December 5, 198[illegible]

See correction letter of 12/13/88

Docket No. 50-482

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TAlexion	

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. 23 TO FACILITY OPERATING LICENSE NO. NPF-42 (TAC NO. 68648)

The Commission has issued the enclosed Amendment No. 23 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 June 24, 1988.

The amendment revises the Technical Specifications and corresponding Bases necessary for Cycle 4 operation. These changes include increased boron concentrations in the refueling water storage tank and accumulators, a revised part power multiplier and changes resulting from revised reactor coolant temperature measurement uncertainties.

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,

/s/

Douglas V. Pickett, Project Manager
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 23 to License No. NPF-42
2. Safety Evaluation

cc w/enclosures:
See next page

DOCUMENT NAME: WC AMEND TAC 68648

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December 5, 198

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

December 5, 1988

Docket No. 50-482

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

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Douglas V. Pickett

Douglas V. Pickett, Project Manager
Project Directorate - IV
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
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2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Bart D. Withers
Wolf Creek Nuclear Operating Corporation

Wolf Creek Generating Station
Unit No. 1

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

KANSAS GAS & ELECTRIC COMPANY

KANSAS CITY POWER AND LIGHT COMPANY

KANSAS ELECTRIC POWER COOPERATIVE, INC.

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 23
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, dated June 24, 1988, 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. 23 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. The Operating 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

José A. Calvo

Jose A. Calvo, Director
Project Directorate - IV
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 5, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 23

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 PAGES	INSERT PAGES
2-4	2-4
2-5	2-5
2-8	2-8
2-10	2-10
B 2-1	B 2-1
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
3/4 5-1	3/4 5-1
3/4 5-10	3/4 5-10
3/4 10-4	3/4 10-4
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3
B 3/4 2-5	B 3/4 2-5
B 3/4 5-2	B 3/4 5-2

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	7.2	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	≥1875 psig	≥1866 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 = 95,700 gpm

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	3.0	2.06	0.6	>89.9% of loop design flow**	>88.9% of loop design flow**
13. Steam Generator Water Level Low-Low	23.5	21.18	2.51	≥23.5% of narrow range instrument span	≥22.3% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	7.5	1.3	0	≥10578 Volts A.C.	≥10355 Volts A.C.
15. Underfrequency - Reactor Coolant Pumps	3.3	0	0	≥57.2 Hz	≥57.1 Hz
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥590.00 psig	≥534.20 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	588.5°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.000671;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) for $q_t - q_b$ between -27% and + 7%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -27%, the ΔT Trip Setpoint shall be automatically reduced by 1.57% of its value at RATED THERMAL POWER; and

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT Trip Setpoint shall be automatically reduced by 0.85% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.8% of ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	$0.00128/^{\circ}F$ for $T > T''$ and $K_6 = 0$ for $T \leq T''$;
T	=	Average temperature, °F;
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 588.5°F);
S	=	Laplace transform operator, s^{-1}; and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.1% of ΔT span.

2.1 SAFETY LIMITS

BASES

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 the W-3 correlation (R-GRID). The W-3 DNB correlation (R-GRID) has 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 minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

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 1.30, 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.

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.

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figure 3.2-3 for four loop operation.

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\,[1.0 + 0.3\,(1.0 - P)]}$,

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes measurement uncertainties of 2.5% for flow and 4% for incore measurement of $F^N_{\Delta H}$.

APPLICABILITY: MODE 1.*

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours; and

*See Special Test Exception Specification 3.10.4.

SURVEILLANCE REQUIREMENTS (Continued)

2) When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.9;

f. The F_{xy} limits of Specification 4.2.2.2e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100%, inclusive,

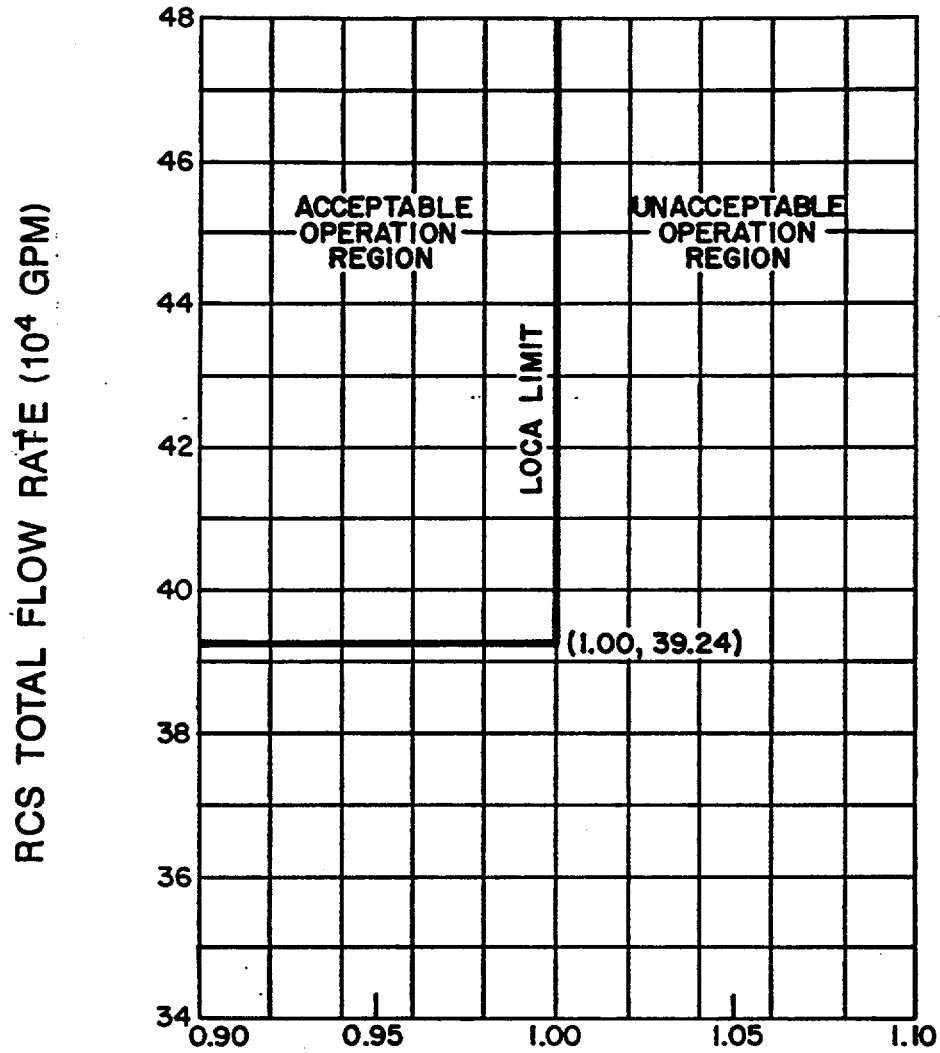
3) Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive, and

4) Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

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



$$R = F_{\Delta H}^{N}/1.49[1.0 + 0.3(1.0-P)]$$

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.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System accumulator shall be OPERABLE with:

a. The isolation valve open and power removed,

b. A contained borated water volume of between 6122 and 6594 gallons,

c. A boron concentration of between 2300 and 2500 ppm, and

d. A nitrogen cover-pressure of between 585 and 665 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce RCS pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce RCS pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 70 gallons by verifying the boron concentration of the accumulator solution, and

c. At least once per 31 days when the RCS pressure is above 1000 psig by verifying that the circuit breaker supplying power to the isolation valve operator is open.

4.5.1.2 Each accumulator water level and pressure channel shall be demonstrated OPERABLE at least once per 18 months by the performance of a CHANNEL CALIBRATION.

3/4.5.4 ECCS SUBSYSTEMS - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.5.4 All Safety Injection pumps shall be inoperable.

APPLICABILITY: MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to an inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 All Safety Injection pumps shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position at least once per 31 days.

*An inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 394,000 gallons,

b. A boron concentration of between 2400 and 2500 ppm of boron,

c. A minimum solution temperature of 37°F, and

d. A maximum solution temperature of 100°F.

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

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 37°F or greater than 100°F.

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6, may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at least once per 30 minutes during PHYSICS TESTS.

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of the following requirements may be suspended:

a. Specification 3.2.3 and 3.4.1.1 - During the performance of startup and PHYSICS TESTS in MODE 1 or 2 provided:

   1) The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

   2) The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

b. Specification 3.4.1.2 - During the performance of hot rod drop time measurements in MODE 3 provided at least three reactor coolant loops as listed in Specification 3.4.1.2 are OPERABLE.

APPLICABILITY: During operation below the P-7 Interlock Setpoint or performance of hot rod drop time measurements.

ACTION:

a. With the THERMAL POWER greater than the P-7 Interlock Setpoint during the performance of startup and PHYSICS TESTS, immediately open the Reactor trip breakers.

b. With less than the above required reactor coolant loops OPERABLE during performance of hot rod drop time measurements, immediately place two reactor coolant loops in operation.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

4.10.4.3 At least the above required reactor coolant loops shall be determined OPERABLE within 4 hours prior to initiation of the hot rod drop time measurements and at least once per 4 hours during the hot rod drop time measurements by verifying correct breaker alignments and indicated power availability.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.1×10^{-4} $\Delta k/k/^{\circ}F$. The MTC value of -3.2×10^{-4} $\Delta k/k/^{\circ}F$ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -4.1×10^{-4} $\Delta k/k/^{\circ}F$.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boration Systems ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) centrifugal charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature equal to or greater than 350°F a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 17,658 gallons of 7000 ppm borated water from the boric acid storage tanks or 83,754 gallons of 2400 ppm borated water from the RWST. With the RCS average temperature less than 350°F, only one boron injection flow path is required.

BORATION SYSTEMS (Continued)

With the RCS temperature below 200°F, one Boration System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable in MODES 4, 5, and 6 provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or an RHR suction relief valve.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2968 gallons of 7000 ppm borated water from the boric acid storage tanks or 14,071 gallons of 2400 ppm borated water from the RWST.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. In the case of the boric acid tanks, all of the contained volume is considered usuable. The required usable volume may be contained in either or both of the boric acid tanks.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boration System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120, and 228 steps withdrawn for the Control Banks and 18, 210 and 228 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

MOVABLE CONTROL ASSEMBLIES (Continued)

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

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 minimum DNBR of 1.30 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.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2, 3/4.5.3, and 3/4.5.4 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and Safety Injection pumps except the required OPERABLE charging pump to be inoperable in MODES 4 and 5 and in MODE 6 with the reactor vessel head on provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or RHR suction relief valve.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The Surveillance Requirements for leakage testing of ECCS check valves ensures that a failure of one valve will not cause an intersystem LOCA. The Surveillance Requirements to vent the ECCS pump casings and accessible, i.e., can be reached without personnel hazard or high radiation dose, discharge piping ensures against inoperable pumps caused by gas binding or water hammer in ECCS piping.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes assuming all the control rods are out of the core. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 23 TO FACILITY OPERATING LICENSE NO. NPF-42

KANSAS GAS & ELECTRIC COMPANY

KANSAS CITY POWER AND LIGHT COMPANY

KANSAS ELECTRIC POWER COOPERATIVE, INC.

WOLF CREEK GENERATING STATION

DOCKET NO. 50-482

1.0 INTRODUCTION

Wolf Creek Unit 1 has completed its third cycle of operation and is commencing its refueling for Cycle 4 operation. This reload core will use 72 new Westinghouse 17x17 fuel assemblies and 60 Annular Burnable Absorber assemblies. The 72 new fuel assemblies will comprise Regions 6A and 6B. Region 6A will be composed of 36 fuel assemblies enriched to 3.73 weight percent U-235 and Region 6B will be composed of 36 fuel assemblies enriched to 4.10 weight percent U-235.

As a result of the Cycle 4 reload core design, the following Technical Specification changes are required.

1. Increase boron concentrations in the refueling water storage tank (RWST) and accumulators.

2. Revise the part power multiplier, reactor coolant temperature measurement uncertainties associated with the reactor trip setpoints for overtemperature delta-T and overpower delta-T, and the reactor coolant flow measurement uncertainty associated with the reactor trip setpoints. Also editorial changes to correct inconsistencies between Technical Specifications and analyses assumptions.

The licensee, in a letter dated June 24, 1988, requested the above changes. The licensee's evaluation was provided in its submittal to justify the proposed Technical Specification changes.

2.0 EVALUATION

Increased Boron Concentration in RWST and Accumulators

The licensee, in Attachment A to its letter dated June 24, 1988, provided the results of its safety evaluation of LOCA and non-LOCA events to support the new Technical Specification values for boron concentration in the RWST and accumulators. It has been concluded that the changes in the RWST and

accumulator boron concentrations do not affect the results of the short-term large or small break LOCA analysis because credit was not taken for the presence of boron in the ECCS water. Regarding long-term LOCA analysis, the Westinghouse Evaluation Model states that the reactor will remain shutdown by borated ECCS water residing in the RCS/Sump. Since credit for the control rods is not taken for a large break LOCA, the borated ECCS water provided by the accumulators and the RWST must have a boron concentration that, when mixed with other water sources, will result in the reactor core remaining subcritical assuming all control rods out. The licensee stated that the effect on the post-LOCA RCS/Sump boron concentration as a result of changing the minimum Technical Specification boron concentration from 2000 to 2400 ppm for the RWST, and from 1900 to 2300 ppm for the accumulators is an increase of about 270 ppm in the RCS/Sump boron concentration. The licensee also proposes changes in boron concentration range in the RWST from 2000-2100 ppm to 2400-2500 ppm and in the accumulators from 1900-2100 ppm to 2300-2500 ppm. It has been confirmed by the licensee's calculation that this proposed increase will provide enough margin to keep the core subcritical during the post-LOCA long-term core cooling. The licensee also confirmed by its evaluation that this proposed increase of boron concentration for the RWST and accumulators is sufficient to maintain shutdown margin during all cases of reactor cooldown when ECCS water is present in the RCS.

Considering the amount of increase in boron concentration for the RWST and accumulators, the licensee has performed an analysis to determine the time following a LOCA that switchover to hot leg recirculation should be initiated to prevent boron precipitation in the reactor vessel. This time has been determined to be 12 hours. The staff finds that this hot leg switchover time should be incorporated in the plant operating procedures. In their letter of June 24, 1988, the licensee has committed to revise the emergency operating procedures to initiate hot leg switchover in 12 hours. This is acceptable.

The licensee also evaluated the non-LOCA events in which boron from the RWST or accumulators is taken credit for, or assumed to be present. This evaluation was performed considering the new fuel in the core as well as the new Technical Specification values. The results of its evaluation indicate that either the transients are not affected by the changes or the results of the transient are within acceptance criteria of each event.

The staff has reviewed the licensee's evaluation and find that they are reasonable and conservative. There are minor editorial changes in the Bases section of the Technical Specifications dealing with boron concentration in ECCS water. They are consistent with the supporting analysis and therefore, acceptable.

<u>Changes of $F^N_{\Delta H}$: Part Power Multiplier, Reactor Coolant Temperature and Flow Measurement Uncertainties and Special Test Exception</u>

The licensee, in Attachment B to its letter dated June 24, 1988, requested a change of the part power multiplier from 0.2 to 0.3 for the calculation of radial peaking factor $F^N_{\Delta H}$. The change is requested for Wolf Creek Unit 1 to

allow optimization of the core loading pattern by minimizing restrictions on $F^N_{\Delta H}$ at low power. This change will also minimize the probability of making rod insertion limit changes in future reload cycles to satisfy peaking factor criteria at low power.

The licensee has reevaluated the Reactor Core Safety Limits due to the increased $F^N_{\Delta H}$ limit to ensure adequate core protection. It has been concluded by the licensee that both LOCA and non-LOCA analyses are not impacted by the proposed $F^N_{\Delta H}$ part power multiplier change. Also, the same increase in the part power multiplier has been approved for a number of operating Westinghouse designed plants. The staff concludes the proposed change of part power multiplier from 0.2 to 0.3 is acceptable for Wolf Creek Unit 1.

The licensee in its submittal also proposes changes of the RCS RTD measurement uncertainty and RCS flow measurement uncertainty in Technical Specifications which affect reactor trip setpoints. These changes are based on improvements in calculation techniques, the use of plant specific inputs, and the change from the revised part power multiplier. The basic setpoint methodology previously established by the NSSS vendor remains unchanged. The staff has reviewed the justification for these changes and find that the proposed changes are reasonable and acceptable.

The licensee proposes to suspend the requirements of Technical Specification 3.2.3 during the performance of startup and low power physics tests. This action requires both the addition of Specification 3.2.3 to Technical Specification 3.10.4, Special Test Exception; and insertion of a footnote in Specification 3.2.3 invoking the special test exception. Specification 3.10.4 allows for suspension of selected requirements when performing startup or physics tests. These tests are often performed at no-flow or low-flow conditions. Specification 3.10.4 requires that the reactor trip setpoints be set to less than or equal to 25% rated thermal power during performance of these tests. Specification 3.2.3 describes margins that must be maintained under four loop operation and is not applicable to startup or low power physics tests. In addition, one of the ACTION statements of Specification 3.2.3 requires lowering the reactor to less than 50% of rated thermal power if the margins are not met. Clearly, the application of Specification 3.2.3 is inconsistent with startup or low power physics tests and an equivalent level of protection is provided by the requirements of Specification 3.10.4. Therefore, the staff concludes that this proposed change is consistent with the intent of the special tests exception provision and it is acceptable.

Based on the information presented in the licensee's letter dated June 24, 1988, the staff has concluded that the licensee's proposed changes to Technical Specifications 3.2.3, 3.1.2.5, 3.1.2.6, 3.5.1, 3.5.5, 3.10.4, Figure 3.2-3, Table 2.2-1 and corresponding Bases are therefore acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

The amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20.

The 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 exposures. 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. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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.

CONCLUSION

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: December 5, 1988

Principal Contributor: C. Liang