

August 4, 1987

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

Dear Mr. Withers:

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SUBJECT: WOLF CREEK GENERATING STATION - AMENEMENT NO. 9 TO FACILITY  
OPERATING LICENSE NO. NPF-42 (TAC NO. 65318)

The Commission has issued the enclosed Amendment No. 9 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 May 7, 1987.

The amendment revises Table 3.3-5 to change the Engineered Safety Feature (ESF) response times for items 2.a. (Containment Pressure-High-1, Safety Injection), 3.a. (Pressurizer Pressure-Low, Safety Injection), and 4.a. (Steamline Pressure-Low, Safety Injection).

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

Sincerely,

*PS*

Paul W. O'Connor, Project Manager  
Project Directorate - IV  
Division of Reactor Projects - III,  
IV, V and Special Projects  
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 9 to License No. NPF-42
2. Safety Evaluation

cc w/enclosures:

See next page

PD4-LA *OM*  
PNoonan  
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PD4-PM  
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JCalvo  
07/21/87

*me*  
PSB  
JCraig  
07/10/87  
*mint*  
*7/16*

*me* 7/27/87

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. 9  
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., (licensees) dated May 7, 1987, 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 Specification

The Technical Specifications contained in Appendix A, as revised through Amendment No. 9, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

*Jose 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: August 4, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 9

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 page is also provided to maintain document completeness.

REMOVE PAGES

3/4 3-29  
3/4 3-30  
3/4 3-33  
3/4 3-34\*  
B 3/4 3-1  
B 3/4 3-2\*\*  
B 3/4 3-3\*\*  
B 3/4 3-4\*\*  
B 3/4 3-5\*\*  
B 3/4 3-6\*\*

INSERT PAGES

3/4 3-29  
3/4 3-30  
3/4 3-33  
3.4 3-34\*  
B 3/4 3-1  
B 3/4 3-2\*\*  
B 3/4 3-3\*\*  
B 3/4 3-4\*\*  
B 3/4 3-5\*\*  
B 3/4 3-6\*\*

\*Overleaf Page

\*\*Pages reissued due to repagination

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

<u>INITIATING SIGNAL AND FUNCTION</u>	<u>RESPONSE TIME IN SECONDS</u>
1. <u>Manual Initiation</u>	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Containment Purge Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Feedwater Isolation	N.A.
h. Auxiliary Feedwater	N.A.
i. Essential Service Water	N.A.
j. Containment Cooling	N.A.
k. Control Room Isolation	N.A.
l. Reactor Trip	N.A.
m. Emergency Diesel Generators	N.A.
n. Component Cooling Water	N.A.
o. Turbine Trip	N.A.
2. <u>Containment Pressure-High-1</u>	
a. Safety Injection (ECCS)	$\leq 29^{(7)}/27^{(4)}$
1) Reactor Trip	$\leq 2$
2) Feedwater Isolation	$\leq 7$
3) Phase "A" Isolation	$\leq 1.5^{(5)}$
4) Auxiliary Feedwater	$\leq 60$
5) Essential Service Water	$\leq 60^{(1)}$
6) Containment Cooling	$\leq 60^{(1)}$
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	$\leq 14^{(6)}$
9) Turbine Trip	N.A.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

<u>INITIATING SIGNAL AND FUNCTION</u>	<u>RESPONSE TIME IN SECONDS</u>
3. <u>Pressurizer Pressure-Low</u>	
a. Safety Injection (ECCS)	≤ 29 <sup>(7)</sup> /27 <sup>(4)</sup>
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 7
3) Phase "A" Isolation	≤ 2 <sup>(5)</sup>
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	≤ 60 <sup>(1)</sup>
6) Containment Cooling	≤ 60 <sup>(1)</sup>
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	≤ 14 <sup>(6)</sup>
9) Turbine Trip	N.A.
4. <u>Steam Line Pressure-Low</u>	
a. Safety Injection (ECCS)	≤ 39 <sup>(3)</sup> /27 <sup>(4)</sup>
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 7
3) Phase "A" Isolation	≤ 2 <sup>(5)</sup>
4) Auxiliary Feedwater	≤ 60
5) Essential Service Water	≤ 60 <sup>(1)</sup>
6) Containment Cooling	≤ 60 <sup>(1)</sup>
7) Component Cooling Water	N.A.
8) Emergency Diesel Generators	≤ 14 <sup>(6)</sup>
9) Turbine Trip	N.A.
b. Steam Line Isolation	≤ 2 <sup>(5)</sup>

TABLE 3.3-5 (Continued)

TABLE NOTATIONS

- (1) Diesel generator starting and sequence loading delays included.
- (2) Diesel generator starting delay not included. Offsite power available.
- (3) Diesel generator starting and sequence loading delay included. RHR pumps not included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.
- (4) Diesel generator starting and sequence loading delays not included. Offsite power available. RHR pumps not included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.
- (5) Does not include valve closure time.
- (6) Includes time for diesel to reach full speed.
- (7) Diesel generator starting and sequence loading delays included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is not included. Response time assumes only opening of RWST valves.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION  
SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MASTER RELAY TEST</u>	<u>SLAVE RELAY TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
1. Safety Injection (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, Turbine Trip, Component Cooling Water, Auxiliary Feedwater-Motor-Driven Pump, Emergency Diesel Generator Operation, Containment Cooling, and Essential Service Water Operation)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
c. Containment Pressure-High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
c. Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

### 3/4.3 INSTRUMENTATION

#### BASES

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#### 3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensure that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

ESF response times specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes 3 and 4) are based on values assumed in the non-LOCA safety analysis. These analyses take credit for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When the sequential operation of the RWST and VCT valves is not included in the response times (Note 7), the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times specified in Table 3.3-5 will assure that the assumptions used for the LOCA and non-LOCA analyses with respect to operation of the VCT and RWST valves are valid.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the Operability of a channel when its Trip Setpoint is found to exceed the deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties in calibrating the instrumentation. In Equation 3.3-1,  $Z + R + S \leq TA$ , the interactive effects of the errors in the rack and the sensor, and the

### 3/4.3 INSTRUMENTATION

#### BASES

#### 3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

"as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for the actuation. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the conditions. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) Feedwater System isolates, (4) the emergency diesel generators start, (5) containment spray pumps start and automatic valves position, (6) containment isolates, (7) steam line isolation, (8) Turbine trip, (9) auxiliary feedwater

## INSTRUMENTATION

### BASES

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#### 3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, (11) essential service water pumps start and automatic valves position, and (12) isolate normal control room ventilation and start Emergency Ventilation System.

#### Engineered Safety Features Actuation System Interlocks

The Engineered Safety Features Actuation System interlocks perform the following functions:

- P-4        Reactor tripped - Actuates Turbine trip, closes main feedwater valves on  $T_{avg}$  below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection block so that components can be reset or tripped.  
Reactor not tripped - prevents manual block of Safety Injection.
- P-11        On increasing pressure P-11 automatically reinstates safety injection actuation on low pressurizer pressure and low steamline pressure and automatically blocks steamline isolation on negative steamline pressure rate. On decreasing pressure; P-11 allows the manual block or Safety Injection on low pressurizer pressure and low steamline pressure and allows steamline isolation on negative steamline pressure rate to become active upon manual block of low steamline pressure SI.

#### 3/4.3.3 MONITORING INSTRUMENTATION

##### 3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated ACTION will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Control Room Emergency Ventilation Systems.

##### 3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the

## INSTRUMENTATION

### BASES

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#### 3/4.3.3.2 MOVABLE INCORE DETECTORS (Continued)

core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring  $F_A(Z)$  or  $F_{\nabla H}^N$  a full incore flux map is used.

Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Neutron Flux channel is inoperable.

#### 3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

#### 3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

#### 3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the facility from locations outside of the control room and that a fire will not preclude achieving safe shutdown. The Remote Shutdown System transfer switches, power circuits, and control circuits are independent of areas where a fire could damage systems normally used to shut down the reactor. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 3 and 19 and Appendix R of 10 CFR Part 50.

#### 3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 2,

## INSTRUMENTATION

### BASES

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#### 3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION (Continued)

"Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

#### 3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the Chlorine Detection System ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control and room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," Revision 1, January 1977.

#### 3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that both adequate warning capability is available for the prompt detection of fires and that Fire Suppression Systems, that are actuated by fire detectors, will discharge extinguishing agents in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of operable fire detectors must be greater.

The loss of detection capability for Fire Suppression Systems, actuated by fire detectors represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

#### 3/4.3.3.9 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the Reactor Coolant System and avoid or mitigate damage to Reactor Coolant System components. The allowable out-of-service times and Surveillance Requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

#### 3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/

## INSTRUMENTATION

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#### 3/4.3.3.10 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION (Continued)

Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

#### 3/4.3.3.11 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be adjusted to values calculated in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitor used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as  $1 \times 10^{-6}$   $\mu\text{Ci/cc}$  are measurable.

#### 3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Although the orientation of the turbine is such that the number of potentially damaging missiles which could impact and damage safety-related components, equipment, or structures is minimal, protection from excessive turbine overspeed is required.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION  
SUPPORTING AMENDMENT NO. 9 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

INTRODUCTION

By letter dated May 7, 1987, Wolf Creek Nuclear Operating Corporation acting as agent for the Kansas Gas & Electric Company, Kansas City Power and Light Company, and Kansas Electric Power Cooperative, Inc. (the licensees), informed the staff that Westinghouse Electric Corporation recently discovered that it had assumed simultaneous, rather than sequential, operation of valves in the calculation of time it takes to get a safety injection (SI) of 2000 ppm borated water into the Reactor Coolant System (RCS). Since the valves that transfer the charging pump suction from the Volume Control Tank (VCT) to the Refueling Water Storage Tank (RWST), which contains 2000 ppm boron, are operated sequentially, it was found that safety injection (ECCS) response times listed in Table 3.3-5 of the Technical Specifications (TS) were not achievable. They are too short by the 15-second delay encountered by the sequential operation of the two valves.

EVALUATION

The primary function of the ECCS is to supply water to the RCS in the event of a loss of coolant accident (LOCA). Since a LOCA is not a reactivity induced accident, the 2000 ppm boron is not immediately needed. It is only needed to maintain subcriticality in the long term. Therefore, for those SI actuation signals that are only intended to provide protection against a LOCA, this 15-second delay in the delivery of 2000 ppm borated water has no effect on the safety analysis.

The only non-LOCA transient impacted by this increased response time is the steam line break event. No other Updated Safety Analysis Report (USAR) Chapter 15 transient takes credit for short-term boration from the RWST.

The licensee compared calculations of the steam line break accident with and without the additional SI delay. The calculations showed no significant change in the consequences. One of the reasons for this is that the additional delay occurs early in the steam line break event when the RCS pressure is high and the SI flow rate is relatively small. In addition, the licensee stated that studies of the steam line break accident have generally shown that the

consequences are not sensitive to large changes in SI flow or boron concentration. The licensee, therefore, concluded that the departure from nucleate boiling design basis for the steam line break analysis is still met and that the conclusions presented in the USAR remain valid.

A Wolf Creek specific review of the steam line break analysis demonstrated that there is sufficient margin available in the analysis such that the USAR conclusions remain valid.

The staff has reviewed the licensee's analysis and finds the licensee's conclusions acceptable. Thus, the staff concludes that the requested technical specification changes satisfy the applicable regulatory requirements and are acceptable.

#### ENVIRONMENTAL CONSIDERATION

The amendment involves a change to a requirement with respect to 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.

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