March 1, 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. 96 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. M94112)

Dear Mr. Carns:

The Commission has issued the enclosed Amendment No. 96 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 November 22, 1995.

The amendment replaces the Technical Specification (TS) requirements associated with the boron dilution mitigation system (BDMS) with alarms, indicators, procedures and controls to allow proper resolution of potential boron dilution events.

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. 96 to NPF-42
2. Safety Evaluation

cc w/encls: See next page

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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. 96
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 November 22, 1995, 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. 96, 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 prior to startup from the eighth refueling outage.

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: March 1, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 96

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
3/4 3-2	3/4 3-2
3/4 3-5	3/4 3-5
3/4 3-6	3/4 3-6
3/4 3-9	3/4 3-9
3/4 3-12	3/4 3-12

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1,2	1
	2	1	2	3*,4*,5*	10
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1,2	2#
b. Low Setpoint	4	2	3	1###,2	2#
3. Power Range, Neutron Flux, High Positive Rate	4	2	3	1,2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1,2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###,2	3
6. Source Range, Neutron Foux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3,4,5	5
7. Overtemperature ΔT Four Loop Operation	4	2	3	1,2	6#
8. Overpower ΔT Four Loop Operation	4	2	3	1,2	6#
9. Pressurizer Pressure-Low	4	2	3	1	6#
10. Pressurizer Pressure-High	4	2	3	1,2	6#

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*Only if the Reactor Trip System breakers happen to be in the closed position and the Control Rod Drive System is capable of rod withdrawal.
#The provisions of Specification 3.0.4 are not applicable.
##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.
###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.
(1) The applicable MODES for these channels noted in Table 3.3-3 are more restrictive and therefore applicable.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE CHANNELS one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours;

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1; and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint; or

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 5 -
a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip Breakers, and suspend all operations involving positive reactivity changes within the next hour.

b. With no channels OPERABLE, open the Reactor Trip Breakers, suspend all operations involving positive reactivity changes and verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 within 1 hour and every 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours; and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is operable.

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 9 - With the number of OPERABLE Reactor Trip Breakers one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one breaker may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1., provided the other breaker is OPERABLE.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour.

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(11)	N.A.	1,2,3*,4*,5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2,4) M(3,4) Q(4,6) R(4,5)	Q	N.A.	N.A.	1,2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1###,2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q	N.A.	N.A.	1,2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	Q	N.A.	N.A.	1,2
5. Intermediate Range, Neutron Flux	S	R(4,5)	S/U(1)	N.A.	N.A.	1###,2
6. Source Range, Neutron Flux	S	R(4,5)	S/U(1),Q(9)	N.A.	N.A.	2##,3,4,5
7. Overtemperature ΔT	S	R	Q	N.A.	N.A.	1,2
8. Overpower ΔT	S	R	Q	N.A.	N.A.	1,2
9. Pressurizer Pressure-Low	S	R	Q	N.A.	N.A.	1
10. Pressurizer Pressure-High	S	R	Q	N.A.	N.A.	1,2
11. Pressurizer Water Level-High	S	R	Q	N.A.	N.A.	1
12. Reactor Coolant Flow - Low	S	R	Q	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level-Low-Low	S	R	Q(15)	N.A.	N.A.	1, 2
14. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	Q	N.A.	1
15. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	Q	N.A.	1
16. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
18. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	R	N.A.	N.A.	2##
b. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-9	N.A.	R(4)	R	N.A.	N.A.	1

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
18. Reactor Trip System Interlocks (Continued)						
d. Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
e. Turbine Impulse Chamber Pressure, P-13	N.A.	R	R	N.A.	N.A.	1
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 16)	N.A.	1, 2, 3*, 4*, 5*
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*
21. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M(17),R(18)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*Only if the Reactor Trip System breakers happen to be closed and the control rod drive system is capable of rod withdrawal.
##Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.
###Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) For Source Range detectors, integral bias curves are obtained, evaluated, and compared to manufacturer's data. For Intermediate Range and Power Range channels, detector plateau curves shall be obtained, evaluated, and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into Mode 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(9) Quarterly surveillance in MODES 3*, 4* and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

(10) Setpoint verification is not required.

**(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(12) Deleted.

**Complete verification of OPERABILITY of the manual reactor trip switch circuitry shall be performed prior to startup from the first shutdown to Mode 3 occurring after August 14, 1992.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 96 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 November 22, 1995, Wolf Creek Nuclear Operating Corporation (WCNOC) (the licensee) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-42) for the Wolf Creek Generating Station (WCGS). The proposed changes would replace the Technical Specification (TS) requirements associated with the boron dilution mitigation system (BDMS) with alarms, indicators, procedures and controls to allow proper resolution of potential boron dilution events. The BDMS was developed to detect and mitigate a boron dilution event in Modes 3, 4 and 5 prior to a complete loss of shutdown margin (criticality). The current analysis of the inadvertent boron dilution event for Modes 3, 4 and 5, described in the WCGS Updated Safety Analysis Report (USAR) Section 15.4.6, was based on the operation of the BDMS. However, as noted in NRC Information Notice 93-32, various concerns have been raised about the inverse count rate ratio and the flux multiplication setpoint used by the BDMS in the boron dilution analyses which may render the system unsatisfactory from an accident analysis standpoint.

The proposed changes would revise TS 3/4.3.1, "Reactor Trip System Instrumentation," by removing reference to the source range boron dilution flux doubling instrumentation and its associated action statement, surveillance and implementation footnotes. The specific changes are listed below:

(1) Table 3.3-1, Reactor Trip System Instrumentation - Delete reference to table notation "**" in items 6a and 6b, concerning the operability of the boron dilution flux doubling instrumentation.

(2) Table 3.3-1, Table Notations - Delete table notation "**" concerning blocking flux doubling signals during startup.

(3) Table 3.3-1, Action Statements - Delete references to verification of position of valves BG-V178 and BG-V601 in Action Statements 5a and 5b.

(4) Table 4.3-1, Reactor Trip System Instrumentation Surveillance Requirements - Delete reference to note 12 in item 6, Source Range, Neutron Flux, Channel Calibration column.

(5) Table 4.3-1, Table Notations - Delete the portion of Table Notation (9) concerning the quarterly verification of boron dilution alarm setpoint. Also delete Table Notation (12) concerning the 18 month surveillance requirement of the boron flux doubling circuitry.

2.0 EVALUATION

As stated above, the BDMS alone could not be shown to prevent WCGS from returning to critical following a boron dilution event. Therefore, the following changes are proposed to the plant hardware and operations:

(1) Two additional high level alarms on the volume control tank (VCT) will be installed. The alarm setpoints will be lower than the current high-high VCT level alarm. This change is acceptable since it improves instrumentation reliability and provides redundancy of alarms. Credit for these alarms has been taken in the revised boron dilution analyses for Modes 3, 4 and 5, as evaluated below.

(2) The normal operating mode of the letdown divert valve will be revised from "AUTO" to "VCT". This change is acceptable since it should enhance operator awareness during planned dilution operations and, thus, reduce the potential for inadvertent dilution during routine plant operations.

(3) An alarm will be installed on the letdown divert valve to annunciate when the valve is not in the "VCT" position. This is acceptable as it will serve to heighten operator awareness of the potential for a dilution event during and following planned plant evolutions.

(4) Plant operating procedures will be revised to require the operation of at least one reactor coolant pump (RCP) in Modes 3, 4 and 5 or have a valve in the flow paths of potential boron dilution sources closed or under administrative control. Operation with at least one RCP in Modes 3, 4 and 5 is currently allowed by TS. The revised boron dilution analyses for Modes 3, 4 and 5 credit the mixing volume associated with the operation of an RCP. Isolating the dilution sources reduces the possibility of an inadvertent dilution. These changes are, therefore, acceptable.

In addition, the BDMS flux multiplication alarm, although not used as the primary signal, will be retained as a plant design feature to provide the plant operators a diverse method for identifying a potential dilution event.

WCNOC provided a detailed safety analysis of the postulated inadvertent boron dilution event in Modes 3, 4, and 5 using the hardware and operational changes proposed above. The event is assumed to be initiated through a malfunction in the reactor makeup control system (RMCS) or by operator error. Since all sources of unborated water are locked out during refueling, the boron dilution event is not analyzed from Mode 6 initiation. The event has been reanalyzed to demonstrate compliance with the Standard Review Plan (SRP), NUREG-0800, Revision 2. SRP 15.4.6 states that, if operator action is required to terminate a dilution event, the following minimum time intervals must be available between the time when an alarm announces an unplanned moderator dilution and the time of total loss of shutdown margin (criticality):

(1) During refueling: 30 minutes

(2) During startup, cold shutdown, hot standby, power operation: 15 minutes

WCNOC has modified these definitions to require the above operator action time intervals to extend from the time when an alarm announces an unplanned moderator dilution to the time when the corrective actions must be initiated in order to prevent a complete loss of shutdown margin. The event is not considered to be terminated (i.e., criticality prevented) until the borated water from the refueling water storage tank (RWST) purges the dilute water in the charging lines and enters the reactor coolant system (RCS). Thus, the reanalysis must show that the operators have at least 15 minutes from the alarm in which to perform the actions required to terminate an inadvertent boron dilution event before the shutdown margin is completely eroded. This is referred to as the net operator response time, or t_{op}. This is consistent with, and, in fact, more conservative than the SRP 15.4.6 requirements and is, therefore, acceptable. System delays for valve manipulations and purging diluted pipes with boric acid are considered and further reduce the net operator response time, as shown below.

The net operator response time (t_{op}) is calculated by the following relationship:

$$t_{op} = t_{crit} - t_{fill} - t_{swap} - t_{purg}$$

where

t_{crit} is the time required to deborate the volume which becomes diluted during the transient from the shutdown margin concentration to the critical concentration.

t_{fill} is the time required to increase the net volume of the RCS and CVCS by the amount required to actuate the high VCT water level alarm.

t_{swap} is the sum of the opening time of the RWST isolation valves and the closing time of the VCT isolation valves.

t_{purg} is the period of time from the end of the valve swapover to the time that borated water from the RWST enters the RCS.

This revised methodology was used to analyze the inadvertent boron dilution events for WCGS, Unit 1 Cycle 9. In all cases, it was verified that the reactor operators have greater than 15 minutes after an alarm in which to perform the actions required to terminate the event before the inadvertent criticality could occur, thereby meeting the SRP requirements.

There may be other dilution events which are not obviously bounded by the analysis described above. For example, for small dilution flow rates, the time required to fill the VCT to the high VCT water level setpoint may be greater than the time required to dilute the RCS to the critical condition. However, the time intervals involved in this case are relatively long and other alarms such as the concentrated boric acid flow or total makeup flow deviation alarms would alert the operator to a potential inadvertent boron dilution. The alarms generated by the nuclear instrumentation system, or the available trend recorders would also be available to alert operators for these slow dilution cases. For these relatively slow transients, WCNOC has proposed an alternate event acceptance criterion. This criterion requires that the time between the start of the event and the complete loss of shutdown margin (inadvertent criticality) be at least 30 minutes. The staff has determined that this provides reasonable assurance that the above-mentioned other indications would alert the reactor operators to an inadvertent boron dilution event and allow initiation of timely corrective actions.

SRP 15.4.6 also requires redundancy of alarms that would alert the operator to an unplanned dilution. The licensee has committed to the installation of redundant alarms for high VCT water level with a lower setpoint (70 percent span) than the current high-high VCT water level alarm (97 percent span) and has taken credit for this alarm in the reanalysis of the dilution events. In addition, the licensee has committed to installation of an alarm on the letdown divert valve (LCV-112A) which will annunciate when the valve is not in the "VCT", or normal, mode. These alarms, in addition to the available boric acid flow indication and deviation alarm, the total makeup blended flow indication and deviation alarm, the VCT high pressure alarm, and the alarms provided by the nuclear instrumentation system, provide an acceptable level of redundancy and diversity to alert operators to an ongoing inadvertent boron dilution event.

3.0 CONCLUSION

The staff has reviewed the proposed TS changes for WCGS which remove reference to the source range boron dilution flux doubling instrumentation. In addition, the staff has evaluated the proposed changes to the plant hardware and operations as well as the revised boron dilution event methodology. Based on the above safety evaluation, the staff finds these proposed changes acceptable.

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 <u>ENVIRONMENTAL CONSIDERATION</u>

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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 (61 FR 3503). 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 <u>CONCLUSION</u>

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

Principal Contributor: L. Kopp

Date: March 1, 1996