



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D. C. 20555  
November 2, 1987

Docket

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:

SUBJECT: WOLF CREEK GENERATING STATION - AMENDMENT NO. 12 TO FACILITY  
OPERATING LICENSE NO. NPF-42 (TAC NO. 65671)

The Commission has issued the enclosed Amendment No. 12 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 16, 1987.

The amendment revises Wolf Creek Generating Station (WCGS) Technical Specification 3/4.3.1, Reactor Trip System Instrumentation, in the area of specified surveillance intervals and out-of-service times for Reactor Protection System Instrumentation. The requested revisions are based on changes approved generically as a result of the Nuclear Regulatory Commission's review of WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System" and WCAP-10271, Supplement 1."

The proposed changes are as follows:

1. Increase the surveillance interval for RPS analog channel operational tests from once per month to once per quarter,
2. Increase the time during which an inoperable RPS analog channel may be maintained in an untripped condition from one hour to six hours, and
3. Increase the time an inoperable RPS analog channel may be bypassed to allow testing of another channel in the same function from two hours to four hours.

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P PDR

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,

/s/

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. 12 to License No. NPF-42
2. Safety Evaluation

cc w/enclosures:  
See next page

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Mr. Bart D. Withers  
Wolf Creek Nuclear Operating Corporation

Wolf Creek Generating Station  
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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. 12  
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 June 16, 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.

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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:

2. Technical Specification

The Technical Specifications contained in Appendix A, as revised through Amendment No. 12, 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: November 2, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 12

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

3/4 3-3  
3/4 3-5  
3/4 3-6  
3/4 3-9  
3/4 3-10  
3/4 3-11  
3/4 3-12  
B 3/4 3-1  
B 3/4 3-2 through  
B 3/4 3-6

INSERT PAGES

3/4 3-3  
3/4 3-5  
3/4 3-6  
3/4 3-9  
3/4 3-10  
3/4 3-11  
3/4 3-12  
B 3/4 3-1  
B 3/4 3-2 through  
B 3/4 3-7 (Reissued due to repagination,  
no changes)

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level-High	3	2	2	1	6#
12. Reactor Coolant Flow - Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6#
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	6#
13. Steam Generator Water Level-Low-Low	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2	(1)
14. Undervoltage-Reactor Coolant Pumps	4-2/bus	2-1/bus	3	1	6#
15. Underfrequency-Reactor Coolant Pumps	4-2/bus	2-1/bus	3	1	6#
16. Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1	6#
b. Turbine Stop Valve Closure	4	4	1	1	11#
17. Safety Injection Input from ESF	2	1	2	1, 2	9

WOLF CREEK - UNIT 1

3/4 3-3

Amendment No. 12

WOLF CREEK - UNIT 1

3/4 3-4

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Power Range Neutron Flux, P-9	4	2	3	1	8
e. Power Range Neutron Flux, P-10	4	2	3	1, 2	8
f. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8
19. Reactor Trip Breakers	2 2	1 1	2 2	1, 2 3*, 4*, 5*	9 10
20. Automatic Trip and Interlock Logic	2 2	1 1	2 2	1, 2 3*, 4*, 5*	9 10

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

- \*Only if the Reactor Trip System breakers happen to be in the closed position and the Control Rod Drive System is capable of rod withdrawal.
- \*\*The boron dilution flux doubling signal may be blocked during reactor startup in accordance with normal operating procedures.
- #The provisions of Specification 3.0.4 are not applicable.
- ##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.
- ###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.
- (1)The applicable MODES and ACTION statements for these channels noted in Table 3.3-3 are more restrictive and therefore applicable.

ACTION STATEMENTS

- ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.
  
- ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:
  - a. The inoperable channel is placed in the tripped condition within 6 hours;
  - b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1; and
  - c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.
  
- ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:
  - a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint; or
  - b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.
  
- ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

- ACTION 5 - a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip Breakers, suspend all operations involving positive reactivity changes and verify valves BG-V178 and BG-V601 are closed and secured in position within the next hour.
- b. With no channels OPERABLE, open the Reactor Trip Breakers, suspend all operations involving positive reactivity changes and verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and every 12 hours thereafter, and verify valves BG-V178 and BG-V601 are closed and secured in position within 4 hours and verified to be closed and secured in position every 14 days.
- ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:
- a. The inoperable channel is placed in the tripped condition within 6 hours; and
- b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.
- ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.
- ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.
- ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour.
- ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 1 hour.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4) M(3, 4) Q(4, 6) R(4, 5)	Q(14)	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q(14)	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	Q(14)	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R(4, 5, 12)	S/U(1), Q(9, 14)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R(13)	Q(14)	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	Q(14)	N.A.	N.A.	1, 2
9. Pressurizer Pressure-Low	S	R	Q(14)	N.A.	N.A.	1
10. Pressurizer Pressure-High	S	R	Q(14)	N.A.	N.A.	1, 2
11. Pressurizer Water Level-High	S	R	Q(14)	N.A.	N.A.	1
12. Reactor Coolant Flow-Low	S	R	Q(14)	N.A.	N.A.	1

WOLF CREEK - UNIT 1

3/4 3-9

Amendment No. 12

TABLE 4.3-1 (Continued)

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

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<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>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
18. Reactor Trip System Interlocks (Continued)						
d. Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
e. Turbine Impulse Chamber Pressure, P-13	N.A.	R	R	N.A.	N.A.	1
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3*, 4*, 5*
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*

WOLF CREEK - UNIT 1

3/4 3-11

Amendment No. 12

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

\*Only if the Reactor Trip System breakers happen to be closed and the control rod drive system is capable of rod withdrawal.

##Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

- (1) If not performed in previous 31 days.
- (2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
- (3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
- (4) Neutron detectors may be excluded from CHANNEL CALIBRATION.
- (5) Detector plateau curves shall be obtained, evaluated and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
- (6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
- (7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
- (9) Quarterly surveillance in MODES 3\*, 4\* and 5\* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Quarterly surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to an increase of twice the count rate within a 10-minute period.
- (10) Setpoint verification is not required.
- (11) At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.
- (12) At least once per 18 months during shutdown, verify that on a simulated Boron Dilution Doubling test signal the normal CVCS discharge valves will close and the centrifugal charging pumps suction valves from the RWST will open within 30 seconds.
- (13) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.
- (14) Each channel shall be tested at least every 92 days on a STAGGERED TEST BASIS.
- (15) The surveillance frequency and/or MODES specified for these channels in Table 4.3-2 are more restrictive and, therefore, applicable.

### 3/4.3 INSTRUMENTATION

#### BASES

#### 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. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service times for the Reactor Protection Instrumentation System," supplements to that report, and the NRC's Safety Evaluation dated February 21, 1985. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

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 boroated water from the RWST. Injection of boroated 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,

## INSTRUMENTATION

### BASES

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#### REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

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 Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the 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 "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

## INSTRUMENTATION

### BASES

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#### REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

Engineered Safety Features components whose aggregate function best serves the requirements of the condition. 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 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 or High Steam Generator Water Level signal, allows 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 of 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.

## INSTRUMENTATION

### BASES

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#### 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 acutation 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 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  $F_Q(Z)$  or  $F_{\Delta H}^N$  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.

## INSTRUMENTATION

### BASES

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#### 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 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 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.

## INSTRUMENTATION

### BASES

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#### FIRE DETECTION INSTRUMENTATION (Continued)

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/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 6  $\mu\text{Ci/cc}$  are measurable.

## INSTRUMENTATION

### BASES

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#### RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION (Continued)

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  $1 \times 10^{-}$

#### 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. 12 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

On February 21, 1985, the NRC staff issued its Safety Evaluation Report (SER) on the Westinghouse Technical Specification Optimization Program for increased surveillance intervals and relaxed out-of-service times for testing and maintenance of the Reactor Trip System (RTS). The Optimization Program proposal was set forth in WCAP-10271, "Evaluation of Surveillance Frequencies and Out-of-Service Times for the Reactor Protection Instrumentation System," and Supplement 1 thereto.

By letter dated June 16, 1987, the licensee proposed changes to the Wolf Creek Technical Specifications based on the Optimization Program. The following documents the NRC staff's review of the requested changes.

EVALUATION

- Change 1 - The licensee proposed to increase the time during which an inoperable RTS analog channel may be maintained in an untripped condition from one hour to six hours (see Table 3.3-1, Action Statements 2.a and 6.a in the plant's proposed Technical Specifications). The staff finds this change acceptable on the basis set forth in the staff's February 21, 1985, SER.
- Change 2 - The licensee proposed to increase the time an inoperable RTS analog channel may be bypassed to allow testing of another channel in the same function from two hours to four hours (see Table 3.3-1, Action Statements 2.b and 6.b in the proposed Technical Specifications). The staff finds this change acceptable on the basis set forth in the staff's February 21, 1985, SER.
- Change 3 - The licensee proposed to add a cautionary note to the action statements (see new note 1 in Table 3.3-1 for Functional Unit 13) and to the surveillance requirements (see new note 15 in Table 4.3-1 for Functional Unit 13) referencing the more

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stringent requirements for Engineering Safety Feature Actuation System (ESFAS) channels for RTS analog channels common to ESFAS. The staff finds this change acceptable on the basis set forth in the staff's February 21, 1985, SER.

- Change 4 - The licensee proposed to delete Action Statement 7 and to reference Action Statement 6 for those reactor trip channels now referencing Action Statement 7 (see Table 3.3-1, Functional Units 11 and 12 in the plant's proposed Technical Specifications). This change, as discussed in Supplement 1 to WCAP-10271, results from the similarity and intent of Action Statements 6 and 7. Further, the staff in a July 24, 1985, letter from Harold R. Denton to L. D. Butterfield identified this as a specific acceptable revision in the model technical specifications provided to the Westinghouse Owners Group (WOG). On this basis, the staff finds this change acceptable as proposed by the licensee.
- Change 5 - The licensee proposed to reference Action Statement 6 in lieu of Action Statement 11 for Functional Unit 16a in Table 3.3-1. This change results from the similarity and intent of Action Statements 6 and 11. The staff in the July 24, 1985, letter identified Action Statement 6 as acceptable for this functional unit in the model technical specifications provided to the WOG. On this basis, the staff finds this change acceptable as proposed by the licensee.
- Change 6 - The licensee has proposed to revise Bases Sections 3/4.3.1 and 3/4.3.2 to include a discussion of WCAP-10271. Since the proposed changes are equivalent to those in the "WOG Guidelines for Preparation of Submittals Requesting Revisions to RPS Technical Specifications," Revision 1, dated September 3, 1985, and encompass the specific comments in the staff's July 24, 1985, letter, the staff finds this change acceptable as proposed.
- Change 7 - In Table 4.3-1 under Functional Units 2.a, 3, 4 and 6 thru 13, the licensee proposed to change the entries under "ANALOG CHANNEL OPERATIONAL TEST" from "M" to "Q(14)." A new note 14 is added under "TABLE NOTATIONS" to this table for a staggered test basis. We find these changes acceptable on the basis set forth in the staff's February 21, 1985, SER.
- Change 8 - In Table 4.3-1 under "ANALOG CHANNEL OPERATIONAL TEST:" for Functional Unit 2.b the entry is revised from "M" to "S/U(1)," for Functional Unit 5 "M" is deleted, for Functional Unit 18 the entries are revised from "M(8)" to "R." Under "TRIP

ACTUATING DEVICE OPERATIONAL TEST" the entries for Functional Units 14 and 15 are revised from "M" to "Q(14)." We find these changes acceptable on the basis set forth in the July 24, 1985, letter from Harold R. Denton to L. D. Butterfield which provided model technical specifications to the WOG related to the staff's generic SER on WCAP-10271.

- Change 9 - In Table 4.3.1 Functional Unit 18.b is deleted in its entirety. The staff identified this change as acceptable in the July 24, 1985, letter to the WOG. On the basis set forth therein, the staff finds this change acceptable as proposed.
- Change 10 - Under "TABLE NOTATIONS" for Table 4.3.1 note 1 is revised to require surveillance of Functional Unit 2.b prior to startup if not performed in the previous 31 days in lieu of the previous 7 days. Note 8 is also deleted since it is no longer referenced. In note 9 "monthly" is changed to "quarterly." On the basis set forth in the staff's July 24, 1985, letter to the WOG, we find these changes acceptable as proposed.

#### 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 or in a surveillance requirement. 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: November 2, 1987

Principal Contributor: Frederick Burrows  
Paul W. O'Connor