Docket No.: 50-482 April 22, 1986

Mr. Glenn L. Koester
Vice President - Nuclear
Kansas Gas and Electric Company
201 North Market Street
Post Office Box 208
Wichita, Kansas 67201

Dear Mr. Koester:

SUBJECT: Wolf Creek Generating Station - Amendment No. 1 to Facility Operating License NPF-42

The Commission has issued the enclosed Amendment No. 1 to Facility Operating License NPF-42 for the Wolf Creek Generating Station. The amendment is in response to your application dated January 20, 1986.

This amendment modifies the Technical Specifications to implement the Relaxed Axial Offset Control mode of operation after fuel burnup of 8000 megawatt days/metric ton of uranium is achieved. The amendment is effective as of its date of issuance.

A copy of our related Safety Evaluation is enclosed. Notice of issuance will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

Paul W. O'Connor

Paul W. O'Connor, Project Manager
PWR Project Directorate #4
Division of PWR Licensing-A

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

cc w/enclosures:
See next page

* SEE PREVIOUS CONCURRENCE

PWR#4/DPWR-A	PWR#4/DPWR-A	PWR#4/DPWR-A
*PO'Connor/mac	*MDuncan	*BJYoungblood
04/18/86	04/18/86	04/18/86
PWOC		

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P PDR

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

April 22, 1986

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See next page

Mr. Glenn L. Koester
Kansas Gas and Electric Company

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.1
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 Kansas Gas and Electric Company acting for itself and Kansas City Power and Light Company and Kansas Electric Power Cooperative, Inc., (the licensee) dated January 20, 1986, 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, 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 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.

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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:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 1 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. KG&E shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Paul W. O'Connor

for B. J. Youngblood, Director
PWR Project Directorate #4
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 22, 1986

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:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 1 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. KG&E shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

B. J. Youngblood, Director
PWR Project Directorate #4
Division of PWR Licensing-A

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 22, 1986

PWR#4/DPWR-A	PWR#4/DPWR-A	OELD	PWR#4/DPWR-A
MDuncan/mac	PO'Connor	M. Karman	BJYoungblood
03/18 /86	03/21 /86	03/25/86	03/21 /86

ISSUANCE OF AMENDMENT NO. 1 TO WOLF CREEK GENERATING STATION Dated April 22, 1986

DISTRIBUTION
Docket File
NRC PDR
Local PDR
PWR#4 Rdg
BJYoungblood
MDuncan
PO'Connor
OELD
ACRS (10)
EJordan
BGrimes
JPartlow
TBarnhart (4)
LJHarmon
WJones
OPA
LFMB

ATTACHMENT TO LICENSE AMENDMENT NO. 1

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 the captioned amendment number and contain marginal lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

AMENDED PAGE	OVERLEAF PAGE
3/4 2-1	
3/4 2-2	
3/4 2-3	3/4 2-4
2-8	2-7
B 3/4 2-1	
B 3/4 2-2	
B 3/4 2-3	B 3/4 2-4

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the allowed operational space defined by Figure 3.2-1.

APPLICABILITY: MODE 1 above 50 PERCENT RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the Figure 3.2-1 limits,

   1. Either restore the indicated AFD to within the Figure 3.2-1 limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the Figure 3.2-1 limits.

*See Special Test Exception 3.10.2.

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

   1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

   2) At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least two OPERABLE excore channels are indicating the AFD to be outside the limits.

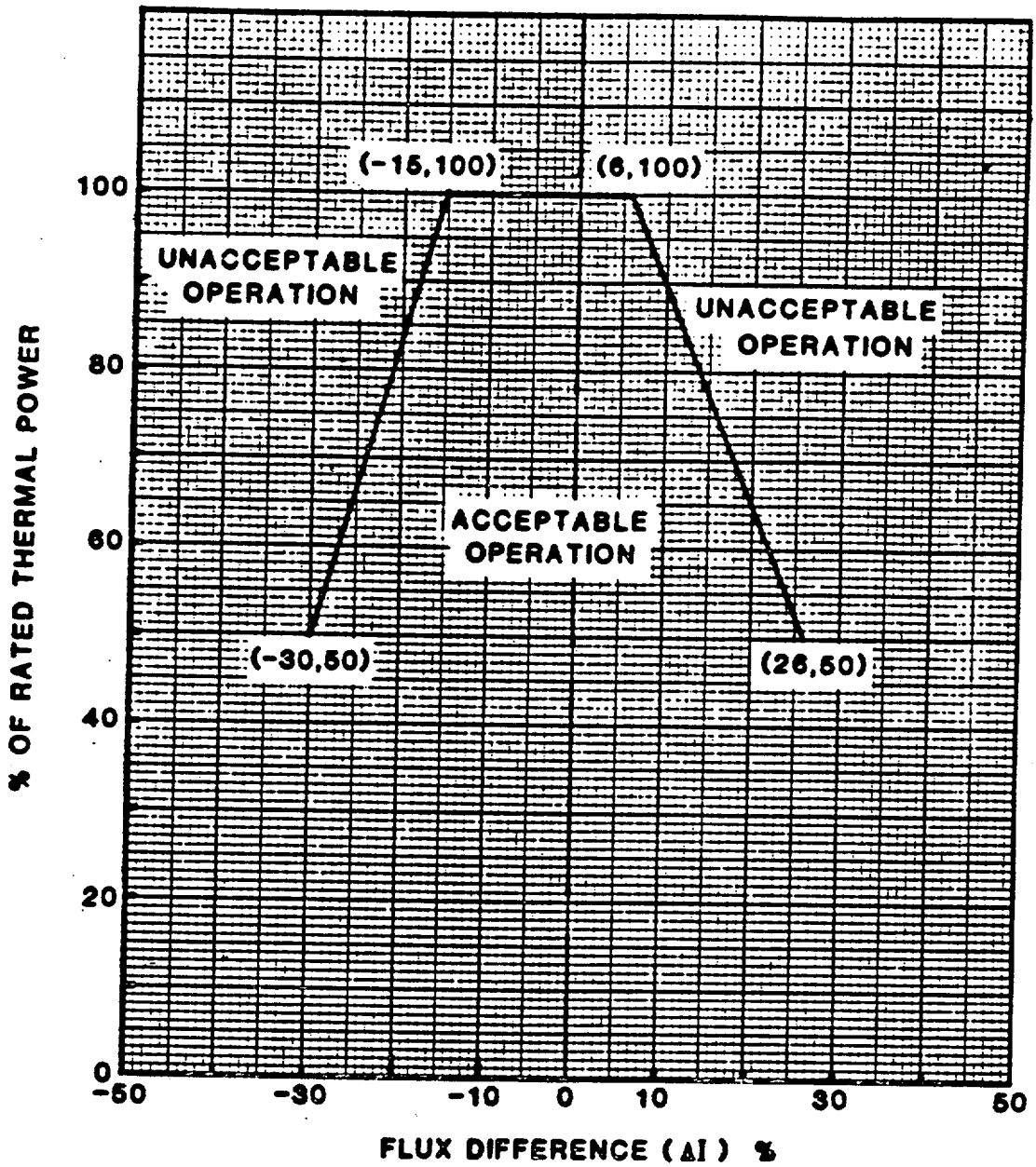

FIGURE 3.2-1
WOLF CREEK UNIT 1
AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.32}{P}\right] [K(Z)] \text{ for } P > 0.5, \text{ and}$$

$$F_Q(Z) \leq [4.64] [K(Z)] \text{ for } P \leq 0.5.$$

Where:

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}, \text{ and}$$

K(Z) = the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 = 8$ s, $\tau_2 = 3$ s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.10;
K_2	=	0.0137/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 = 28$ s, $\tau_5 = 4$ s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

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 1.05% 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 3.3% of ΔT span.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short-term transients, and (b) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of F_Q limit times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the AFD limits and the THERMAL POWER is greater than 50% of RATED THERMAL POWER.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded, and (2) in the event of a

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position,

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6,

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

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

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

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

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

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR, completely offset any rod bow penalties. This margin includes the following:

1) Design limit DNBR of 1.30 vs. 1.28,
2) Grid spacing (K_s) of 0.046 vs. 0.059,
3) Thermal Diffusion Coefficent of 0.038 vs. 0.059,
4) DNBR Multiplier of 0.86 vs. 0.88, and
5) Pitch Reduction.

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 1 TO 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

INTRODUCTION

By letter dated January 20, 1986, the Kansas Gas & Electric Company (KG&E) proposed a license amendment to revise the Technical Specifications for Wolf Creek Generating Station, Unit 1, after 8000 MWD/MTU burnup in Cycle 1. These changes would implement a Westinghouse developed Power Distribution Control methodology called Relaxed Axial Offset Control (RAOC).

EVALUATION

Westinghouse reactors have for a number of years operated under a power distribution control system called Constant Axial Offset Control (CAOC), which ensures peaking factors will remain below values assumed as input for accident analyses during normal operation of the power plant. Basically CAOC achieves its result by requiring plant operation within a ±5% flux difference (Δ I) around a measured target value. By controlling the axial power distribution, the possible skewing of the axial xenon distribution is limited, thus minimizing xenon oscillations and their effects on the power distribution.

Plants have varying degrees of margin to the peaking factor limits which can be supported by CAOC. Westinghouse developed RAOC to directly determine the allowed band of Δ I operation required to support any plant specific peaking factor limit. We approved RAOC for referencing in licensing actions in a letter to E. P. Rahe (W) from C. Thomas (NRC), "Acceptance for Referencing of Licensing Topical Report WCAP-10216(P)(NS-EPR-2649)," dated February 28, 1983.

The affected sections of the Technical Specifications are: 3/4.2.1, Figure 3.2-1, Table 2.2-1, and Bases Section 3/4.2.1 and Figure B 3/4.2.1. The changes have all been made in accordance with those outlined in WCAP-10216PA. The procedure outlined in WCAP-10216PA was used for the analysis performed by Westinghouse for Cycle 1 of the Wolf Creek Generating Station Unit 1. The analysis is applicable only after the unit reaches 8000 MWD/MTU burnup in Cycle 1 and thereafter until the end of Cycle 1. Cycle specific evaluations are made

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by Westinghouse to determine if the allowable Δ I band curve remains valid or requires revision (by Technical Specification change).

We have reviewed the submittal for the RAOC technical specification change for Wolf Creek; the methodology used has previously been reviewed and approved by the staff. The proposed technical specification changes are those outlined in the topical (WCAP-10216PA). Therefore, the proposed technical specification changes are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of 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 exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (51 FR 9728) on March 20, 1986 , and consulted with the state of Kansas. No public comments were received, and the state of Kansas did not have any comments.

We have 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: P. O'Connor, PWR#4
M. Chatterton, PARS

Dated: April 22, 1986