WOLF CREEK
NUCLEAR OPERATING CORPORATION

Robert C. Hagan
Vice President Nuclear Assurance

June 7, 1994

VT 94-0006

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Washington, D. C. 20555

Subject: Docket No. 50-482: Revision to Technical Specification 2.2, Table 2.2-1

Gentlemen:

This letter transmits an application for amendment to Facility Operating License No. NPF-42 for Wolf Creek Generating Station (WCGS). This license amendment request proposes revising Technical Specification Table 2.2-1, Reactor Trip System Instrumentation Setpoints, to change the OTΔT axial flux difference (AFD) limits to reflect results of the Cycle 8 core maneuvering analysis.

Attachment I provides a safety evaluation including a description of the proposed change. Attachment II provides a no significant hazards consideration determination and Attachment III provides an environmental impact determination. The specific change to the technical specifications proposed by this request, as well as the requested administrative changes, are provided in Attachment IV.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Kansas State Official. This proposed revision to the WCGS Technical Specifications will be fully implemented following formal NRC approval, and prior to startup from the seventh refueling outage.

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If you have any questions concerning this matter, please contact me at (316) 364-8831, extension 4553, or Mr. Richard D. Flannigan, at extension 4500.

Very truly yours,

Robert C. Hagan

Robert C. Hagan

RCH/jra

Attachments I - Change Description and Safety Evaluation
II - No Significant Hazards Consideration Determination
III - Environmental Impact Determination
IV - Proposed Technical Specification Change

cc: G. W. Allen (KDHE), w/a
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STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Robert C. Hagan, of lawful age, being first duly sworn upon oath says that he is Vice President Technical Services of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the content thereof; that he has executed that same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

NOTARY PUBLIC

By Robert C. Hagan

Robert C. Hagan
Vice President
Technical Services

SUBSCRIBED and sworn to before me this 7th day of June, 1994.

Sandra L. Elliott

Notary Public

Expiration Date 5/14/95

ATTACHMENT I

SAFETY EVALUATION

Safety Evaluation

Proposed Change

This license amendment request proposes to revise Technical Specification 2.2, Table 2.2-1, Reactor Trip System Instrumentation Setpoints. This revision changes values for the overtemperature delta-temperature (OTΔT) axial flux difference (AFD) limits to account for revised peaking limits resulting from the Cycle 8 core design analysis. The specific changes proposed are:

- Increase OTΔT Z from its current value of 4.86 to 5.39,
- Increase the positive slope of OTΔT f1(ΔI) from its current value of 1.56% ΔT/%ΔI to 1.84% ΔT/%ΔI,
- Reduce the OTΔT f1(ΔI) negative breakpoint from -25% ΔI to -23%ΔI,
- Increase the negative slope of OTΔT f1(ΔI) from its current value of 1.8% ΔT/%ΔI to 2.27% ΔT/%ΔI, and
- Reduce the allowable value of OTΔT from its current value of 1.9% ΔT Span to 1.3% ΔT Span.

Background/System Description

The WCGS power rerate program was performed during the middle of Cycle 7 to increase the reactor thermal power by 4.5%, i.e., from 3411 MWt to 3565 MWt [3]. Along with the power rerate, plant operation changes initially included a 5 °F T_{hot} reduction to extend steam generator life. During the power ascension to 3565 MWt, however, WCGS was unable to achieve 3565 MWt before the turbine control valves reached a valves-wide-open condition. The subsequent removal of the 5 °F T_{hot} reduction further increased power output [4].

Safety analyses conducted for the power rerate in Cycle 7 are applicable to continued operation at 3565 MWt in Cycle 8. The Cycle 8 maneuvering analysis results require the AFD limits to be changed to decrease the allowed operating limits. The change also affects the OTΔT setpoint uncertainty, so that values listed in Table 2.2-1 of the WCGS Technical Specifications must be changed.

Evaluation

Analyses and evaluations were performed to assess the impact of continued operation at 3565 MWt with revised AFD limits. It was concluded that the changes could be implemented with minimal impact to Updated Safety Analysis Report (USAR) [2] analyses and to the Technical Specifications [1]. This safety evaluation supports the proposed change through evaluation of the impact on the core thermal-hydraulic analyses, core design, and all LOCA and non-LOCA events. A discussion of each of these items is provided in this section. Other items not specifically cited in this safety evaluation have been reviewed and have been found to be bounded by the evaluations performed for the power rerate program [3].

Table 1 presents the revised operating parameters proposed for Cycle 8. Table 1 also lists the range of temperatures analyzed for the power rerate program [3], for comparison.

The technical specification changes necessary for Cycle 8 operation are summarized in Table 2. Table 2 provides the current and proposed revised values for the OTΔT AFD limits, and a brief explanation for each change.

Thermal-Hydraulic Analysis

A complete description of the thermal-hydraulic methods used by the Wolf Creek Nuclear Operating Corporation (WCNOC) for DNB evaluations is provided in the license amendment request for the proposed Cycle 7 rerate of WCGS [3].

No changes to the thermal-hydraulic methods and core thermal limits are required due to the revised AFD limits proposed in this amendment request. The limiting ANS Condition II transient with respect to departure from nucleate boiling (DNB), the Loss of Flow event, has been reviewed to ensure that minimum DNB ratio requirements are maintained. It was determined that the thermal-hydraulic analysis results were not affected by the revised AFD limits.

Core Design

The operational and transient limits/setpoints applicable for Cycle 8 operation at 3565 MWt with Tavg = 586.5 °F have been examined. Specifically, the allowable AFD limits versus technical specification power limits and the OTΔT/OPΔT trip limits were examined.

The AFD limits are set based upon the results of a maneuvering analysis as described in the WCGS Reload Safety Evaluation Methodology [5]. The maneuvering analysis was performed for Cycle 8 operation. As a result of the calculation, the allowable ΔI limits at 100% power were revised to more restrictive limits than are currently allowed.

The OTΔT and OPΔT trip limits are established to protect Centerline Fuel Melting and Reactor Protection System DNB limits. These limits are established based upon the results of the maneuvering analysis. The maneuvering analysis includes pin peaking margin calculations based on bounding xenon transients. Additionally, boron dilution and overcooling transients are specifically modeled to verify OTΔT/OPΔT trip limits. This analysis was performed for Cycle 8 operation. The resultant OTΔT f(ΔI) breakpoints and slopes produce a more restrictive operating band than is currently allowed. The OPΔT limit remains unchanged.

The cycle specific core kinetics parameters checked in the Reload Safety Analysis Checklist were evaluated for Cycle 8 operation. Key safety parameters including moderator temperature coefficient, shutdown margin, Doppler coefficients and end-of-life trip reactivity were calculated and found to be bounded by the assumed values used in the safety analyses.

Non-LOCA Analyses

A review of the USAR Chapter 15 non-LOCA accidents was performed to determine the transients affected by the revised AFD limits. It was determined that none of the non-LOCA accidents are affected by the revised AFD limits.

LOCA and LOCA Related Evaluations

Since the AFD limits are not used in the small break or large break LOCA analyses, the proposed change does not affect the results originally calculated for Cycle 7 operation. Therefore, the LOCA and LOCA related evaluations performed for the WCGS power rerate program remain valid for Cycle 8 operation.

Accidents of a Different Type

The revised AFD limits for Cycle 8 operation do not create the possibility of any new accidents of a type not previously considered in the USAR. The USAR Chapter 15 accident analyses assume initial conditions resulting in the worst case conditions consistent with the parameters listed in Table 1. Therefore, proposed operation is bounded by previous analyses and does not create a new or unanalyzed condition.

Conclusions

The results of this safety evaluation confirm the acceptability of continued plant operation at 3565 MWt. This justification is based on the application of revised AFD limits versus technical specification power limits. The proposed change is necessary to account for revised peaking limits resulting from the Cycle 8 core design analysis. Evaluations and analyses confirm no increase in transient specific fuel rod failure and support the conclusion that all safety analysis acceptance criteria continue to be met.

The proposed change to Technical Specification Table 2.2-1 does not involve an unreviewed safety question because operation of the WCGS with this change would not:

1. Increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Operation at 3565 MWt does not affect any of the mechanisms postulated in the USAR to cause 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 accidents previously evaluated in the USAR are not increased.

2. Create a possibility for an accident or malfunction of a different type than previously evaluated in the USAR. There is no new type of accident or malfunction being created and the method and manner of plant operation remains unchanged. The accidents assumed to occur at the current license conditions are the same as those for the proposed conditions. The proposed changes do not change the plant configuration in a way that introduces a new potential hazard to the plant. For this reason, the possibility of a new accident that is different from any already evaluated in the USAR is not created.

3. Reduce the margin of safety as defined in the bases for any Technical Specification. The analyses and evaluations discussed in this safety evaluation demonstrate that all applicable USAR acceptance criteria continue to be met for the proposed operating conditions. Therefore, it is concluded that the margin of safety as described in the bases to any Technical Specification is not reduced.

Based on the above discussions and the no significant hazards consideration determination presented in Attachment II, the proposed change does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report; or create the possibility for an accident or a malfunction of a different type than any previously evaluated in the safety analysis report; or reduce the margin of safety as defined in the basis for any technical specification. Therefore, the proposed change does not adversely affect or endanger the health or safety of the general public or involve a significant safety hazard.

References

1. NUREG 1136, "WCGS Technical Specifications."

2. Wolf Creek Generating Station Updated Safety Analysis Report.

3. NA 93-0001, "Proposed Revision to License and Technical Specifications for Power Rerate," letter from R. C. Hagan to U.S. NRC, January 5, 1993.

4. NA 94-0018, "Proposed Revision to Technical Specifications to Revise Power Rerate Parameters," letter from R. C. Hagan to U.S. NRC, February 7, 1994.

5. NSAG-007, Rev. 0, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station," W. S. Kennamore, et. al., January, 1992.

Table 1
Cycle 8 Operating Parameters

Parameter	Proposed Cycle 8 Conditions 10% SGTP	Upper Bound 1.8 °F T_{hot} Increase 10% SGTP	Lower Bound 15 °F Thot Reduction 10% SGTP
NSSS Power, MWt	3579	3579	3579
Reactor Power, MWt	3565	3565	3565
Thermal Design Flow			
Per loop, gpm	93600	93600	93600
Total flow, gpm	374400	374400	374400
Reactor Flow, Total, (Mlbm/hr)	139.79	139.4	142.9
Reactor Coolant Press, psia	2250	2250	2250
Core Bypass, %	8.4	8.4	8.4
Fuel Design	17x17 V5H w/IFMs	17x17 V5H w/IFMs	17x17 V5H w/IFMs
Reactor Coolant Temperature, °F			
Core Outlet	623.3	625.0	608.5
Vessel Outlet	618.2	620.0	603.2
Core Average	591.1	593.0	575.1
Vessel Average	586.5	588.4	570.7
Vessel/Core Inlet	554.8	556.8	538.2
Steam Generator Outlet	554.5	556.6	538.0
Vessel DT	63.4	63.2	65.0
Steam Generator			
Steam Temperature, F	536.3	538.4	519.4
Steam Pressure, psia	934	950	807
Steam Flow, total, Mlbm/hr	15.91	15.92	15.83
Feedwater Temp, F	446	446	446
Zero Load Temp, F	557	557	557
SG Tube Plugging, %	10	10	10

Table 2
Summary of Technical Specifications Changes

Technical Specification	Page	Description Of Change	Reason For Change
Table 2.2-1 Item 7	2-4	Overtemperature ΔT Z increased from 4.86 to 5.39	OTΔT setpoint optimized with assumed full power ΔT and revised f1(ΔI).
Table 2.2-1 Note 1	2-8	Overtemperature ΔT f1(ΔI) positive slope increased from 1.56 % ΔT / % Δ I to 1.84 % ΔT / % ΔI	f1(ΔI) positive slope increased based on the new maneuvering analysis.
Table 2.2-1 Note 1	2-8	Overtemperature ΔT f1(ΔI) negative breakpoint reduced from -25 % ΔI to -23 % ΔI	f1(ΔI) negative breakpoint reduced based on the new maneuvering analysis.
Table 2.2-1 Note 1	2-8	Overtemperature ΔT f1(ΔI) negative slope increased from 1.8 % ΔT / % ΔI to 2.27 % ΔT / % ΔI	f1(ΔI) negative slope increased based on the new maneuvering analysis.
Table 2.2-1 Note 2	2-8	Overtemperature ΔT Allowable Value reduced from 1.8 % ΔT Span to 1.3 % ΔT Span	OTΔT setpoint optimized with assumed full power ΔT and revised f1(ΔI).

ATTACHMENT II

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

No Significant Hazards Consideration Determination

This license amendment request proposes to revise Technical Specification 2.2, Table 2.2-1, Reactor Trip System Instrumentation Setpoints. This revision changes values for axial flux difference limits to account for revised peaking limits resulting from the Cycle 8 core design analysis.

Standard I - 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. Operation at 3565 MWt does not affect any of the mechanisms postulated in the USAR to cause LOCA or non-LOCA design basis events. Analyses, evaluations and minimum 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.

Standard II - Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated

There is no new type of accident or malfunction being created. The proposed change provides revised operating limits necessary to support Cycle 8, and does not change the method and manner of plant operation. The safety design bases in the USAR have not been altered. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Standard III - Involve a Significant Reduction in the Margin of Safety

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. The analyses and evaluations discussed in the safety evaluation (Attachment I) demonstrate that all applicable safety analysis acceptance criteria continue to be met for the proposed operating conditions. Items not specifically cited in this safety evaluation have been reviewed and have been found to be bounded by the evaluations performed for Reference 1. Therefore, it is concluded that the margin of safety, as described in the bases to any technical specification, is not reduced.

Based on the above discussions, it has been determined that the requested technical specification revision does not involve a significant increase in the probability or consequences of an accident or other adverse condition over previous evaluations; or create the possibility of a new or different kind of accident or condition over previous evaluation; or involve a significant reduction in a margin of safety. Therefore, the requested license amendment does not involve a significant hazards consideration.

ATTACHMENT III

ENVIRONMENTAL IMPACT DETERMINATION

Environmental Impact Determination

This amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) as specified below:

(i) the amendment involves no significant hazards consideration

As demonstrated in Attachment II, the proposed change does not involve any significant hazards consideration.

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite

The proposed change does not involve a change to the facility or operating procedures which would cause an increase in the amounts of effluents or create new types of effluents. Reactor core and coolant activities are not affected by the revised axial flux difference (AFD) limits since the activities are based on cycle length and core burnup. Primary and secondary parameters that could affect radiological releases are bounded by the parameters for the 1.8 °F T_{hot} increase and 15 °F T_{hot} reduction as outlined in Attachment I, Table 1. Therefore, radiological consequences for operation with the revised AFD limits are bounded by the analyses performed previously.

(iii) there is no significant increase in individual or cumulative occupation radiation exposure

Changing the AFD setpoints would not adversely affect the operation of the reactor, and would not affect any system that would affect occupational radiation exposure. The proposed change does not create additional exposure to personnel nor affect levels of radiation present. The proposed change will not result in any increase in individual or cumulative occupational radiation exposure.

Based on the above, it is concluded that there will be no impact on the environment resulting from the proposed change, and that the proposed change meets the criteria specified in 10 CFR 51.22 for a categorical exclusion from the requirements of 10 CFR 51.21 relative to requiring a specific environmental assessment by the Commission.

ATTACHMENT IV

PROPOSED TECHNICAL SPECIFICATION CHANGE