NUREG-0881
Supplement No. 3

Safety Evaluation Report

related to the operation of Wolf Creek Generating Station, Unit No. 1

Docket No. STN 50-482

Kansas Gas and Electric Company, et al.

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

August 1983

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Table of Contents

TABLE OF CONTENTS (Continued)

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Kansas Gas and Electric Company (KG&E), acting as applicant and agent for the owners, filed an application for an operating license for the Wolf Creek Generating Station (WCGS) (Docket No. 50-482) located in Coffey County, Kansas. KG&E is one of two utilities that joined together under the acronym SNUPPS (Standardized Nuclear Unit Power Plant System) to submit applications for operating licenses (OL) for a standard plant design for review under the Commission's standardization policy using the duplicate plant option described in Appendix N of Part 50 of the Code of Federal Regulations, Title 10. The other SNUPPS OL application submitted for review was by Union Electric Company (UE) for the Callaway Plant (Docket No. 50-483) located in Callaway County, Missouri.

In April 1982, the Nuclear Regulatory Commission (NRC) issued its Safety Evaluation Report (SER) (NUREG-0881) for the application filed by KG&E. The first supplement (SSER #1) was published in August 1982 and the second (SSER #2) in June 1983. These documents identified a number of items that were not resolved with the applicant. These items were categorized as:

1. Outstanding items which needed resolution prior to the issuance of an operating license.

2. Items for which the staff had completed its review and had determined positions for which there appeared to be no significant disagreement between the applicant and the staff. Further information was needed, however, to confirm these positions.

3. Items for which the staff had taken positions and would require implementation and/or documentation after the issuance of the operating license. These would be conditions to the operating license.

The purpose of this supplement (SSER#3) is to provide the staff evaluation of the items that have been resolved, address changes to the SER which resulted from the receipt of additional information and to provide resolutions of board notifications which have been served to various ASLBs. Copies of this SER supplement are available for inspection at the NRC Public Document Room, 1717 H Street NW, Washington, D.C. and at the William Allen White Library, Emporia State University, 1200 Commercial Street, Emporia, Kansas. Single copies may be purchased from the sources indicated on the inside front cover.

The NRC Project Manager assigned to the OL application for Wolf Creek is Mr. Joseph J. Holonich. Mr. Holonich may be contacted by calling (301) 492-7793 or writing:

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1.7 Summary of Outstanding Items

Listed below is an update of all of the outstanding items that require resolution prior to issuance of the operating license. The status of each of these items is given along with the document section where the item appears. The resolution of Outstanding Items B(3) is described in this supplement.

Part A*

A(1) Seismic and dynamic qualification of seismic Category I mechanical and electrical equipment (SER Section 3.10).**

A(2) Environmental qualification of safety-related electrical equipment (SER Section 3.11).**

A(3) TMI Action Plan (SER Section 22)

- I.A.1.1 Shift Technical Advisor
- I.D.1 Control room design review
- III.A.1.2 Upgrade emergency support facilities

A(4) Onsite Emergency Preparedness (SSER #2 Section 13.3).

Part B*

B(1) High-energy pipe break hazards analysis (closed SSER #1).

B(2) Pump and valve operability assurance program (SER Section 3.9.3.2).

B(3) Fire protection program - alternate shutdown panel (Closed SSER #3).

B(4) TMI Action Plan (SER Section 22)

- I.C.1 Guidance for evaluation and development of procedures for transients and accidents.
- I.C.8 Pilot monitoring of selected emergency procedures for near-term operating license applications.
- II.B.2 Plant shielding to provide access to vital areas and protect safety equipment for postaccident operation (closed SSER #2)

1.8 Confirmatory Items

The following is an update of each of those confirmatory items in Section 1.8 of the SER. As a result of a September 30, 1982 meeting held between the staff and representatives from SNUPPS, UE, KG&E and Bechtel and the Power Systems Branch site audit held April 5-8, 1982 at the Callaway and Wolf Creek Plants several items have been resolved. In addition, several new concerns arose from the site visits and the removal of a number of license conditions.

*Part A lists the site-specific items while Part B contains the SNUPPS items which are common to both Wolf Creek and its sister plant Callaway.

**This item includes both plant-specific and duplicate-plant information.

Part A*

A(1) UHS dam dispersiveness (Closed SSER #1).

A(2) Main dam seepage (SER Section 2.5.6.8).

A(3) Site-specific seismic structural analysis (Closed SSER #1).

A(4) Identification of base metal and heat-affected zone surveillance material (Closed SSER #2).

A(5) Pressure-temperature limits (Closed SSER #2).

A(6) Fire protection site visit (SER Section 9.5.1.8).

A(7) Security plan (Closed SSER #1).

A(8) TMI Action Plan (SER Section 22).

II.K.1 IE Bulletins on measures to mitigate small break LOCAs and loss-of-feedwater accidents.

III.A.2 Improving licensee emergency preparedness--long-term.

Part B*

B(1) Additional seismic instrumentation and control room indication (Closed SSER #1).

B(2) Analysis of steam generator tube plugging (SER Section 3 7.4).

B(3) Testing of pressure isolation valves (Closed SSER #2).

B(4) Fuel assembly structural response to seismic and loss-of-coolant accident (LOCA) forces (Closed SSER #2)

B(5) Preservice inspection testing program (SER Sections 5.2.4.1 and 6.6.1).

B(6) Steam generator inservice inspection (SER Sections 5.4.2.2).

B(7) ECCS analysis (Closed SSER #1).

B(8) Steam generator level control and protection (SER and SSER #1 Section 7.3.2.8).

B(9) Capability for safe shutdown following loss of a bus supplying power to instruments and controls (SER Section 7.4.3.1).

*Part A lists the site-specific items while Part B contains the SNUPPS items which are common to both Wolf Creek and its sister plant Callaway.

B(10) Operator actions required to maintain safe shutdown from outside control room (SER Section 7.4.3.2).

B(11) Reactor coolant temperature indicators on the auxiliary shutdown panel (SER and SSER #3 Section 7.5.2.1).

B(12) Volume control tank level control and protection interaction (SER and SSER #3 Section 7.6.7.2).

B(13) Boron dilution control (SER Sections 7.6.7.3 and 15.2.3.1 and SSER #3 Section 7.6.7.3).

B(14) Environmental qualification of control systems (SER Section 7.7.11.3).

B(15) Circuitry for automatic transfer of diesel generator from test to auto control mode (Closed SER #3).

B(16) Diesel generator reliability qualification testing (Closed SER #3).

B(17) Circuitry for bypass of protective circuitry (Closed SER #3).

B(18) Circuitry for inservice testing per Regulatory Guide 1.108 (Closed SSER #3).

B(19) Low and or degraded grid voltage (SER and SSER #3 Section 8.3.1.2).

B(20) Use of regulating-type transformer as isolation device (Closed SSER #3).

B(21) Isolation of control room and remote circuits (SER and SSER #3 Section 8.3.1.6).

B(22) Sequencing of loads on the offsite power system (SER and SSER #3 Section 8.2.2.3).

B(23) Submerged electrical equipment (Closed SSER #3).

B(24) Separation between redundant safety-related cables inside control panels (Closed SSER #3).

B(25) Compliance with position 1 of Regulatory Guide 1.63 (SER Section 8.3.3.6)

B(26) Monitoring of rocker arm lube oil system temperature for diesel generators (SER Section 9.5.7).

B(27) Reactor coolant pump locked rotor accident (Closed SSER #1).

B(28) TMI Action Plan (SER Section 22)

II.D.1 Performance testing of BWR and PWR relief and safety valves (SER and SSER #3).

B(28) TMI Action Plan (SER Section 22) (continued)

II.E.1.1 Recommendation GS-2, physical locking of isolation valve.

II.E.4.2 Containment isolation dependability.

II.F.1 Additional accident monitoring instrumentation Attachments 1, 2, and 3 (Attachment 3 Closed SSER #2).

II.K.2.13 Thermal Mechanical Report--Effect of High-Pressure Injection on Vessel Integrity for Small-Break LOCA With No Auxiliary Feedwater (Closed SSER #2).

II.K.3.2 Report on overall safety effect of PORV isolation system (Closed SSER #2).

II.K.3.11 Justification of use of certain PORVs.

III.D.1.1 Integrity of systems outside containment likely to contain radioactive material.

B(29) Test of engineered safeguards P-4 interlock (SSER #1 Section 7.3.2.2).

B(30) Automatic indication of block of signals initiating auxiliary feedwater following trip of main feedwater pumps (SSER #1 Section 7.3.2.7).

B(31) Actuation of valve component level windows on the bypassed and inoperable status panel (SSER #1 Section 7.5.2.2).

B(32) Post accident monitoring (SSER #2 Section 7.5.2.3.1).

B(33) Indicators, alarms, and test features provided for instrumentation used for safety functions (SSER #3 Section 7.3.2.9).

B(34) Interlocks for reactor coolant system pressure control during low-temperature operative (SSER #3 Section 7.6.7.1)

B(35) Capacity and capability of offsite circuits (SSER #3 Sections 8.2.2.1).

1.9 License Conditions

The following is an update of each of the license conditions described in Section 1.9 of the SER. License Conditions B(15) and B(16) have been removed based on information received during the September 30, 1982 meeting. License Conditions B(8), B(9), B(12), B(13), B(14) have also been removed and made new confirmatory issues or incorporated into existing confirmatory items.

Part A*

A(1) Compliance with Appendix R of 10 CFR 50, Fire Protection (SER Section 9.5.1.7)**

*Part A lists the site-specific items while Part B contains the SNUPPS items which are common to both Wolf Creek and its sister plant Callaway.

**This item includes both plant-specific and duplicate-plant information.

Part B*

B(1) Surveillance of Hafnium control rods (SER and SSER #2 Section 4.2.3.1(10)).

B(2) The applicant must provide an initial inservice inspection program which conforms to the applicable ASME Code Edition and 10 CFR 50 (SER Sections 5.2.4 and 6.6.1).

B(3) The applicant must implement the secondary water chemistry monitoring and control program proposed in the SNUPPS FSAR (through Revision 6) and their letter dated May 8, 1981 (SER Section 5.4.2.3).

B(4) Sensor time response testing (SER Section 7.2.2.1).

B(5) Tests of engineered safeguards P-4 interlocks (Removed SSER #1).

B(6) Automatic indication of block of signals initiating auxiliary feedwater following trip of the main feedwater pumps (Removed SSER #1).

B(7) Steam generator level control and protection (Removed SSER #1).

B(8) Indicator, alarms, and test features provided for instrumentation used for safety functions (Removed SSER #3).

B(9) Reactor coolant temperature indications on the auxiliary shutdown panel (Removed SSER #3).

B(10) Actuation of valve component level windows on the bypassed and inoperable status panel (Removed SSER #1).

B(11) Post accident monitoring (Removed SSER #2).

B(12) Interlocks for reactor coolant system (RCS) pressure control during low temperature operation (Removed SSER #3).

B(13) Volume control tank level control and protection interaction (Removed SSER #3).

B(14) Boron dilution control (Removed SSER #3).

B(15) Bypass of protective trips on diesel generator (Removed SSER #3).

B(16) Installation of battery discharge alarm (Removed SSER #3).

B(17) TMI Action Plan (SER Section 22)

II.B.3 Post accident sampling capability.

B(18) Operation restriction above 90% of full power (SSER #1 Section 15.2.3.3).

B(19) Experienced PWR operator or startup engineer required onshift for one year or until sufficient operating experience is acquired (SSER #1 Section 18).

*Part A lists the site-specific items while Part B contains the SNUPPS items which are common to both Wolf Creek and its sister plant Callaway.

2 SITE CHARACTERISTICS

2.5 Geology and Seismology

2.5.2 Seismology

For the purpose of licensing of facilities in the Southeastern U. S., the NRC has taken a position, based primarily on the advice of the U. S. Geological Survey (USGS), that any reoccurrence of the 1886 Charleston, S.C. earthquake (Modified Mercalli Intensity (MMI) X, estimated Magnitude about 7) would be confined to the Charleston area. That is, the Charleston earthquake is assumed to be associated with a geologic structure in the Charleston area. Nuclear power plants in the region east of the Appalachian Mountains are, therefore, usually controlled in their seismic design, according to Appendix A to 10 CFR Part 100, by the maximum historical earthquake not associated with a geologic structure. This controlling earthquake is typically an MMI VII or VIII.

The position recently received from the USGS clarifies their original recommendation and indicates that:

> "Because the geological and tectonic features of the Charleston region are similar to those in other regions of the eastern seaboard, we conclude that although there is no recent or historical evidence that other regions have experienced strong earthquakes, the historical record is not, of itself, sufficient grounds for ruling out the occurrence in these other regions of strong seismic ground motions similar to those experienced near Charleston in 1886. Although the probability of strong ground motion due to an earthquake in any given year at a particular location in the eastern seaboard may be very low, deterministic and probabilistic evaluations of the seismic hazard should be made for individual sites in the eastern seaboard to establish the seismic engineering parameters for critical facilities."

This clarification is not intended to recommend that we categorically consider a Charleston-type event in the seismic design of all nuclear plants in the eastern seaboard of U. S. The USGS does believe, however, that an earthquake of this size should not be categorically ruled out at locations away from Charleston based solely on the statement in the December 30, 1980 USGS letter which states, "Consequently, earthquakes similar to the 1886 event should be considered as having the potential to occur in the vicinity of Charleston and seismic engineering parameters should be determined on that basis." Instead, this clarification provides guidance that indicates that such a conclusion should be reached only after deterministic and probabilistic evaluations of the seismic hazard for individual sites have been made.

Although this USGS clarification appears to deal with plants in the eastern U. S., the staff transmitted it to boards for all plants east of the Rocky Mountains but does not regard this issue as an open item. Therefore, this board notification does nct change the staff conclusions presented in the SER (NUREG-0881) and no further information is needed from the applicant.

4 REACTOR

4.5 Reactor Materials

4.5.2 Reactor Internals and Core Support Materials

A recent board notification relates to failures of the support pins that are attached to the bottom of the control rod drive guide tubes in Westinghouse designed reactors. The support pins align the bottom of the control rod drive guide tube assembly into the top of the upper core plate in a manner that provides lateral support and accommodates thermal expansion of the guide tube relative to the core plate. Westinghouse's analysis indicated that the failures were caused by stress corrosion cracking. Westinghouse now recommends a revised heat treatment for the pins, a revised pin body design and a reduction in the torque on the lock nut. The applicant has advised that the pins will be replaced and installed in conformance with current Westinghouse recommendations prior to fuel load[1]. The staff has been following this problem including the Westinghouse program, agrees with the Westinghouse analysis, and concurs in the revisions that have been made in the design.

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[1]Nicholas A. Petrick (SNUPPS) letter to Harold R. Denton (NRC), Subject: "Licensing Issues," July 28, 1983.

6 ENGINEERED-SAFETY-FEATURES

6.3 Emergency Core Cooling Systems

During a test conducted in the Semiscale facility during July 1982 in which the "feed and bleed" mode of core coolang was being tested, uncovery of the core simulator occurred. This core simulator uncovery was not expected to occur. In their announcement, the staff stated that there was insufficient information to draw any conclusion from the results, and that these results did not adversely impact its position regarding reliance on bleed and feed cooling.

After the staff completed the evaluation of this information it concluded that Semiscale Test S-SR-2 does not exhibit any new phenomena and can be adequately predicted by our computer codes. The staff's evaluation and conclusions were based on RELAP-5 analyses of both the Semiscale S-SR-2 test and a corresponding feed and bleed mode of operation for a typical Westinghouse 4-loop plant. Further, regardless of the conclusions that may have been reached from this test regarding the viability of the feed and bleed mode of cooling, feed and bleed cooling is not a design basis requirement considered necessary to meet the Commission's Regulations for any LWRs currently licensed or being considered for a license.

Also, based on the results of additional RELAP-5 calculations of the Semiscale Test S-SR-2 it was concluded that the test did not exhibit any new phenomena and that the RELAP-5 code adequately predicted the test data.

7 INSTRUMENTATION AND CONTROLS

7.3 Engineered Safety Features Actuation System

7.3.2 Resolution of Issues

7.3.2.9 Indicator, Alarm, and Test Features Provided for Instrumentation Used for Safety Functions

As noted in the SER (NUREG-0881), the applicant has committed to provide additional indicators and alarms on the plant computer. The staff has reviewed the indicators and alarms to be provided and found them acceptable. This represents an adequate commitment to License Condition B(8) which is no longer required. However, until the applicant has formally notified the staff of completion of installation of this design, this will continue to be carried as Confirmatory Item B(33). Complete implementation of this design is required prior to fuel load.

7.5 Information Systems Important to Safety

7.5.2 Resolution of Issues

7.5.2.1 Reactor Coolant Temperature Indicators on the Auxiliary Shutdown Panel

As noted in the SER (NUREG-0881), the applicant has committed to temperature indication design criteria which were found acceptable by the staff. This represents an adequate commitment to License Condition B(9) which is no longer required. However, until the applicant has (a) formally notified the staff of completion of installation of this design and (b) provide the design features in the FSAR including information verifying that, even with a single failure, the reactor coolant hot-leg and cold-leg temperature indication will be available at the auxiliary shutdown panel for a loop having an operable steam generator, this will be carried as Confirmatory Item B(11). Complete implementation of this design is required prior to fuel load.

7.6 Interlock Systems Important to Safety

7.6.7 Resolution of Issues

7.6.7.1 Interlocks for Reactor Coolant System Pressure Control During Low-Temperature Operation

As noted in the SER (NUREG-0881), the applicant has agreed to provide a modified design which the NRC staff has reviewed and found to be acceptable. This represents an adequate commitment to License Condition B(12) which is no longer required. However, until the applicant has formally notified the staff of completion of installation of this design, this will remain as Confirmatory Item B(34). Complete implementation of this design is required prior to fuel load.

7.6.7.2 Volume Control Tank Level Control and Protective Interaction

As noted in the SER (NUREG-0881), the applicant has committed to an acceptable design which the staff will require. This represents an adequate commitment to License Condition B(13) which is no longer required. However, until the applicant has (a) formally notified the staff of completion of installation of the required design, and (b) provided the design features in the FSAR, this will remain as Confirmatory Item B(12). Complete implementation of this design is required prior to fuel load.

7.6.7.3 Boron Dilution Control

As noted in the Wolf Creek SER (NUREG-0881), the applicant has provided an acceptable commitment to a design for terminating boron dilution. This represents an adequate commitment to License Condition B(14) which is no longer required. However, until the applicant has formally (a) provided the design features in the FSAR and (b) notified the staff of completion of installation of this design, this will remain as Confirmatory Item B(13). Complete implementation of this design is required prior to fuel load.

8 ELECTRIC POWER SYSTEMS

8.2 Offsite Power System

8.2.2 Compliance with GDC 17

8.2.2.1 Capacity and Capability of Offsite Circuits

In regard to the offsite circuits within the SNUPPS standardized power block from the interface to the Class 1E busses, the staff documented in the SER that it had reviewed and found acceptable the applicant's description and analysis of compliance with GDC 5, 17, and 18 included in Revision 7 to the FSAR. The subject routing of offsite circuits was subsequently reviewed as part of the confirmatory site visit held during the week of April 5-8, 1983. As a result of this site visit, a new concern was identified.

The two ESF transformers XNB01 and XNB02 shown on Figure 8.3-1 of the FSAR are separated by a 3-hour fire wall as documented in Revision 7 to the SNUPPS FSAR. During the site visit, the staff expressed the concern that an oil fire at one transformer could overflow with the fire suppressant water sprinklers operating such that the fire could go around the fire wall and cause damage to the other offsite circuit. This concern will be pursued with the applicant and the results of the staff evaluation will be reported in a supplement to the SER. Until this issue is resolved, it will be carried as Confirmatory Item B(35).

8.2.2.3 Sequencing of Loads on the Offsite Power System

In the SER (NUREG-0881), the staff indicated that the reliability study for the SNUPPS solid state load sequencer would be verified during a confirmatory site visit. At a September 30, 1982 meeting, the applicant presented reports J-104-0221-05 and J-104-0256-06 for staff review. Based on review of these reports the staff is unable to conclude that the reliability of the offsite power to the Class 1E busses will not be compromised by using the same load sequencer to sequence loads on both the onsite and offsite power sources. This item has been discussed with the applicant and the results of the staff evaluation will be included in a supplement to the SER.

8.3 Onsite Emergency Power Systems

8.3.1 Onsite AC Power System's Compliance with GDC 17

8.3.1.1 Compliance with the Guidelines of Regulatory Guide 1.9, Revision 1

b. Automatic Transfer of Diesel From Test to Automatic Control Mode

In the SER the staff indicated that the SNUPPS design for automatic transfer of the diesel generator from test to automatic control modes would be verified during a confirmatory site visit. At a September 30, 1982 meeting with the applicant, the staff reviewed drawings E-03KJ01A(Q) Revision 6 and E-03KJ03A(Q) Revision 7 and confirmed the system design. Confirmatory Item B(15) is, therefore, closed.

d. Diesel Generator Reliability Qualification Testing

In the SER the staff required that the 300 start-and-load test results, as well as the test result for the other tests recommended by IEEE 387, be provided for staff audit verification. Subsequently, the applicant provided the subject test results during the staff's site audit held during the week of April 5-9, 1983. The staff reviewed results from about 20 of the 300 diesel start-and-load tests and also the margin qualification test required by IEEE Standard 387. Based on the review of these tests and results, the staff concludes that there is reasonable assurance that a diesel generator of the design to be used at Wolf Creek has been successfully tested in accordance wtih the qualification test guidelines of IEEE Standard 387-1977 as augmented by Regulatory Guide 1.9 and is, therefore, acceptable.

Preoperational testing required on each diesel generator by IEEE 387 as augmented by Regulatory Guides 1.9 and 1.108 will be performed as part of the Wolf Creek preoperational test program and the results will be reviewed.

If any significant problems are found, they will be addressed in a supplement to the SER. This resolves Confirmatory Item B(16).

g. Diesel Generator Protective Trips

In the SER the staff required, as a condition to the license, that the generator ground overcurrent and voltage restrained overcurrent protective trips be bypassed or be designed with two-of-three logic in accordance with the guidelines of position 8 or Regulatory Guide 1.9 (Revision 1). By Revision 8 to the FSAR, the applicant committed to meet position 8 of Regulatory Guide 1.9. Confirmatory Item B(17) is, therefore, resolved.

Also, in the SER the staff indicated the SNUPPS design for protective trip circuits (that are bypassed or two-out-of-three logic) that have been implemented in accordance with the staff position would be verified during a confirmatory site visit. At a September 30, 1982 meeting with the applicant, the staff reviewed (a) drawings numbered E-03NE10(Q) Revision 9 and E-03NE11(Q) Revision 9 which show the bypass logic, and (b) drawings numbered E-03KJ01B(Q) Revision 3 and E-03KJ038(Q) Revision 3 which show the two-out-of-three logic. Based on the review of these drawings the staff confirmed the design. Therefore, License Condition B(15) is no longer required.

h. Test Capability

In the SER the staff indicated that the design capability to simulate the parameters of operation outlined in Regulatory Guide 1.108 would be verified during a confirmatory site visit. This verification will be accomplished as part of the staff review of the Wolf Creek preoperational testing program. If any significant problems are found, they will be addressed in a supplement to the SER. This resolves Confirmatory Issue B(18).

8.3.1.2 Low and/or Degraded Grid Voltage Condition

In the SER the staff indicated that the design for the low and/or degraded grid voltage condition was acceptable pending documentation of the proposed design description in the FSAR and verification of design implementation. In Revision 7 to the FSAR, the applicant provided the required documentation. At a September 30, 1982 meeting with the applicant, the staff reviewed drawings E-03NB12(Q) Rev. 6, E-03NB13(Q) Rev. 6, E-03NB14(Q) Rev. 6, E-03NB15(Q) Rev. 6, E-03NB01(Q) Rev. 8, and E-03NB02(Q) Rev. 8. Based on the review of these drawings, the staff confirmed the design. This portion of Confirmatory Item B(19) is complete.

Also, the staff indicated in the SER that the Wolf Creek voltage drop analysis and testing would be verified. The applicant has not yet submitted the analysis and test results. This portion of B(19) remains confirmatory. The results of the staff confirmation will be reported in a supplement to the SER.

8.3.1.3 Nonsafety Loads Powered From the Class 1E AC/Distribution System

In the SER the staff indicated that testing performed to demonstrate the isolation capability of a regulating type transformer were unacceptable for long-duration faults. The applicant committed to perform additional analysis substantiated by test to demonstrate long-duration fault capability of the transformer. In letters dated May 9 and 13, 1983, the applicant provided the results of the requested test subtantiated by analysis. In addition, the applicant provided in amendment 11 to the FSAR a modified design to incorporate redundant circuit breakers in the transformer primary circuit. Based on the modified design and the results of the additional tests, the staff concludes that the design for isolating non-Class 1E circuits is acceptable. Therefore, Confirmatory Item B(20) is resolved.

8.3.1.6 Electrical Independence Between Local and Control Room Panels

The staff indicated in the SER that the SNUPPS design for isolation of diesel generator control circuits between the control room and remote panels would be verified during the staff confirmatory site visit. At a September 30, 1982 meeting with the applicant, the staff reviewed drawings numbered E-03KJ01A(Q) Rev. 6 and E03KJ03A(Q) Rev. 7. Based on these drawings, the staff confirmed the isolation of the subject diesel generator controls. This item is, therefore, closed.

A new item was identified during the staff's site audit held during the week of April 5-9, 1983. Redundant load sequencers are located in the same area of the control room and their output relays are mounted back-to-back in a common panel. It is the staff concern that there is insufficient separation.

In regard to a single failure of one load sequencer, it is the staff position, in accordance with Section 4.6 of IEEE Standard 308-1974, that the load sequencer be physically separated from its redundant counterpart or mechanically protected as required to prevent the occurrence of a common failure mode. This item will be pursued with the applicant. The results of the staff evaluation will be reported in a supplement to the SER.

In regard to the design basis event (exposure fire), it is the staff position that at least one of the two redundant load sequencers located in the control room be electrically independent of load sequencing controls located at a remote panel. Any failure of the load sequencers shall not cause loss of either onsite or offsite AC power such that AC power cannot be reestablished in a short period of time at the remote panel. The results of the staff evaluation will be reported in a supplement to the SER.

8.3.2 Onsite DC System Compliance with GDC 17

8.3.2.1 DC Monitoring and Annunciation

In the SER the staff required, as a condition to the license, that a battery discharge alarm be provided in the control room. In Revision 8 to the FSAR, the applicant documented a design change in which a control room computer alarm for battery high rate of discharge will be installed. License Condition B(16) is, therefore, no longer required.

8.3.3 Common Electrical Features and Requirements

8.3.3.1 Compliance with GDC 2 and 4

8.3.3.1.1 Submerged Electrical Equipment as a Result of a LOCA

In the SER the staff indicated that the control circuit design for unqualified submerged solenoid-operated isolation valves would be verified during the staff's confirmatory site visit to assure that submergence will not cause the valves to spuriously open. At a September 30, 1982 meeting with the applicant, the staff reviewed drawing E-03M04(Q) Rev. 4. Based on this drawing, the staff confirmed that there is reasonable assurance that submergence will have no adverse effect on electric power systems and will not cause the valves to change position. Confirmatory Item B(23) is, therefore, closed.

8.3.3.3 Physical Independence (Compliance with GDC 17)

Separation for Cables Inside and Approaching Panels

In the SER the staff exprcssed the concern that conduit, steel plate and other material proposed as possible barriers for separation between cables inside panels may not provide the equivalent of six inches of free air space. To resolve this concern in the SER, the staff required that the adequacy of each type of barrier be subtantiated by test. Subsequently, in SSER No. 1, this test requirement was changed so that the adequacy of barriers would be determined by observation during the staff's confirmatory site visit. During the site visit held the week of April 5-9, 1983, the staff observed the subject barriers and concluded that their use does not pose a significant safety problem. For the most part cables are separated by greater than 6 inches of free air space and, thus do not require a barrier. For those few locations where separation is less than six inches, a metal barrier accompanied by free air space, with a few deviations, were observed and judged to be acceptable. Based on the staff judgment, Confirmatory Item B(24) is closed.

9 AUXILIARY SYSTEMS

9.5 Othe Auxiliary Systems

9.5.1 Fire Protection Review

9.5.1.5 Alternate Shutdown

Review of the SNUPPS fire protection of safe shutdown capability included the list of equipment and components identified in Section 3.11(B) of the SNUPPS Final Safety Analysis Report (FSAR) as being necessary for hot and/or cold shutdown, the safe cold shutdown analysis in FSAR Section 5.4A, the remote shutdown capability described in FSAR Section 7.4, the cable separation discussed in FSAR Section 8.3 and the fire hazards analysis and design comparison with Appendix R in FSAR Section 9.5. We also reviewed the control room fire hazards analysis submitted by letter dated November 15, 1982.

The applicant's safe shutdown analysis and fire hazards analysis demonstrates that redundancy exists for systems needed for hot and cold shutdown. The safe shutdown analysis included components, cabling and support equipment needed to achieve hot and cold shutdown. Thus, in the event of a fire anywhere in the plant, at least one train of systems would be available to achieve and maintain hot shutdown and proceed to cold shutdown.

For hot shutdown at least one train of the following safe shutdown systems would be available: Auxiliary feedwater (AFW) system, steam generator atmospheric dump valves, reactor coolant system, and the chemical and volume control system. For cold shutdown at least one train of the residual heat removal (RHR) system would be available. The RHR system would be used for long-term decay heat removal and provides the capability to achieve cold shutdown within 72 hours after a fire. The availability of these systems includes the components, cabling and support equipment necessary to achieve cold shutdown. The support equipment includes the diesel generators, emergency service water system, component cooling water system, and the necessary ventilation systems.

The applicant's fire hazards analysis demonstrated that except for inside containment and inside the control room, redundant systems and cabling needed for safe shutdown are separated in accordance with III.G.2.a, b, or c of Appendix R. For the control room, the applicant has provided alternate shutdown capability outside the control room in accordance with III.G.3 of Appendix R. Inside the containment there is at least 20 feet between redundant safe shutdown divisions or between diverse system such as the letdown isolation valves and the power operated relief and block valves. Thus, the requirements of III.G.2.d are met for separation inside containment.

The applicant performed an electrical train separation study in order to ensure that at least one train of the above equipment is available in the event of a fire in areas which might affect these components. Safe shutdown equipment and cabling were identified and traced through each fire area from the components to the power source. Additional equipment and cabling considered as associated either because of a shared common power source or common enclosure or whose fire induced spurious operation could affect shutdown were also identified. Extensive use of computer program checks were used to ensure separation. Each circuit and raceway is identified in the computer program, and the identification includes the applicable separation group. The program is used to check that cables of a particular separation group are routed through the appropriate raceways.

We have reviewed the applicant's method of determining that the separation criteria of Appendix R are met and have reviewed the associated circuits identified by the applicant and the actions necessary or modifications made to prevent spurious operation that would affect safe plant shutdown. Based on our review we conclude that the applicant has adequately addressed the effects of associated circuit interaction and that the necessary isolation devices and procedures are adequate to ensure that such circuit interactions will not prevent safe shutdown. We further conclude that the applicant's methodology for verifying that separation is in accordance with Appendix R, Item III.G.2 is acceptable.

The applicant's analysis indicated that the only area outside containment where redundant divisions are not separated by barriers in accordance with III.G.2 is the control room. Alternate shutdown measures were required for the control room in order to assure the availability of the safe shutdown sytems. In the event that a fire disables the control room the remote shutdown panel associated with train B equipment located in a separate fire area of the auxiliary building provides an alternative to fire protection separation within the control room. The control functions and indications provided at the remote shutdown panel are electrically isolated or otherwise separate and independent from the control room. Refer to Section V.C of this SER for further discussion of alternative shutdown capability.

Based on the above, the systems identified for achieving and maintaining safe shutdown in the event of a fire are acceptable and the methodology used to assure adequate protection of safe shutdown systems is in accordance with Section III.G of Appendix R and therefore is acceptable.

V.C Alternative Shutdown Capability

Section 7.4 of the SNUPPS FSAR describes the remote shutdown panels' capability. Section 5A of the FSAR and the control room fire hazard analysis describe remote shutdown capability for equipment not on the remote shutdown panel. The design objective of the remote shutdown system for the purposes of this evaluation is to achieve and maintain cold shutdown in the event of a fire in the control room. The train B remote shutdown panel will be the primary alternative shutdown panel since the necessary instruments and controls on this panel are isolated or isolable from the control room.

The turbine driven AFW pump, train B motor driven AFW pump, associated AFW controls, the atmospheric dump valves for steam generators B and D, the group B pressurizer backup heaters, and the train B letdown isolation valve can be controlled at the train B alternate shutdown panel for maintaining hot standby. Separate isolation switches provided at local stations for control of support systems and cold shutdown systems will be used in conjunction with a procedural approach using pre-planned operator actions to maintain hot standby and to achieve and maintain cold shutdown within 72 hours.

The design of the remote shutdown system complies with the performance goals outlined in Section III.L of Appendix R. Reactivity control is accomplished by manual scram before the operator leaves the control room and boron addition via the chemical and volume control system using the refueling water storage tank (RWST) and the charging pumps. The reactor coolant makeup function is also performed by the charging pumps and RWST. Reactor coolant inventory is assured by maintaining reactor coolant pump seal cooling and seal injection, and by isolating all possible paths of inventory loss such as PORVs, RHR suction lines, normal and excess letdown lines and the reactor vessel head vent. All these operations including reactor scram can be accomplished from outside the control room. Reactor decay heat removal to hot shutdown is accomplished by the AFW system through the steam generators and atmospheric dump valves. Decay heat removal to cold shutdown is achieved by the residual heat removal system. The following instruments on the alternate shutdown panel will be used to monitor process variables:

Pressurizer level
Reactor coolant system pressure (wide range)
Steam generator level (wide range)
AFW flow
Reactor coolant cold leg temperature (T_C)
Reactor coolant hot leg temperature (T_H)
Source range nuclear instrument

The above instrumentation will be isolated from the control room on the train B alternate shutdown panel. Isolated valve position indication for the AFW system, letdown isolation valve, and the atmospheric dump valves are also located on the train B panel.

The staff has reviewed actions required by the procedures for achieving and maintaining safe plant shutdown following a fire. For hot standby the immediate actions are mainly precautionary measures to assure no spurious operations occur due to the control room fire. Some operations require cutting a control power cable at the equipment to ensure that a fault in the control room does not prevent certain equipment operation. Such actions may be required for the fuel oil transfer pumps, fuel pool cooling system and some ventilation dampers that are not immediately necessary for or detrimental to maintaining hot standby conditions. These actions will be described in the procedures. For achieving and maintaining cold shutdown local operation of RHR isolation valves, letdown valves and certain CCW system valves may be required and will be in the cold shutdown procedures. The staff has reviewed the proposed actions and manpower requirements and conclude they are in accordance with III.L.4 and III.L.5 to Appendix R since they can be accomplished exclusive of fire brigade members and are straightforward and uncomplicated such that cold shutdown can be achieved within 72 hours.

Based on our review, the staff concludes that the alternative shutdown capability for the control room meets the requirements of Appendix R, Section III.L, and is therefore acceptable. This closes Outstanding Item B(3).

15 ACCIDENT ANALYSIS

15.4 Radiological Consequences of Design-Basis Accidents

15.4.4 Steam Generator Tube Rupture Accident

Analysis of the results of the steam generator tube rupture accident which occurred at the Ginna Nuclear Power Plant determined that certain of the assumptions usually made in staff design basis accident caculations for off-site doses were non-conservative for part of the accident. For this reason, the staff has required reduced radioiodine limiting conditions for operation for the Ginna primary coolant activity. The staff is continuing to review the accident as it relates to Ginna and other plants. At this time, the staff does not intend to require lower primary coolant activity in the Technical Specifications for Wolf Crrek because of its capability for high head injection for small break accidents. Should future review of this matter result in additional requirements for Wolf Creek, these will be documented in a future supplement to the SER.

17 QUALITY ASSURANCE

As part of an NRC contract with the Idaho National Engineering Laboratory a draft report was prepared by EG&G which identified nuclear plant structures, systems, and components that are important to safety and ranked these items into three categories in accordance with their importance to safety. In addition, the report developed graded quality assurance guidelines which were applicable to each of the three categories for both the construction and operational phases. The results discussed in the EG&G document provided information to the staff on one approach for developing guidance on General Design Criterion 1 of Appendix A of 10 CFR Part 50.

Although this work is an attempt to classify plant items "important to safety" and suggest commensurate QA requirements, it is not clear that this is the appropriate level of detail for NRC review of GDC 1. The staff is continuing to review this information. At this time the staff concludes that the information and conclusions are not readily implementable in the NRC review. Therefore, this report has no significance in the staff's review of the Wolf Creek application.

22 TMI-2 REQUIREMENTS

II.D.1 Performance Testing of Boiling Water Reactor and Pressurized Water Reactor Relief and Safety Valves

As required by NUREG-0737, "Clarification of TMI Action Plan Requirements", Item II.D.1, all PWR plant licensees and applicants are required to demonstrate that their pressurizer safety valves (SV), power operated relief valves (PORVs), PORV block valves, and all associated discharge piping will function adequately under conditions predicted for design basis transients and accidents. In response to this requirement, the Electric Power Research Institute (EPRI), on behalf of the PWR Owners Group, has completed a full scale valve testing program and the Owners Group has submitted these test results to the staff. Additionally, each PWR plant applicant for an OL was required to submit a report by fuel load which would demonstrate the operability of these valves and the associated piping.

The applicant responded to this requirement with a submittal that contains information from the EPRI valve test program results which apply to Wolf Creek (Petrick; July 1, 1982). The applicant has also responded with a submittal which states that the safety and relief valve discharge piping and supports have been verified to insure functionability and to have no adverse affect on valve operability (Petrick; January 7, 1983).

The staff has not completed a detail review of the applicant's submittals; however, bassed on a preliminary review we find that the general approach of using the EPRI test results to demonstrate operability of the safety valves, PORVs and PORV block valves is acceptable. The applicant's submittal notes that SNUPPS utilizes safety valves, PORVs and PORV block valves of the same size and model, that performed satisfactorily for test sequences considered representative or that bound conditions to which the SNUPPS valves could be exposed.

In summary, based on preliminary review, we have concluded that the applicant's general approach to responding to this TMI item is acceptable and provides adequate assurance that the SNUPPS Reactor Coolant System Overpressure Protection Systems can adequately perform their intended functions for the period during which we complete our detailed review. If the completion of our detailed review reveals that modifications or adjustments to safety valves, PORVs, PORV block valves, or associated piping are needed to assure that the Overpressure Protection Systems can perform their intended functions, we will require that the applicant make appropriate modifications. We will continue to carry this issue as part of Confirmatory Item B(28) until the staff has completed its detailed review.

References

Nicholas A. Petrick (SNUPPS) letter to Harold Denton (NRC), Subject: "NUREG-0737 Item II.D.1," July 1, 1981.

Nicholas A. Petrick (SNUPPS) letter to Harold Denton (NRC), Subject: "NUREG-0737 Item II.D.1," January 7, 1983.

II.K.2.17 Potential for Voiding in the Reactor Coolant Systems During Transients

An evaluation performed by H. Etherington of the ACRS titled "Flow Blockage by Steam During Natural Circulation in PWRs" was made publicly available in accordance with a request from the Science Advisor, House Committee on Interior and Insular Affairs. The evaluation is primarily for plants with once through steam generators (B&W design), but some of the discussion relates to plants with inverted U-tube steam generators (Westinghouse and Combustion Engineering designs). The staff is in general agreement with the points identified in Mr. Etherington's evaluation; however, all of his concerns regarding the phenomena of natural circulation flow blockage have been previously identified by the staff. In addition, the staff does not believe that the evaluation results adversely impact our present position regarding reliance on natural circulation or the validity of feed and bleed cooling as a defense in depth measure. Finally, the staff requirements concerning testing and training for natural circulation and void formation in the Wolf Creek Generating Station are discussed in Section 14, "Intitial Test Program," of the SER.

II.K.3.31 Plant-Specific Calculations To Show Compliance with 10 CFR 50.46

During certain cold leg small break loss-of-coolant accident (SBLOCA) scenarios in Westinghouse and Combustion Engineering designed reactors, core level depression is expected to occur for a brief period of time. The minimum level of depression expected in the core is to the elevation of loop pump suction piping. Semiscale Test S-UT-8, a small break loss-of-coolant accident (SBLOCA) test, run at the Semiscale facility was a 5% cold leg break with 1.5% bypass flow which resulted in a brief complete uncovery of the core prior to loop seal clearing. The same behavior was seen in RELAP-5 calculations for the test and for a full scale PWR.

Level depression below the elevation of the pump suction piping before loop seal clearing is a new phenomenon that had not been seen previously. This behavior has been shown to be very sensitive to several factors including bypassflow and condensation in the U tubes. The vendor codes should be reviewed to see if they adequately model liquid storage in the U tubes. As part of the generic resolution of II.K.3.30, Westinghouse will be required to calculate the results of this test. If the codes cannot model this phenomenon, the consequences of a SBLOCA may be more severe than had previously been calculated. However, the staff does not expect this new phenomenon will be shown to result in violation of 10 CFR 50.46 limits.

APPENDIX A

Chronology of NRC Staff Radiological Safety Review of Wolf Creek

The following is an update of the Chronology through July 6, 1983.

April 8, 1983	Order Permitting Parties to Reply to Intervenor's Objection to Prehearing Conference Order issued by the ASLB.
April 5-9, 1983	NRC site audit for Power Systems Branch review. (Summary issued May 12, 1983).
April 12, 1983	Representatives from NRC, KG&E, UE, SNUPPS, and Bechtel Corporation met to discuss matters related to NUREG-0737, Supplement 1. (Summary issued April 18, 1983).
April 20, 1983	Letter to applicant requesting additional information on the SNUPPS Environmental Qualification Plan.
April 22, 1983	Letter from SNUPPS in response to Generic Letter No. 83-10C.
April 26, 1983	Letter from SNUPPS on containment recirculation sump testing.
April 29, 1983	Letter from the applicant providing additional information for the review of the Wolf Creek Emergency Plan.
April 29, 1983	Letter from the applicant transmitting Revision 10 of the Radiological Emergency Response Plan.
May 5, 1983	Order of Expedited Briefing Schedule Concerning Applicant's Objection and Motion for Adoption by the ASLB.
May 5, 1983	Memorandum and Order Ruling Upon Intervenor's Objection to Prehearing Conference Order issued by the ASLB.
May 9, 1983	Letter from SNUPPS concerning the Power Systems Branch Review.
May 9, 1983	Letter from SNUPPS concerning fillet weld requirements.
May 9, 1983	Letter from SNUPPS concerning NRC request for information on seismic qualification of equipment.
May 13, 1983	Letter from SNUPPS concerning the Power Systems Branch Review of SNUPPS.

May 18, 1983	Letter from the applicant transmitting the 1982 Annual Reports.
May 23, 1983	Letter from the applicant informing the NRC of a change in the fuel load date from October 1984 to August 1984.
May 26, 1983	Letter to the applicant on the NRC's review of the fillet weld requirements.
May 27, 1983	Letter from SNUPPS concerning the environmental qualification of safety-related electrical equipment.
May 31, 1983	Letter to the applicant requesting additional information on the SNUPPS Environmental Qualification Plan.
May 31, 1983	Letter from the applicant transmitting Revision 10 to Wolf Creek FSAR Addendum.
June 6, 1983	Letter from the applicant transmitting changes to the Wolf Creek Quality Assurance Program.
June 10, 1983	Letter from the applicant transmitting changes to the Wolf Creek Quality Assurance Program.
June 13, 1983	Letter from the applicant changing the classification of the local containment leak rate test.
June 15-17, 1983	NRC Caseload Forecast Panel Visit to the site.
June 16, 1983	Letter to the applicant transmitting 2 xerox copies of NUREG-0881, Supplement No. 2 (SSER #2).
June 20-24, 1983	NRC equipment qualification audit at the Callaway and Wolf Creek sites.
June 27, 1983	Letter to the applicant transmitting 20 printed copies of NUREG-0881, Supplement 2 (SSER #2).
June 30, 1983	Letter from the applicant discussing the Kansas Corporation Commission's order denying Kansas City Power and Light Company's application for a 345 kv transmission line from Wolf Creek to KCPL's West Gardner Substation in Johnson County, Kansas.
July 1, 1983	Letter from the applicant informing the staff of a change in the operator licensing examinations reflecting the new fuel load date.
July 6, 1983	Letter to the applicant on the NRC's review of the fillet weld requirements.

Appendix C

Nuclear Regulatory Commission
Unresolved Safety Issues

A-17 Systems Interaction in Nuclear Power Plants

An affidavit filed by an NRC staff witness on the Shoreham proceeding raised several questions regarding the progress and effort being made by the staff to address Unresolved Safety Issue A-17, "Systems Interaction in Nuclear Power Plants."

Current licensing requirements and safety review procedures used by the NRC staff are designed to address many different types of potential system interactions. Adherence to the defense-in-depth principle and our related licensing requirements, such as the single failure criterion, result in redundant, independent and physically separated safety systems, and protection of each of the redundant safety systems against events such as high energy line breaks, missiles, high winds, flooding, seismic events and fires. We believe that these licensing requirements, supplemented by the review procedures of the Standard Review Plan (which provides for interdisciplinary reviews of safety systems and takes into account known potential system interactions significant to safety) provides reasonable assurance that operation of licensed nuclear power plants will not result in undue risk to the health and safety of the public.

Furthermore, the staff's program on Unresolved Safety Issue A-17 was initiated to confirm that present review procedures and safety criteria provide an acceptable level of independence for systems required for safety by evaluating the potential for the more important undesirable interactions between and among systems. To date, the program has provided no indication that present review procedures and criteria do not provide reasonable assurance that the effects of potential systems interactions on plant safety will be within the effects of plant safety previously evaluated (i.e., within the design-basis envelop).

The allegations filed by a staff witness' affidavit to the Shoreham hearings have also been associated with his formal Differing Professional Opinion. The NRC provides each staff member the opportunity to make known his best professional judgment on matters relating to the mission of the NRC in a written statement entitled a Differing Professional Opinion. The Differing Professional Opinion has been processed in accordance with formal NRC procedures, and the Office Director considered the Differing Professional Opinion resolved July 11, 1983. The Director adopted the recommendations from an independent review of the Differing Professional Opinion which were (1) to perform a formal review of the updated plan to resolve USI A-17 by the Office Divisions and the Advisory Committee for Reactor Safeguards, and (2) to assign a dedicated Task Manager.

The Director concluded from the independent review that (1) the Differing Professional Opinion recommendation of requiring plants to perform systems interaction reviews would be premature and is not justified based upon results of the USI A-17 task to date, (2) no examples of safety significant systems interactions were identified in the Differing Professional Opinion, and (3) no deficiencies in the Standard Review Plan were identified.

The NRC staff continues to be confident that current regulatory requirements and procedures provide an adequate degree of public health and safety pending the resolution of USI A-17. In summary, the conclusion in SER Appendix C (page C-14) concerning this unresolved safety issue remains unchanged.

APPENDIX D

NRC STAFF CONTRIBUTORS AND CONSULTANTS

This supplement is a product of the NRC staff. The following staff members were principal contributors to this report.

Name	Title	Review Branch
R. Stevens	Reactor Engineer	Instrumentation and Control
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R. Eberly	Auxiliary Systems Engr.	Chemical Engr.

APPENDIX H

ERRATA TO WOLF CREEK SER SUPPLEMENT NO. 2

Page	Line	
1-2	23	Change Section 3.6.1 to Section 3.9.3.2
1-3	13	Change "no longer required" to "discussed in this supplement"
5-1	24	Change "have" to "has"
17-1	23, 24	Delete reference
17-1	32, 33	Delete reference
D-1	2	Change "CONTRIBUTIONS" to "CONTRIBUTORS"

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15. SUPPLEMENTARY NOTES

Docket STN 50-482

16. ABSTRACT (200 words or less)

Supplement No. 3 to the Safety Evaluation Report related to operation of the Wolf Creek Generating Station, Unit No. 1 updates the information contained in the Safety Evaluation Report, dated April 1982 and Supplement Nos. 1 and 2, dated August 1982 and June 1983, respectively. Supplement No. 3 contains resolutions of open issues and confirmatory items and addresses Board Notifications.

The Safety Evaluation and its supplements pertain to the application for a license to operate the Wolf Creek Generating Station, Unit No. 1 filed by Kansas Gas and Electric Company on February 19, 1980. The Construction Permit CPPR-147 was issued on May 17, 1977. The facility is located in Coffey County, Kansas.

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