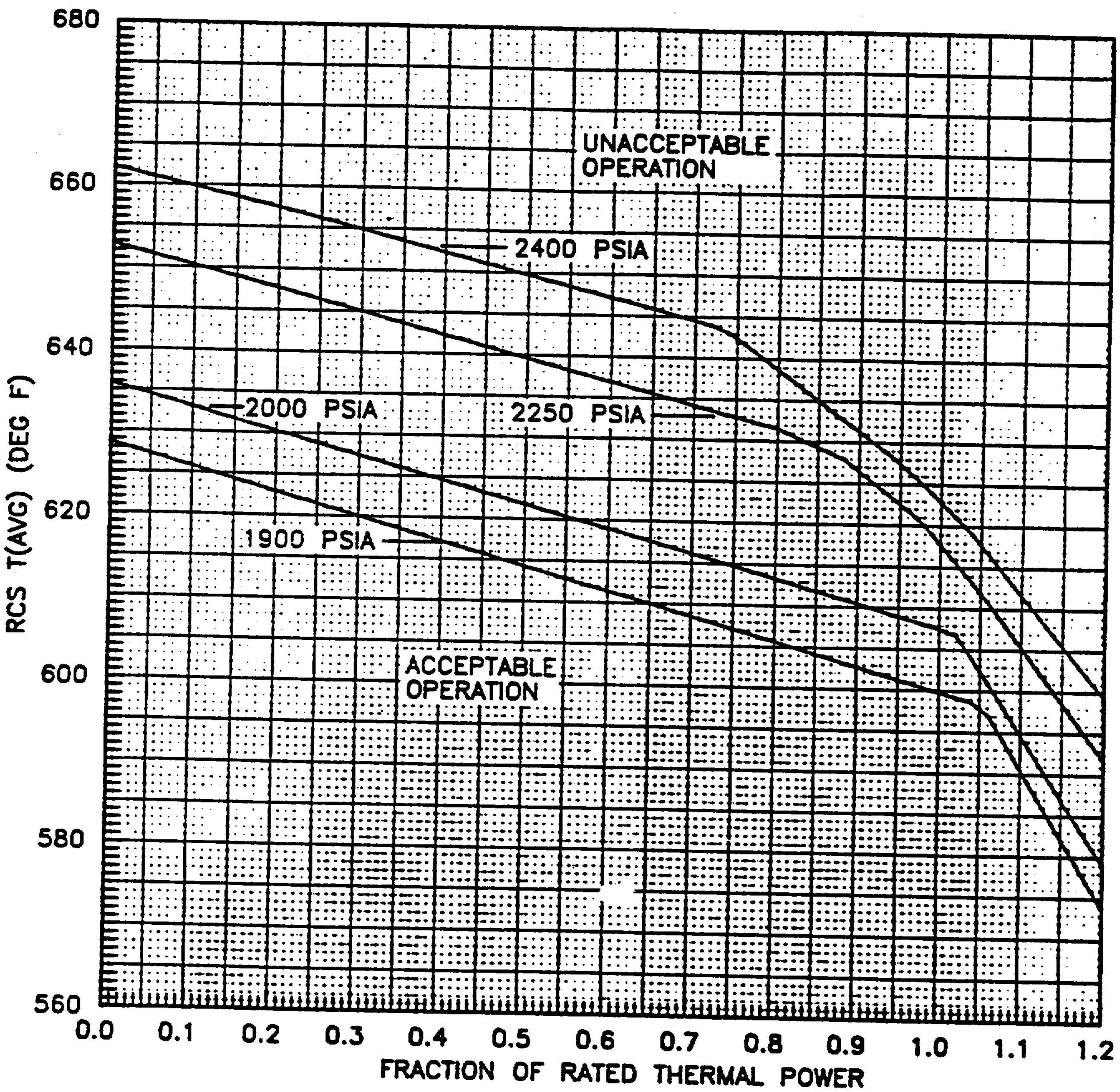

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

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	7.2	3.40	2.49	See Note 1	See Note 2
8. Overpower ΔT	5.5	1.43	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 = 93,750 gpm

MEASUREMENT UNCERTAINTIES OF 2.5% FOR FLOW AND 4.0% FOR INCORE MEASUREMENT OF $F^N_{\Delta H}$ ARE INCLUDED IN THIS FIGURE

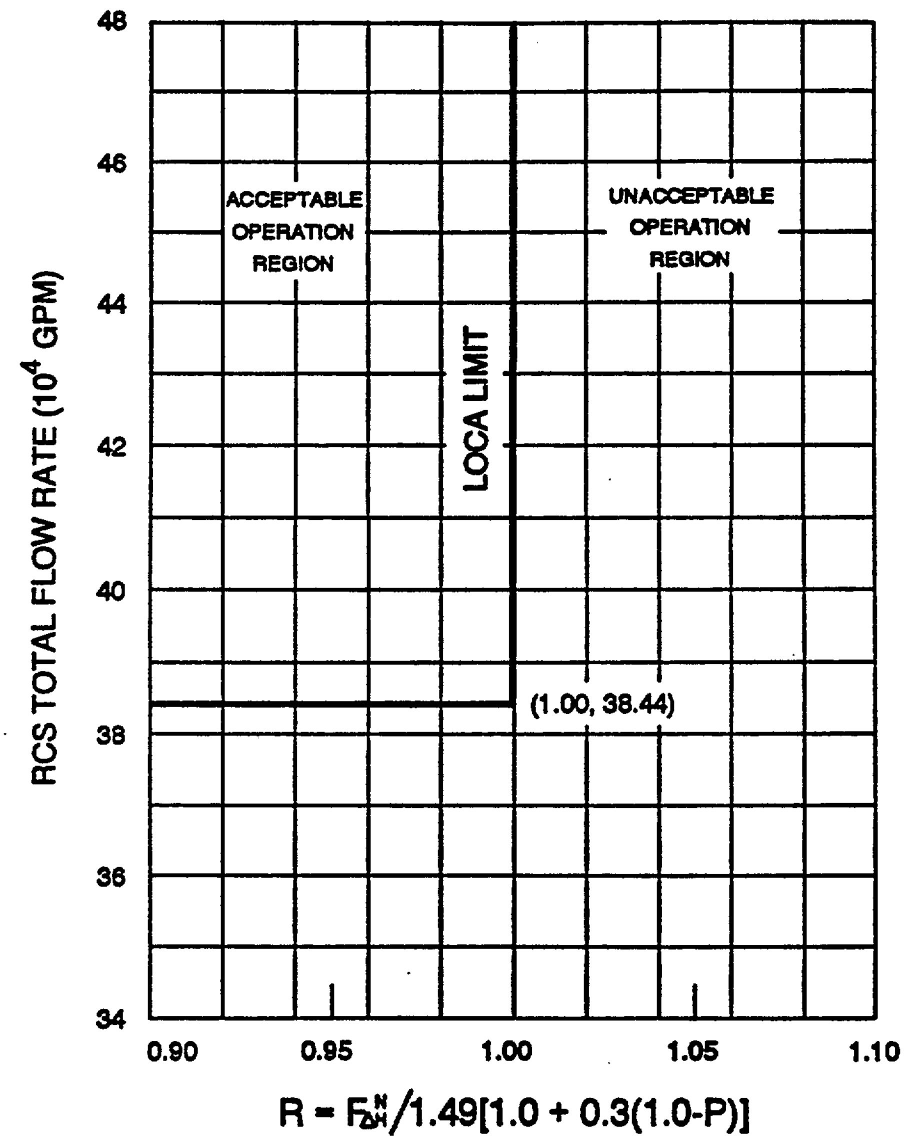

FIGURE 3.2-3
RCS TOTAL FLOW RATE VERSUS R
FOUR LOOPS IN OPERATION

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. 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 a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

   1. A nominal 50% of RATED THERMAL POWER,

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

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

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of Figure 3.2-3:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the most recently obtained value of R obtained per Specification 4.2.3.2, is assumed to exist.

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

4.2.3.5 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.

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

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through DNBR correlations. DNBR correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for the WRB-1 correlation). For plant conditions which fall outside the range of applicability of the WRB-1 correlation, the W-3 correlation is used.

In addition, DNB margin is maintained by performing safety analyses to a higher valve than the correlation limit, called the safety analysis limit DNBR. The margin between the safety analysis limit DNBR and the correlation limit DNBR is used to cover known DNBR penalties and provide margin for design flexibility.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the applicable safety analysis limit DNBR, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1+ 0.3\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

2.1.1 REACTOR CORE (Continued)

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping and valves are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at greater than or equal to 125% (3110 psig) of design pressure, to demonstrate integrity prior to initial operation.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specification 3.1.3.6 are maintained, and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on Figure 3.2-3, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the safety analysis DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 11.4% DNBR, completely offset any rod bow penalties. This is the margin between the correlation DNBR limit (1.17) and the safety analysis limit DNBR (1.32).

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

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.9 was determined from expected power control manuevers over the full range of burnup conditions in the core.

When RCS flow rate and $F_{\Delta H}^{N}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figure 3.2-3. Measurement errors of 2.5% for RCS total flow rate and 4% for $F_{\Delta H}^{N}$ have been allowed for in determination of the design DNBR value.

The measurement error 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 venture which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, an inspection is performed of the feedwater venture 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 shown on Figure 3.2-3. This surveillance also provides adequate monitoring to detect any core crud buildup.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective ACTION is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

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 is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

QUADRANT POWER TILT RATIO (Continued)

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 DNB-related parameters 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 FSAR assumptions and have been analytically demonstrated adequate to maintain a DNBR above the safety analysis limit DNBR (1.32) throughout each analyzed transient. The indicated T_{avg} valve of 592.5°F and the indicated pressurizer pressure value of 2220 psig correspond to analytical limits of 595°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.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the safety analysis limit DNBR (1.32) during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing decay heat even in the event of a bank withdrawal accident; however, single failure considerations require that three loops be OPERABLE. A single reactor coolant loop provides sufficient heat removal if a bank withdrawal accident can be prevented; i.e., by opening the Reactor Trip System breakers.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5 are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

SAFETY VALVES (Continued)

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. NPF-42

WOLF CREEK NUCLEAR OPERATING CORPORATION

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

By letter dated March 5, 1991 (Ref. 1), Wolf Creek Nuclear Operating Corporation (WCNOC) (the licensee), requested an amendment to Facility Operating License NPF-42 for the Wolf Creek Generating Station (WCGS) Unit 1. The proposed amendment presented changes to the Technical Specifications due to proposed modifications to reduce the reactor coolant system (RCS) thermal design flow, to replace the W-3 critical heat flux (CHF) correlation with the WRB-1 CHF correlation, and to increase the low pressurizer pressure reactor trip setpoint limit. These measures have been taken in anticipation of the need to provide compensatory thermal margin to accommodate any future actual RCS flow degradation due to steam generator tube plugging.

2.0 DISCUSSION

The proposed change in RCS thermal design flow (TDF) is from the current Technical Specification value of 95,700 gpm/loop to a new value of 93,750 gpm/loop, a reduction of approximately 2 percent. The proposed reduced RCS flow requirement was chosen to reasonably bound potential future need to account for steam generator tube plugging or sleeving of up to 4 percent of the tubes in each steam generator without requiring extensive reanalysis. The critical heat flux correlation was changed from W-3 to WRB-1 to obtain more margin to offset the proposed decrease in RCS TDF. Also, the low pressurizer pressure reactor trip setpoint safety analysis limit (SAL) was increased from 1860 psia to 1900 psia to ensure protection against vessel exit boiling with reduced RCS flow.

2.1 Core Thermal Limits

In light of the potential decrease in RCS flow, the licensee re-calculated core thermal limits with the WRB-1 critical heat flux correlation. The WRB-1 critical heat flux (CHF) correlation was used in place of the W-3 correlation which was used for analysis documented in the current Updated Safety Analysis Report (USAR). The WRB-1 correlation is less limiting and offsets the proposed decrease in required RCS flow. This CHF correlation has been previously reviewed and approved by the NRC (Ref. 2) and is therefore acceptable. The licensee found that the WRB-1 CHF correlation provides sufficient margin to offset the effects of the proposed RCS thermal design flow. The margin was partially utilized to maintain the existing DNB limits, and the balance is identified as generic DNB margin available for use in future applications. Changing from the W-3 based correlation to the WRB-1 CHF correlation while maintaining the existing DNB core thermal and axial offset limits results in a redefinition of the safety analysis limit DNBR from 1.30 to 1.32. This results in an increase in the generic DNBR margin (i.e., the margin between the WRB-1 CHF correlation limit DNBR (1.17) and the safety analysis limit DNBR (1.32)). We have found this application of the WRB-1 CHF correlation for thermal hydraulic analysis to be acceptable.

The low pressurizer pressure reactor trip setpoint SAL was increased from 1860 psia to 1900 psia in this evaluation. This change was made to ensure that vessel exit boiling limits (VEBL) would not be exceeded during depressurization transients with the reduced RCS flow rate and is therefore acceptable. This reduction in low pressurizer pressure reactor trip setpoint SAL is reflected in this proposed Technical Specification change.

2.2 Evaluation of Non-LOCA Accidents Previously Analyzed

The licensee stated that all non-LOCA transients and accidents included in the USAR were reevaluated for sensitivity to and the potential effects of reduced RCS thermal design flow. Critical statepoints in the original transient analyses, i.e., the transient thermal-hydraulic conditions at the time of minimum DNBR, were identified and reevaluated with a 2 percent flow reduction and reduced inlet temperatures using WRB-1 CHF correlation. In all cases, the evaluations gave acceptable results when compared with the revised SAL DNBR of 1.32.

For the transients where the critical statepoint conditions fell outside the range of applicability of the WRB-1 CHF correlation, the statepoints were reevaluated using the W-3 CHF correlation assuming a 2 percent reduction in the flow rate. Sufficient accident specific margin was found to be available for these transients to accommodate both the penalty from reduced flow and the increased generic DNB margin included in this evaluation.

2.3 LOCA and LOCA Related Analysis

Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks with the Reactor Coolant Boundary - (USAR 15.6.5)

(a) Large Break LOCA

The licensee stated that the large break LOCA and long-term core cooling calculations were previously analyzed in the USAR for a reduced RCS thermal design flow (TDF) of 93,200 gpm/loop and steam generator plugging levels of up to 10 percent. Therefore the request for a reduction in TDF from 95,700 gpm/loop to a reduced value of 93,750 gpm/loop is bounded by the 93,200 gpm/loop analysis. We therefore find this acceptable.

(b) Small Break LOCA

The licensee stated that the originally licensed small break LOCA analysis for WCGS was performed using the Westinghouse WFLASH Evaluation Model and assumed a thermal design flow of 95,700 gpm/loop with no steam generator tube plugging. Subsequent analysis were performed per Three Mile Island Action Plan Item II.K.3.31 with the NOTRUMP small break LOCA evaluation model to demonstrate that the WFLASH evaluation model was bounding. This analysis was reviewed and approved by the NRC for application to WCGS (Ref. 5). Subsequent generic analyses have been performed using the NOTRUMP code to assess the effects of RCS thermal design flow of 93,200 gpm/loop and 10 percent steam generator tube plugging on small break LOCAs (Refs. 6 and 7). These analyses have shown that: (1) steam generator plugging levels up to 10 percent continue to provide effective heat sink to the primary side with reduced TDF, and (2) tube plugging levels of up to 15-20 percent would have no effect on core uncovery and therefore no effect on peak cladding temperatures.

Based on the above, it is concluded that the 2 percent reduction of RCS thermal design flow to 93,750 gpm/loop will have no significant effect on the PCT for small break LOCA which is already well below that for the large break LOCA, i.e., 1,917.6°F versus 2,163.5°F.

3.0 EVALUATION

As a result of the modifications associated with the reduction in RCS thermal design flow, increase in low pressurizer pressure reactor trip setpoint limit and change in the critical heat flux correlation (CHF) from the W-3 to the WRB-1 CHF, changes to the plant's Technical Specifications were proposed. The following Technical Specifications were examined.

Figure 2.1-2, page 2-2 - "Reactor Core Safety Limit - Four Loops in Operation"

The parameter of pressure in the RCS Tavg versus Fraction of Rated Power curve was redrawn to increase the 1860 psia value to 1900 psia. This was to reflect the modifications in TDF and the CHF correlation and a shift in the steam generator safety valve actuation line. This is acceptable as discussed above in Section 2.0.

Table 2.2-1, page 2-4, "Reactor Trip System Instrumentation Trip Setpoints"

Functional Unit 9 - "Pressurizer Pressure Low" was modified. The trip setpoint was changed from equal or greater than 1875 psig to equal or greater than 1915 psig and the allowable value was changed from equal or greater than 1866 psig to equal or greater than 1906 psig. The footnote for loop design flow was changed from 95,700 gpm to 93,750 pgm. These changes were found to be acceptable as discussed above in Section 2.0.

Figure 3.2-3, page 3/4 2-9, - "RCS Total Flow Rate Versus R - Four Loops in Operation"

The total flow rate value of 392,400 gpm was changed to 384,440 gpm. This is acceptable as it is four times the new TDF value of 93,750 gpm/loop and includes the flow measurement uncertainty value of 2.5 percent.

Bases 2.1.1 Reactor Core, page B 2-1

This page deletes reference to the W-3 CHF correlation. An insert explains the use of the WRB-1 CHF correlation in place of the W-3 CHF correlation. This editorial change is acceptable as discussed above in Section 3.0.

Page B 3/4 2-4, Heat Flux Hot Channel Factor, and RCS Flow Rate and Nuclear Hot Channel Factors

A portion of a sentence was changed from "below the design DNBR value" to "below the safety analysis DNBR value." Also reference to the generic margin of 9.1 percent DNBR was eliminated together with a listing of the margins and replaced by a generic margin of 11.4 percent DNBR. A sentence was added which stated - "This is the margin between the correlation DNBR limit (1.17) and the safety analysis limit DNBR (1.32)." These editorial changes are acceptable as discussed in Section 2.0.

Bases 3/4.2.5, DNB Parameters, page 3/4 2-6

A sentence with reference to a "minimum DNBR of 1.30" was changed to "DNBR above the safety analysis limit DNBR (1.32)." This editorial change is acceptable as discussed in Section 2.0.

Bases 3/4.4.1, Reactor Coolant Loops and Coolant Circulation, page B 3/4 4-1

A sentence with "DNBR above 1.30" was changed to "DNBR above the safety analysis limit DNBR (1.32)." This editorial change is acceptable as discussed in Section 2.0.

The impact of changing: (1) the RCS thermal design flow, (2) the low pressurizer pressure reactor trip setpoint limit, and (3) the critical heat flux correlation from W-3 to WRB-1 for the Wolf Creek plant on the UFSAR Chapter 15 accidents has been evaluated by the licensee. The staff has found that the former conclusions in the UFSAR remain valid and the Technical Specification changes have been determined to be acceptable as described in Sections 2.0 and 3.0.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 13673). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

REFERENCES

1. Letter, Forrest T. Rhodes, Wolf Creek Nuclear Operating Corporation, to NRC, March 5, 1991.

2. Letter, John Stolz, NRC, to C. Eicheldinger, Westinghouse Electric Corporation, April 19, 1978.

3. SLNRC 86-01, "Steam Generator Single-Tube Rupture Analysis for SNUPPS Plants Callaway and Wolf Creek," dated January 8, 1986.

4. WCAP-10961-P, "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment," October 1985.

5. Letter from Paul W. O'Connor, NRC, to Glenn L. Koester, KG&E, November 17, 1986.

6. Ciani, S., et al., "Simulation of Small Break Type Behavior of PUN and SPES using the NOTRUMP Code," Proceedings of the Specialist Meeting on Small Break LOCA Analyses in LWR's, Pisa, Italy, June 1985.

7. Lee, N., "Limiting Countercurrent Flow Phenomena in Small Break LOCA Transients," Proceedings of the Specialist Meeting on Small Break LOCA Analyses in LWR's, Pisa, Italy, June 1985.

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