April 4, 1996

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

SUBJECT: WOLF CREEK GENERATING STATION - AMENDMENT NO. 99 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. M94882)

Dear Mr. Carns:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated March 8, 1996, as supplemented by letter dated March 26, 1996.

The amendment reduces the calculated thermal design flow of the reactor coolant system and increases the trip setpoint of the low pressurizer pressure.

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

James C. Stone, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-482

Enclosures: 1. Amendment No. 99 to NPF-42
2. Safety Evaluation

cc w/encls: See next page

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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. 99
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 8, 1996, as supplemented by letter dated March 26, 1996, 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. 99, 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

James C. Stone

James C. Stone, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 4, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 99

FACILITY OPERATING LICENSE NO. NPF-42

DOCKET NO. 50-482

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
2-2	2-2
2-4	2-4
3/4 2-16	3/4 2-16

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.

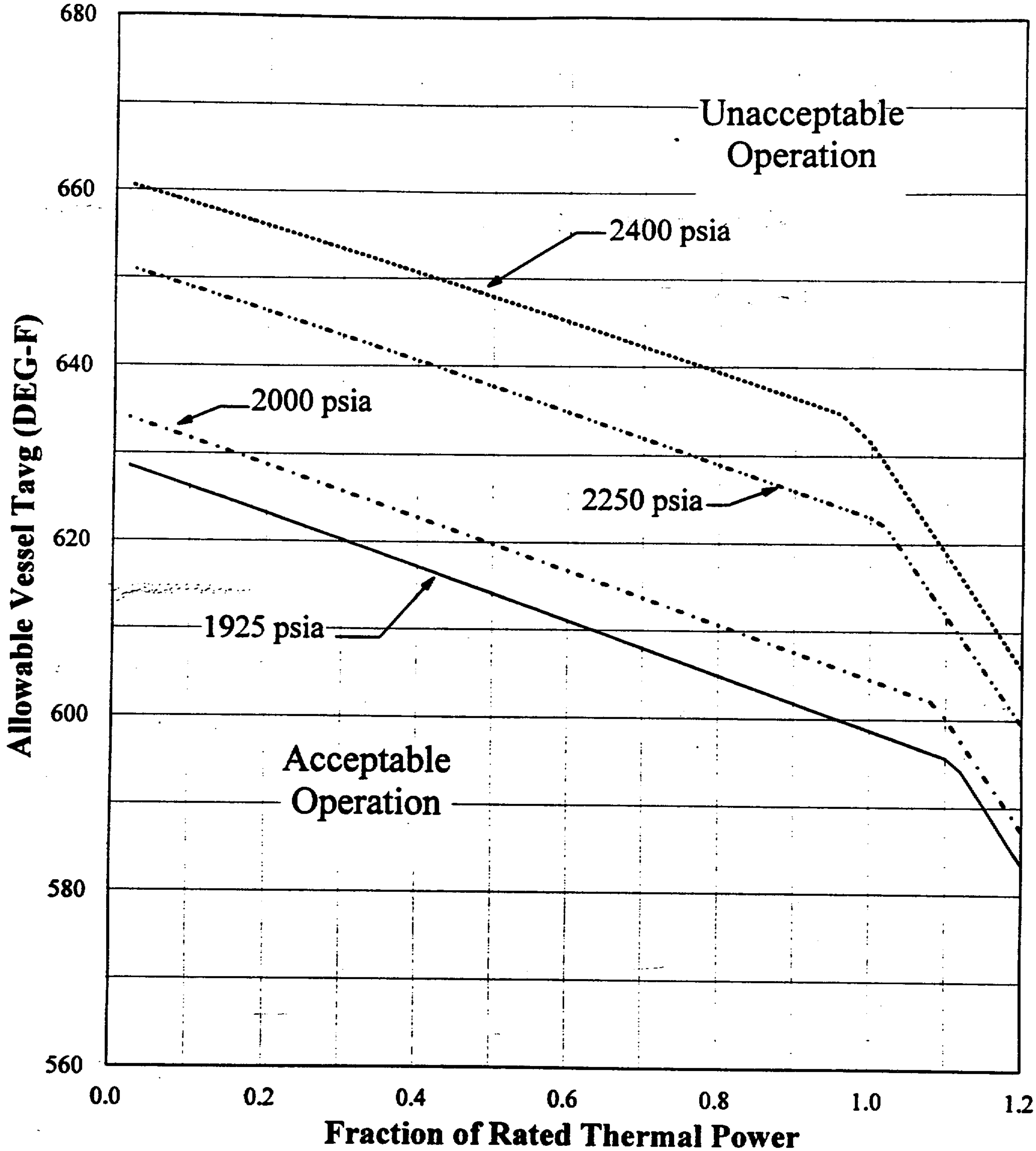

FIGURE 2.1-1
REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

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	$\leq 10^5$ cps	$\leq 1.6 \times 10^5$ cps
7. Overtemperature ΔT	7.0	5.39	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	≥1940 psig	≥1931 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 = 90,324 gpm

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}	≤590.5°F
2. Indicated Pressurizer Pressure	≥2220 psig*
3. Reactor Coolant System Flow Rate	≥37.1 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.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 99 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 8, 1996, as supplemented by letter dated March 26, 1996, Wolf Creek Nuclear Operating Corporation (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-42) for the Wolf Creek Generating Station. The proposed changes would revise Technical Specification (TS) Figure 2.1-1, "Reactor Core Safety Limit - Four Loops in Operation," Table 2.2-1, "Reactor Trip System Instrumentation Setpoints," and Table 3.2-1, "DNB Parameters."

Specifically, the TS changes would:

1. Modify Figure 2.1-1, Reactor Core Safety Limit - Four Loops in Operation to account for reduction in TS reactor coolant system (RCS) flow.

2. Change Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints as follows because of the reduction in TS RCS flow:

   The functional unit 9, Pressurizer Pressure-Low trip setpoint, is changed from ≥1915 to ≥1940 psig and the allowable value is changed from ≥1906 to ≥1931 psig.

   The footnote **, Loop design flow, is changed from 93,600 to 90,324 gpm.

3. Modify Table 3.2-1, DNB Parameters, for the reactor coolant flow rate from $\geq 38.4 \times 10^4$ to $\geq 37.1 \times 10^4$ gpm to assure that TS flow margin exist to support future cycles of operation.

The March 26, 1996, supplemental letter forwarded information on the results of the analyses performed and did not change the staff's original no significant hazards determination published in the Federal Register on March 13, 1996 (61 FR 10389).

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2.0 BACKGROUND

The WCGS Cycle 9 fuel loading pattern was designed to be a low leakage loading pattern (LLLP). The loading pattern was optimized such that it would minimize the number of new fuel assemblies to be purchased and reduce the neutron fluence at the core periphery. During the Cycle 9 reload design process, concerns were raised about the tendency for large power gradients at the core periphery in LLLPs to influence hot leg streaming. An increase in hot leg streaming could result in a biased T_{hot} measurement such that the indicated T_{hot} would be greater than the actual bulk temperature in the hot leg.

Because RCS flow is calculated based on a flow calorimetric which is dependent on the hot leg temperature measurement, an increase in hot leg streaming could lead to a calculated RCS flow below the value specified in the TS.

3.0 EVALUATION

The license amendment request proposed to revise the Wolf Creek Generating Station (WCGS) Technical Specifications to allow plant operation at 100 percent rated thermal power (RTP) with a 3.5 percent reduction in thermal design flow (TDF) and an increase in the low pressurizer pressure trip setpoint. This revision in thermal design flow represents a decrease in TDF from the current value of 374,400 gpm to 361,296 gpm. The corresponding reactor coolant average temperature (T_{avg}) will remain at the current value of 586.5°F. However, the decreased TDF will result in a slight decrease in the core inlet temperature (T_{in}) and an approximate 1°F increase in the hot leg temperature (T_{hot}). The low pressurizer pressure trip setpoint will be raised from the safety analysis limit (SAL) of 1915 psig to 1940 psig, a 25 psi increase, to preclude the occurrence of departure from nucleate boiling (DNB), ensuring that core thermal protection is provided for all conditions of operation.

The reduction in TDF increases the enthalpy of the coolant exiting the reactor vessel. Because reactor vessel delta T is used as an indicator of core power, it is necessary to limit the enthalpy of the coolant exiting the vessel to a subcooled state. The over temperature and over pressure delta T trip functions provide assurance that the exit enthalpy conditions are maintained. Because the decrease in TDF impacts the over temperature delta T at the low end of the allowable pressurizer pressure range, the pressurizer pressure trip setpoint is being raised from 1915 psig to 1940 psig. This will assure the vessel exit boiling limits are protected by the existing over temperature delta T trip setpoint. Therefore, the staff finds the proposed change to the pressurizer pressure trip setpoint acceptable.

The Limiting Condition for Operation (LCO) flow value listed in Technical Specification Table 3.2-1 is, by definition, the minimum measured flow. Minimum measured flow is defined as 102.5 percent of TDF (using a flow measurement uncertainty of 2.5 percent). The proposed new TDF of 370,328 gpm, is rounded up to 371,000 gpm to provide additional margin. Therefore, the technical specification LCO flow value listed in Table 3.2-1 will change from 384,000 gpm to 371,000 gpm (approximately a reduction of 3.5 percent).

The Updated Safety Analysis Report (USAR) transient and accident analyses have been evaluated by the licensee using the 3.5 percent reduction in flow (total flow 371,000 gpm) and assuming operating parameters consistent with a T_{avg} equal to 588.4°F.

The USAR Chapter 15 non-LOCA events are categorized in the following sections:

- 15.1 Increase in Heat Removal by the Secondary System
- 15.2 Decrease in Heat Removal by the Secondary System
- 15.3 Decrease in Reactor Coolant System Flow Rate
- 15.4 Reactivity and Power Distribution Anomalies
- 15.5 Increase in Reactor Coolant System Inventory
- 15.6 Decrease in Reactor Coolant System Inventory
- 15.7 Radioactive Release from a Subsystem or Component

With the exception of USAR Section 15.7, each USAR section is further classified into RCS heatup events or RCS cooldown events, preparing a basis from which each event's sensitivity to the RCS flow reduction may be determined. The heatup events are generally comprised of USAR Sections 15.2, 15.3, portions of 15.4 and 15.6. The cooldown events are comprised of USAR Sections 15.1, and portions of 15.4 and 15.5.

The following events were either analyzed or evaluated by the licensee based on the proposed TS changes:

Feedwater Malfunction (USAR 15.1.2)
Excessive Increase in Secondary Steam Flow (USAR 15.1.3)
Inadvertent Opening of a Steam Generator Relief or Safety Valve (USAR 15.1.4)
Main Steam Line Break (USAR 15.1.5)
Loss of Electrical Load/Turbine Trip (USAR 15.2.3)
Loss of Non-Emergency AC Power (USAR 15.2.4)
Loss of Normal Feedwater (USAR 15.2.7)
Feedwater Line Break (USAR 15.2.8)

Partial Loss of Forced Reactor Coolant Flow (USAR 15.3.1)
Complete Loss of Forced Reactor Coolant Flow (USAR 15.3.2)
Reactor Coolant Pump Shaft Seizure (USAR 15.3.3)
Reactor Coolant Pump Shaft Break (USAR 15.3.4)

Uncontrolled RCCA Withdrawal from Subcritical or Low Power Startup Condition (USAR 15.4.1)
Uncontrolled RCCA Withdrawal at Power (USAR 15.4.2)
Rod Cluster Control Assembly Misoperation (USAR 15.4.3)
Startup of an Inactive Loop (USAR 15.4.4)

Boron Dilution (USAR 15.4.6)
Loading an Operation of a Fuel Assembly in an Improper Position (USAR 15.4.7)
RCCA Ejection Event (USAR 15.4.8)

Inadvertent Actuation of the ECCS During Power Operation (USAR 15.5.1)
CVCS Malfunction that Increases Reactor Coolant Inventory (USAR 15.5.2)

Accidental Depressurization of the Reactor Coolant System (USAR 15.6.1)
Steam Generator Tube Rupture (USAR 15.6.3)
Loss-of-Coolant Events (LOCA) (USAR 15.6.5)
- Large Break LOCA Analysis
- Small Break LOCA Analysis
- Post-LOCA Long-Term Core Cooling
- Hot Leg Switchover to Prevent Potential Boron Precipitation
- Radiological Consequences
- Rod Ejection Mass and Energy Release for Dose Calculation
- Blowdown Reactor Vessel and Loop Forces

Mass and Energy Releases
- LOCA Mass and Energy Release Evaluation (USAR 6.2 and 6.3)
- Secondary Pipe Rupture Mass and Energy Release (Inside and Outside Containment)
- IE-79-22, Control and Protection Interaction (SLB/RWAP)

The results of the analyses listed above, were evaluated by the licensee. For those events where the DNBR or pressure limit was the limiting parameter, these requirements were within the acceptable limits. The limiting DNBR accident was the Complete Loss of Forced Reactor Coolant Flow (USAR 15.3.2) for which the DNBR was 1.77 compared to the required value of 1.76 using the WRB-2 correlation (Ref. 2). The limiting pressure event was the Loss of Electrical Load/Turbine trip (USAR 15.2.3) for which the pressure value from the analysis was 2,735 psia compared to the acceptance value of 2,750 psia (110 percent of the RCS design pressure). Other considerations, such as peak clad temperature (PCT), mass and energy releases, and radiological consequences were found to be within the acceptable limits. The PCT for the large break LOCA was 1,916°F and 1,510°F for the small break LOCA. These PCT values are acceptable as they are below the 2,200°F requirement. The mass and energy effects on containment were analyzed and were found to be bounded. The radiological consequences for the primary and secondary sides were found to have no significant changes compared to the current USAR. Therefore, we find the results of the analyses listed above to be acceptable.

The footnote on Table 2.2-1 that lists the loop design flow is changed from 93,600 gpm to 90,324 gpm. This is the flow in one reactor coolant loop and is derived by dividing the revised TDF (excluding the flow measurement uncertainty of 2.5 percent) in all loops (316,296 gpm) by the number of loops (4). This results in the flow in one reactor coolant loop of 90,324 gpm. The staff finds this change acceptable.

Figure 2.2-1 is being changed to account for the 3.5% decrease in reactor coolant system flow and the 25 psi increase in the pressurizer pressure trip setpoint to ensure continued protection from DNB. The staff finds that this change is consistent with the licensee's safety analysis and acceptable.

4.0 SUMMARY

Non-LOCA and LOCA safety analyses and evaluations were performed to confirm the acceptability of a 3.5 percent reduction in TDF and a 25 psi increase in the Low Pressurizer Pressure Trip setpoint. Each event assumed initial conditions for Cycle 9 consistent with those listed in Table 1 from Reference 1 and assumed the appropriate uncertainties and steady state errors for core power, RCS temperature, RCS pressure and RCS flow rate as listed in Table 3 from Reference 1. The safety analyses were either performed or evaluated at the lower and upper bound conditions, depending on which was limiting. By performing the analyses at these conditions, the proposed condition is assured to be bounded.

The DNB analyses were performed in accordance with Westinghouse's revised thermal design procedure (RTDP) using the WRB-2 correlation. The uncertainties and steady state errors assumed in the initial condition for these analyses were treated statistically in the DNB analysis and are therefore initiated from nominal conditions in the event analyses. DNB analyses which fall outside the range of applicability of the RTDP methodology were analyzed utilizing the W-3 correlation and therefore were initiated from the same conditions as the remainder of the non-LOCA event analyses.

Approved codes were used in the analyses and all the applicable acceptance criteria for each event were found to be met. Therefore, we find the results of the analyses and evaluations and the changes to the Technical Specifications to be acceptable.

5.0 EXIGENT CIRCUMSTANCES

The Commission's regulations, 10 CFR 50.91, contain provisions for issuance of amendments when the usual 30-day public notice period cannot be met. One type of special exception is an exigency. An exigency is a case where the staff and licensee need to act promptly (before the expiration of a 30-day prior comment period).

Under such circumstances, the Commission notifies the public in one of two ways: by issuing a Federal Register notice providing an opportunity for hearing and allowing at least two weeks for prior public comments, or by issuing a press release discussing the proposed changes, using the local media. In this case, the Commission used the first approach.

The need for a license amendment became apparent when the licensee completed the reload core design in mid January 1996. However, before the amendment could be submitted, a reanalysis of the accidents in the FSAR was required to ensure all acceptance criteria continue to be met. These analyses required a significant period of time to complete (from mid January until the end of February). Further compressing the time was the fact that Wolf Creek entered the eighth refueling outage a month early (originally scheduled for March 2, 1996, but moved up to February 2, 1996) because of the icing problems encountered and the subsequent failure of five control rods to fully insert on the ensuing manual trip of the reactor.

Whether or not the predicted hot leg streaming will cause the calculated RCS flow to be below the current TS value will not be known until Wolf Creek restarts and reaches 100 percent power. However, if the calculated RCS flow is below the TS value, the TS requires power to be reduced to less than 50 percent within 2 hours and less than 5 percent within 72 hours. Without the timely issuance of this amendment, operation at Wolf Creek could be severely restricted. Plant restart is currently scheduled for March 30, 1996. To avoid the potential for an unnecessary plant shutdown, this amendment is needed before reaching 100 percent power. Therefore, the NRC staff finds that exigent circumstances exist in that the Commission and licensee must act quickly and that time does not allow publication of a notice allowing 30 days prior for public comment.

The NRC staff has reviewed the circumstances surrounding the amendment request and finds that the circumstances could not have been avoided and the licensee made a timely request for the amendment. Therefore, the staff finds that the license amendment may be issued in an exigent manner pursuant to 10 CFR 50.91(a)(6).

There were no public comments in response to the notice published in the Federal Register.

6.0 BASIS FOR 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.

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The amendment reflects revised core design parameters affected by the Cycle 9 core reload geometry, and instrumentation setpoint changes needed to ensure accurate measurement of reactor thermal power in order to allow the unit to operate at rated thermal power during Cycle 9. Each USAR Chapter 15 event was evaluated to determine the impact of the reduction in thermal design flow.

The events in which the margin to the acceptance criteria was decreased were reanalyzed to support the 3.5 percent flow reduction. Generally, the RCS heat-up events fall into this category as the reduction in RCS flow results in decreased heat removal capacity. Evaluations of these events were performed using bounding core state parameters based on the previous Safety Analysis submitted in support of the WCGS power rerate program, approved in WCGS Technical Specification Amendment 69. Results of the analyses and evaluations performed for the reduction in thermal design flow for Cycle 9 indicate that all acceptance criteria for USAR Chapter 15 events continue to be met.

Therefore, the probability of occurrence and the consequences of an accident evaluated previously in the USAR are not increased.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The amendment does not change the method and manner of plant operation, nor is any new equipment being installed. Neither the proposed reduction in thermal design flow nor the increase in the low pressurizer pressure trip setpoint will create the possibility of an event of a different type than previously evaluated in the USAR.

Also, the changes are bounded by the current conditions with respect to system dynamic loading, environmental equipment qualification, and rejection of heat to the ultimate heat sink. These analyses are bounded by the current analyses due to the conclusion that the mass and energy releases will not be impacted by the proposed change. This conclusion is also based on the fact that the current operating conditions bound the proposed operating conditions with respect to the secondary system operating parameters.

Therefore, the changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The amendment does not change the plant configuration in a way that introduces a new potential hazard to the plant and does not involve a significant reduction in the margin of safety. The analyses and evaluations discussed in the safety evaluation demonstrate that all applicable design criteria continue to be met for the changes.

The low pressurizer pressure trip setpoint is chosen at a conservatively low value (1885 psig) for the safety analyses. The safety margin (to prevent DNB) is provided by setting the technical specification limit for the low pressurizer pressure trip setpoint at its current value of 1915 psig. Increasing this reactor trip setpoint 25 psi (from 1915 psig to 1940 psig) results in a net benefit to all analyses which assume its use, as well as off setting a potential reduction in the margin of safety for this parameter, caused by the reduction in TDF. Therefore, the current safety analysis limit of 1885 psig will continue to be used in the WCGS event analyses.

Therefore, it is concluded that the margin of safety, as described in the bases to any technical specification, is not reduced.

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.

7.0 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.

8.0 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 made a final finding that the amendment involves no significant hazards consideration. 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.

9.0 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.

10.0 REFERENCES

1. Letter from O. Maynard, Wolf Creek Nuclear Operating Corporation, to USNRC, dated March 8, 1996.

2. Letter from O. L Maynard, Wolf Creek Nuclear Operating Corporation, to USNRC, dated March 26, 1996.

Principal Contributor: H. Balukjian

Date: April 4, 1996