NUREG-0881
Supplement No. 1

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 1982

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APPENDICES

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Nuclear Regulatory Commission's Safety Evaluation Report (NUREG-0881) for the application filed by the Kansas Gas and Electric Company, as applicant and agent for the owners, for a license to operate the Wolf Creek Generating Station, Unit 1 (Docket No. STN 50-482), was issued in April 1982. At that time, the staff identified items that were not yet 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.

At its 265th meeting on May 6, 1982, the Advisory Committee on Reactor Safeguards completed its review of the application. The Committee in its May 11, 1982 letter to Chairman Palladino of the NRC concluded that if due consideration is given to the items mentioned in its letter, and subject to the satisfactory completion of construction, staffing, training, and preoperational testing, there is reasonable assurance that the Wolf Creek Generating Station, Unit No. 1 can be operated at power levels up to 3425 thermal megawatts without undue risk to the health and safety of the public.

The purpose of this supplement to the Safety Evaluation Report (SER) is to provide the staff evaluation of the open items that have been resolved, to address changes to its safety evaluation which resulted from the receipt of additional information from the applicant, and to address those recommendations that are contained in the Advisory Committee on Reactor Safeguards letter of May 11, 1982. That letter is included as Appendix G to this supplement to the Safety Evaluation Report. The staff's response to the recommendations in the Committee's letter is given in Section 18 of this supplement to the SER.

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 operating license application for Wolf Creek is Jon B. Hopkins. Mr. Hopkins may be contacted by calling (301) 492-7144 or writing:

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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

1.7 Summary of Outstanding Items

The complete resolution of one of the outstanding items and the partial resolution of another outstanding item identified in the SER are described in this supplement. All outstanding items are listed below. The resolution of these items will be discussed in a future supplement. The staff will complete its review of these items before the operating license is issued.

A(1) Seismic and dynamic qualification of seismic Category I mechanical and electrical equipment

A(2) Environmental qualification of safety-related electrical equipment

A(3) TMI Action Plan

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

B(1) Closed

B(2) Pump and valve operability assurance program

B(3) Fire protection program - alternate shutdown panel

B(4) TMI Action Plan

- 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 applicants

- II.B.2 Plant shielding to provide access to vital areas and protect safety equipment for post-accident operation

1.8 Confirmatory Items

The following is an update of each of those confirmatory issues in Section 1.8 of the SER which have been completed. Additionally, these are 3 new confirmatory items.

A(1) UHS dam embankment material (Section 2.5.6)

A(3) Site-specific seismic structural analysis (Sections 3.7 and 3.8)

A(7) Security plan (Section 13.6)

B(1) Additional seismic instrumentation and control room indication (Section 3.7.4)

B(7) ECCS analysis (Section 6.3.5)

B(27) Reactor coolant pump locked rotor accident (Section 15.3.6)

B(29) Test of engineered safeguards P-4 interlock (Section 7.3.2.2) - new confirmatory item

B(30) Automatic indication of block of signals initiating auxiliary feedwater following trip of the main feedwater pumps (Section 7.3.2.7) - new confirmatory item

B(31) Actuation of valve component level windows on the bypassed and inoperable status panel (Section 7.5.2.2) - new confirmatory item

1.9 License Conditions

The following is an update of four license conditions in Section 1.9 of the SER which are no longer required. Additionally, there are two new license conditions.

B(5) Test of engineered safeguards P-4 interlock (Section 7.3.2.2)

B(6) Automatic indication of block of signals initiating auxiliary feedwater following trip of the main feedwater pumps (Section 7.3.2.7)

B(7) Steam generator level control and protection (Section 7.3.2.8)

B(10) Actuation of valve component level windows on the bypassed and inoperable status panel (Section 7.5.2.2)

B(18) Operations restriction above 90% of full power (Section 15.2.3.3) - new license condition

B(19) Experienced PWR operator or startup engineer required onshift for one year or until sufficient operating experience is acquired (Section 18, Item 3) - new license condition

2 SITE CHARACTERISTICS

2.5 Geology and Seismology

2.5.6 Dams

The following sections summarize the staff's review of additional geotechnical engineering informati、:[1,2] provided by the applicant after the issuance of the SER. This information addresses the staff's concerns stated in SER Section 2.5.6. The item of concern was the dispersive characteristics of the UHS dam embankment material. The staff's evaluation of this item is in accordance with the General Design Criteria (GDC) in Appendix A to 10 CFR 100, the Standard Review Plan (SRP) (NUREG-0800, July 1981), and current licensing policy.

UHS Dam Embankment Material

The embankment material was from borrow sources within the ultimate heat sink (UHS) reservoir. The borrow material was residual soil formed as a result of weathering of the limestone and shale bedrock. Laboratory tests (SCS test and pinhole test) performed on potential borrow material during the design stage did not indicate a significant dispersive potential. However, when the same tests were repeated on soil samples from the completed UHS dam, the results indicated dispersive behavior for some of the samples. When the samples showing dispersive behavior were tested using water from John Redmond Reservoir instead of the distilled water used in the original tests (the water in the cooling lake is primarily water pumped from the John Redmond Reservoir), the samples did not show dispersive behavior. Hence, in the field where the UHS water is pumped from the John Redmond Reservoir, the UHS dam embankment material is not likely to have any dispersive potential. FSAR Table 2.5-67 gives the dispersive characteristics test data.

UHS Dam

Section 2.5.6 of the SER (NUREG-0881) describes the foundations, embankment geometry and construction details, and riprap protection of the UHS dam. Relevant important items are: (1) the dam is founded on the bedrock; (2) the bedrock is horizontally bedded limestone and shale; (3) the impervious bedrock is free from open joints and cracks; (4) the clay dam has 4H:1V side slopes on both sides; and (5) the entire exposed surface (upstream, top, and downstream surface) of the dam is protected by 4-ft thick stone riprap underlain by two 18-in. thick layers of graded sand filter. The important feature is that the filter is designed to prevent migration of clay particles from the embankment into the stone riprap.

Even if the embankment material were to be of a dispersive nature, the dam design and construction (such as the excavation to impervious bedrock over the whole foundation area, 4H:1V side slopes, and sand filter under the riprap) precludes the likelihood of piping failures as a result of erosion of dispersive clay. Hence, the dam, as designed and constructed, is considered safe, even if the embankment material were to be a dispersive clay.

UHS Dam Filling Test

Although the UHS dam is designed to be safe against failure by erosion of dispersive clay, the dam was tested by filling the UHS reservoir only and monitoring the downstream slope for 30 days.[1,2] It was filled and maintained with water from the John Redmond Reservoir to an elevation of 1069.0 to 1069.5 ft for the 30-day observation period. The pumping operation required to fill and maintain the water elevation in the UHS basin (elevation 1069.0 to 1069.5 ft) and downstream (elevation 1055.0 ft) of the dam were monitored. Nine monuments were established on the UHS dam and their vertical and horizontal movements were monitored during the initial filling and the following 30-day observation period. The downstream slope was visually inspected for signs of distress of slope or piping erosion of embankment material. The downstream flow was observed to determine if it were carrying eroded soil particles. The net quantity of water pumped from the downstream side to maintain a water elevation of 1055.0 ft was 347,400 gal for 30 days or 0.154×10^{-4} cfs/lineal ft of dam. This measured seepage compares favorably with the estimated seepage of 0.23×10^{-4} csf/lineal ft of the dam. The movements of the monuments during the initial filling and observation period were a maximum of 0.5 in. vertical and 1.0 in. horizontal. The visual inspection of the downstream slope did not reveal any signs of geometrical distress of the slope or signs of piping or erosion. The water pumped from the downstream side was clear and did not show any soil particles being transported with it.

If there were any short-term problem (erosion of dispersive material) with this dam, the eroded material would have surfaced through the downstream slope and resulted in noticeable seepage, and possibly the loss of part of the dam section. Hence, considering (1) the order of magnitude of the seepage, (2) the measured deformation of the dam during the test, and (3) the successful completion of the test, it can be concluded that the UHS dam is safe for at least 30 days' service against piping failure as a result of internal erosion of dispersive material within the embankment.

Monitoring Program

The applicant will be required to monitor the movement of the dam using nine monuments established on the UHS dam. This will be a part of the Technical Specifications and the details of this monitoring program will be reviewed by the staff when the applicant dockets the Technical Specifications.

Conclusions

Section 2.5.6 of the SER (NUREG-0881) presents the safety evaluation of the UHS dam that, along with the evaluation presented in this supplement, completes the staff safety evaluation of the UHS dam. The staff concludes that the information, including analyses and substantiation presented by the applicant, is sufficient to demonstrate that the safety-related ultimate heat sink dam is stable, will remain functional under both static and dynamic safe shutdown earthquake (SSE) loading conditions, and meets the requirements of 10 CFR 50, Appendix A.

In addition, the applicant's investigation and analyses meet the provisions of Regulatory Guide 1.27 with respect to geotechnical engineering. Therefore, confirmatory item A(1) is resolved.

References

1. Dames and Moore, "Final Report, Surveillance of Earthwork, UHS and UHS Dam," Wolf Creek Generating Station Unit No. 1, Report for Kansas Gas and Electric Company, August 1981.

2. Letter to H. Denton of NRC from G. Koester of KG&E, dated February 4, 1982, Subject: Response to geotechnical engineering questions.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Determination of Break Locations and Dynamic Effects

In Section 1.7 of the Wolf Creek SER (NUREG-0881) dated April 1982, the staff identified an outstanding item regarding the high energy pipe break hazards analysis (Section 3.6.1). When the SER was issued, much of the information in this section was either preliminary or incomplete. The methods and criteria discussed in the FSAR were found to be acceptable, but the final analyses had not been performed for the determination of postulated pipe break locations or for jet impingement effects inside containment.

In Revision 9 to the FSAR, the applicant has provided an update to Section 3.6.2 of the SNUPPS FSAR. Table 3.6-3, which summarizes the piping stresses and usage factors used to postulate high-energy break types and locations, and Table 3.6-4, which summarizes the high-energy pipe break effects analyses results, have both been updated and completed. The information contained in these tables and the piping isometric of Figure 3.6-1 showing pipe break locations, have been reviewed. The information provided is in agreement with the applicant's methods and procedures which were previously found to be acceptable. Therefore, based on its review of the applicant's submittal, the staff finds that the applicant complies with Standard Review Plan Section 3.6.2 and satisfies the applicable portions of General Design Criterion 4. The staff considers outstanding item B(1) to be closed.

3.7 Seismic Design

See Section 3.8

3.7.4 Seismic Instrumentation Program

In Revision 8 to the SNUPPS FSAR, the applicant has committed to providing a discrete response spectrum recorder at the containment foundation with the capability of providing immediate control room indication. This resolves confirmatory item B(1).

3.8 Design of Seismic Category I Structures

The Wolf Creek site-specific seismic Category I structures which were not designed for SNUPPS enveloping seismic loads were analyzed for SSE site-specific design spectra established in accordance with Regulatory Guide 1.60 anchored at 0.12g zero period acceleration (ZPA). However, on the basis of a study conducted for the NRC staff by Lawrence Livermore Laboratories, it was found that more appropriate SSE site-specific design response spectra for the Wolf Creek site should be represented by Regulatory Guide 1.60 response spectra anchored at 0.15g ZPA.

In view of this finding the applicant was requested to re-evaluate the site-specific seismic Category I structures on the basis of site-specific design response spectra anchored at 0.15g ZPA. In the reevaluation the applicant assumed structures to be fixed at the base for the analysis in the horizontal directions. For the analysis in the vertical direction, the applicant found it more conservative to use the original FLUSH analysis results adjusted linearly upward by 25% to reflect the rise from 0.12g to 0.15g.

The results of the reevaluation, as reported in applicant's letter KMLNRC 82-192 to NRC dated May 3, 1982, indicate that the stresses in the site-specific structures remain within allowable limits imposed in the original design.

On the basis of its review of the information related to structures as provided by the applicant, the staff concurs with this conclusion.

The results of the reevaluation, as reported in applicant's letter KMCNRC 82-229 to NRC dated August 5, 1982, indicate that the stresses in the Essential Service Water System Nuclear Class 3 piping including supports are acceptable.

On the basis of its review, the staff concurs with this conclusion.

This resolves confirmatory item A(3).

6 ENGINEERED SAFETY FEATURES

6.3 Emergency Core Cooling System

6.3.5 Perfrrmance Evaluation

In Revision 9 to the SNUPPS FSAR, the applicant has documented the analysis result forwarded in the letter from N. Patrick (SNUPPS) to H. Denton (NRC) dated January 7, 1982. This resolves confirmatory item B(7).

7 INSTRUMENTATION AND CONTROLS

7.3 Engineered Safety Features Actuation System

7.3.2 Resolution of Issues

7.3.2.2 Test of Engineered Safeguards P-4 Interlock

In Revision 8 to the SNUPPS FSAR, the applicant has provided information on the testing of P-4 interlocks. This FSAR information provides an adequate commitment to license condition B(5). However, this will be carried as confirmatory item B(29) until the applicant has formally notified the staff of completion of installation of this design.

7.3.2.7 Automatic Indication of Block of Signals Initiating Auxiliary Feedwater Following Trip of the Main Feedwater Pumps

In Revision 8 to the SNUPPS FSAR, the applicant has committed to provide automatic indication of the block of the signals which initiate auxiliary feedwater on loss of both main feedwater pumps on the bypassed and inoperable status panel. This FSAR information provides an adequate commitment to license condition B(6). However, until the applicant formally notifies the staff that this design has been implemented, this will be carried as confirmatory item B(30).

7.3.2.8 Steam Generator Level Control and Protection

In Revision 9 to the SNUPPS FSAR, the applicant has provided information pertaining to modification of the ESFAS logic design such that a two-out-of-four high steam generator level signal will isolate main feedwater flow. This FSAR information provides an adequate commitment to license condition B(7). 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(8).

7.5 Information Systems Important to Safety

7.5.2 Resolution of Issues

7.5.2.2 Actuation of Valve Component Level Windows on the Bypassed and Inoperable Status Panel

In Revision 8 to the SNUPPS FSAR, the applicant has provided information pertaining to the bypass indication occurring when a valve leaves the required position. This FSAR information provides an adequate commitment to license condition B(10). However, this item will be carried as confirmatory item B(31) until the applicant has formally notified the staff that this design has been implemented.

13 CONDUCT OF OPERATIONS

13.6 Industrial Security

The applicant has submitted security plans entitled "Wolf Creek Generating Station Physical Security Plan," "Wolf Creek Generating Station Safeguards Contingency Plan," and "Wolf Creek Generating Station Security Training and Qualifications Plan," for protection against radiological sabotage. The plans were reviewed in accordance with Section 13.6 "Physical Security" of the July 1981 edition of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP, NUREG-0800).

As a result of the staff's evaluation, certain portions of these plans were identified as requiring additional information and upgrading to satisfy the requirements of Section 73.55 and Appendices B and C of 10 CFR 73.

The applicant filed revisions to these plans which satisfied these requirements. We conclude that the revised plans comply with the Commission's regulations contained in 10 CFR 50 and 73. This resolves confirmatory item A(7).

An ongoing review of the progress of the implementation of these plans will be performed by the staff to assure conformance with the performance requirements of 10 CFR 73.

The identification of vital areas and measures used to control access to these areas, as described in the plan, may be subject to amendments in the future.

The applicant's security plans are being protected from unauthorized disclosure in accordance with Section 73.21 of 10 CFR 73.

15 ACCIDENT ANALYSIS

15.2 Moderate Frequency Transients

15.2.3 Increased Core Reactivity Transients

15.2.3.3 Rod Cluster Control Assembly Malfunctions

In the event of a dropped rod cluster control assembly, or group of assemblies, the reactor will typically scram on a neutron flux negative rate trip, and analysis indicates that thermal limits will not be exceeded for the event. However, if the rod locations are such that the reactor does not scram, the automatic controller may return the reactor to full power and with a single failure the control could result in a power overshoot. It is anticipated that a detailed analysis will show that, if this occurs, thermal limits will not be exceeded. However, that analysis has not yet been approved by the NRC staff and it is thus assumed that departure from nucleate boiling could occur. The staff has accepted an interim position for operating reactors which consists of a restriction on operations above 90% of full power such that either the reactor is in manual control or rods are required to be out greater than 215 steps. This restriction will be applied to the Wolf Creek Generating Station. With this restriction, thermal limits will not be exceeded. We will require this as a condition of the operating license.

15.3 Infrequent Transients and Postulated Accidents

15.3.6 Reactor Coolant Pump Locked Rotor Accident

The locked rotor accident was analyzed by postulating an instantaneous seizure of one reactor coolant system pump rotor. The reactor flow would decrease rapidly and a reactor trip would occur as a result of a low-flow signal. A thermal analysis of the hot rod in the core was performed and revealed a maxiumum cladding temperature of 1854°F. The peak coolant system pressure during the locked rotor accident (2630 lbs/in.^{2}) indicates that the integrity of the reactor coolant system pressure boundary will be maintained. However, the above analysis was conducted assuming offsite power is available. The staff's position, as reflected in SRP Section 15.3.3, calls for an analysis that assumes offsite power is unavailable. The staff requested the applicant to reanalyze this accident according to the above SRP section.

The applicant responded in a letter from N. Petrick, SNUPPS to H. Denton, NRC, dated February 4, 1982. In the response, based on a grid and plant circuitry analysis, the applicant assumed a 2-second delay between the reactor trip and the loss of offsite power. The applicant's analysis shows that, since the peak clad temperature and the peak pressure for this accident occur at about 2 seconds after the reactor trip, the loss of offsite power at that time does not increase the peak pressure or the peak temperature. Consequently, no additional fuel would fail due to the loss of offsite power 2 seconds after the reactor trip. The offsite doses released due to the above accident assuming the most severe single active failure (the failing open of a steam relief valve) are conservatively calculated to be within the 10 CFR 100 limits.

This resolves confirmatory item B(27).

18 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

On April 21 and 22, 1982 a subcommittee of the Advisory Committee on Reactor Safeguards met with representatives of the applicant and the NRC staff to consider the applicant's application for a license to operate the Wolf Creek Generating Station, Unit No.1. The meeting was held in Emporia, Kansas. On May 6, 1982 at its 265th meeting, the full Advisory Committee on Reactor Safeguards met with representatives of the applicant and the staff to consider the application. The Committee identified a number of items that it believed should be considered by the applicant and the staff and stated that if due consideration is given to these items, and subject to satisfactory completion of construction, staffing, training, and preoperational testing, there is reasonable assurance that the Wolf Creek Generating Station, Unit No. 1 can be operated at power levels up to 3425 MWt without undue risk to the health and safety of the public. The Committee's letter from P. Shewmon to Nunzio J. Palladino, dated May 11, 1982, is included as Appendix G to this supplement to the Safety Evaluation Report.

The purpose of this section is to respond to the items identified in the Committee's May 11, 1982 letter. A discussion of each of these items follows.

Item 1

The Wolf Creek Generating Station will be the first commercial nuclear power plant in the State of Kansas. The ACRS commented that it should be assured that state and local agencies are qualified to respond to possible emergency situations associated with the operation of Wolf Creek.

Response

FEMA will provide the NRC with formal findings and determinations on the adequacy of the State and local plans for those areas surrounding the Wolf Creek Generating Station. The NRC will review the FEMA findings and determinations as to whether State and local emergency plans are adequate and whether there is reasonable assurance that they can be implemented. No full power operating license will be issued unless a finding is made by NRC that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

Item 2

The Commmittee reviewed KG&E's management organization, experience, and training programs and were favorably impressed by the general competence and attitude of KG&E's personnel; however, the Committee emphasized the importance of KG&E's building a strong in-house capability for analyzing and understanding the nuclear-thermal-hydraulic behavior and systems performance of the plant.

Response

In a letter dated July 19, 1982, the applicant outlined his intentions toward addressing this ACRS concern. This includes obtaining relevant computer code models and the training of employees in the use of these models.

The staff believes that the intentions of the applicant, as outlined in their letter, are responsive to the ACRS comment.

Item 3

The applicant intends to have a technical assistant to the plant superintendent through fuel load and experienced operator consultants on-shift for a period of 1 year after startup. The ACRS commented that they believe these personnel should be retained until the operating organization has developed an experience base involving those operational duties of importance to public safety. Further, the ACRS commented that this experience base should be defined by the NRC staff in consultation with operational experts and incorporated into the regulatory requirements instead of using arbitrary operating time periods as a basis for measuring skill.

Response

The applicant has significant nuclear experience, both Navy and commercial. The technical assistant to the Plant Superintendent is a consultant with 8 years Navy nuclear power experience and 11 years of commercial nuclear power experience. He has also held an SRO license and served as Shift Supervisor for several years. He has been retained on contract through fuel load.

In a letter dated December 7, 1981, the applicant has committed to provide on each shift an experienced, previously licensed PWR operator. Four shift consultants have already been assigned and their commercial nuclear experience ranges from 10 to 13 years. These shift consultants will remain on shift for approximately the first year of operation.

The staff believes that the experience to be gained by on-shift personnel during the period from fuel load through the achievement of a nominal 100% power at the completion of startup testing will be far greater than that to be gained during an equal period of time with the plant operating at its designed level, and this experience will be adequate for the Wolf Creek staff to safely operate the plant. However, should this not be the case, the staff will assure that the applicant retains the experienced personnel on shift until the staff feels that the operating staff is sufficiently proficient.

The NRC staff will, therefore, condition the applicant's license to require an experienced formerly licensed PWR operator or PWR startup engineer on shift during startup testing for at least one year and until attainment of a nominal 100% power or until sufficient operating experience has been achieved by the operating staff.

Item 4

KG&E has proposed, as an alternative to a Shift Technical Advisor (STA), that at least one SRO on each shift have the training and background required for an STA. This approach appears to the ACRS to meet the need which originally led to the requirement of an STA. However, it is not clear to the ACRS that the level of training given to the SROs will correspond to that intended for STAs, and the ACRS recommends that the staff review this matter carefully.

Response

The staff has reviewed the applicant's STA program based on the guidance given on Shift Technical Advisors in Item I.A.1.1 of NUREG-0737, and the emergency staffing plan of NUREG-0654. Recently, SECY 82-111 was issued which updates the requirements for emergency response capability. Further, both INPO and NRC are now conducting studies aimed at determining the required number and qualifications of shift personnel. These studies could result in new rules regarding shift staffing that could change present STA guidance. This issue, therefore, remains unresolved.

Item 5

The ACRS does not have confidence that all vital aspects of the ultimate heat sink and associated systems have margins sufficient to provide an appropriate level of resistance to a lower probability, more severe earthquake (than the design basis Safe Shutdown Earthquake). The ACRS recommends therefore that the seismic margins inherent in the components of the ultimate heat sink and associated systems be investigated further and that any needed modifications be made before the plant resumes operation after the second refueling.

Response

On August 11, 1982 the staff met with the ACRS Subcommittee on Extreme External Phenomena to discuss this matter. Based on discussions held at that meeting the staff is considering the actions necessary to develop criteria, beyond that presently employed to evaluate seismic sufficiency, for seismic events of lower probability than the Safe Shutdown Earthquake (SSE) designated for a given site. The Subcommittee indicated that they would give this matter further consideration and that they would meet again with the staff in the near future.

22 TMI-2 REQUIREMENTS

I.D.1 Control Room Design Review

The Wolf Creek Safety Evaluation Report dated April 1982 stated that Kansas Gas and Electric Company (KG&E) performed a human factors evaluation on the Wolf Creek plant-specific panels (RL 013 and RL 014). The findings of the evaluation, conducted by Essex Corporation, were documented in KG&E letter to the NRC dated January 15, 1982. Subsequently, KG&E developed responses to the Essex findings and documented these in a letter to the NRC dated March 10, 1982. The staff has received further clarification of the KG&E responses during several telephone conversations with the applicant. These clarifications, along with a proposed implementation schedule for corrective actions, were documented in a KG&E letter to the NRC dated June 29, 1982.

The applicant's proposed resolutions to the documented human engineering discrepancies for the Wolf Creek site-specific panels RL 013 and RL 014 are acceptable to the staff. The implementation schedule is acceptable only if all corrective actions are completed such that they can be audited by the staff prior to issuance of the operating license.

This item remains open due to the SNUPPS portion of the control room.

APPENDIX A

CHRONOLOGY OF NRC STAFF RADIOLOGICAL SAFETY REVIEW OF WOLF CREEK

The following is an update of the chronology through July 31, 1982.

March 19, 1982	Representatives from NRC, KG&E, Bechtel, Sargent & Lundy & SNUPPS met in Bethesda, Maryland to discuss matters related to the seismic site-specific structural analysis applicable to Wolf Creek. (Summary issued March 25, 1982)
March 26, 1982	Letter from applicant providing information on seismic design margins in structures and components.
March 31, 1982	Letter from applicant concerning geology.
March 31, 1982	Letter from applicant concerning station security.
April 2, 1982	Letter from applicant concerning safety evaluation report proposed tech specs.
April 2, 1982	Letter from applicant concerning long term operability of deep draft pumps.
April 12, 1982	Letter to applicant enclosing comments on draft environmental statement.
April 12, 1982	Letter from applicant concerning NUREG-0737, Item I.D.1.
April 29, 1982	SNUPPS letter concerning NUREG-0737, Item II.D.1.
April 30, 1982	Letter from applicant concerning Revision 1 to the Security Training and Qualifications Plan, Revision 2 to the Physical Security Plan, Revision 2 to the Safeguards Contingency Plan, and copies of the revised Security Plan drawings.
April 30, 1982	Representatives from UE, KG&E, SNUPPS, Bechtel & NRC met in Bethesda, Maryland to discuss the alternate shutdown panel and related fire protection. (Summary issued June 17, 1982)
May 3, 1982	Letter from applicant transmitting the seismic design margin evaluation for the essential service water system pumphouse, safety-related manholes, valve-house and discharge structure.

May 13, 1982	Representatives from NRC, UE, KG&E & SNUPPS met in Bethesda, Maryland to discuss matters related to surveillance of Hafnium control rods for Callaway and Wolf Creek. (Summary issued May 19, 1982).
May 14, 1982	Letter from applicant transmitting the 1981 Annual Reports.
May 14, 1982	Letter to applicant transmitting the ACRS Report for Wolf Creek.
May 14, 1982	Letter from SNUPPS concerning long term operability of deep draft pumps.
May 18, 1982	Letter to SNUPPS concerning use of Subsection NB-4436, NC-4436, and ND-4436 in the Winter 1981 Addenda to ASME Section III for SNUPPS projects.
May 20, 1982	Letter from applicant transmitting comments on Wolf Creek Draft Environmental Statement (DES).
May 21, 1982	SNUPPS letter transmitting Revision 9 to SNUPPS FSAR.
May 24, 1982	Letter from applicant concerning meteorology.
May 25, 1982	Letter from SNUPPS concerning fillet weld requirements.
May 27, 1982	Letter to applicant concerning SNUPPS FSAR - request for additional information - mechanical engineering.
May 28, 1982	Letter from applicant concerning the LPZ.
June 1, 1982	Letter from applicant concerning storage of safeguards information.
June 7, 1982	SNUPPS letter concerning status of Callaway and Wolf Creek SER issues.
June 11, 1982	Letter to applicant transmitting 2 copies of FES (NUREG-0878).
June 11, 1982	Letter from applicant providing a construction progress update.
June 21, 1982	Letter to applicant transmitting 20 copies of the FES for Wolf Creek (NUREG-0878).
June 29, 1982	Letter from applicant concerning human factors evaluation of Wolf Creek site-specific control room panels, RL 013 and RL 014.

July 2, 1982	Letter to applicant concerning human factors control room design review technical evaluation report.
July 6, 1982	Letter from SNUPPS concerning Regulatory Guide 1.97.
July 6, 1982	Letter to applicant concerning control of heavy loads - NUREG-0612 - Wolf Creek.
July 8, 1982	Letter to applicant concerning request for additional information for the review of the Wolf Creek Plant, Unit 1 regarding structural engineering.
July 14, 1982	Letter from SNUPPS concerning testing of pressure isolation valves.
July 15, 1982	Letter to applicant concerning safeguards information storage at Wolf Creek.
July 19, 1982	Letter from applicant concerning the ACRS letter.
July 23, 1982	Letter from applicant concerning structural engineering.
July 30, 1982	Letter from applicant transmitting Revision 9 to the FSAR Addendum.

APPENDIX D

NRC STAFF CONTRIBUTORS AND CONSULTANTS

This Supplement No. 1 to the SER is a product of the NRC staff. The following NRC staff members were principal contributors to this report. No consultants contributed to this report.

NAME	TITLE	REVIEW BRANCH
B. Jagannath	Geotechnical Engineer	Hydrol. & Geotech.
R. Stevens	Reactor Engineer	Instr. & Control
D. Kunze	Plant Protection Analyst	Physical Security
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R. Eckenrode	Human Factors Engineer	Human Factors Eng.
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D. Terao	Mechanical Engineer	Mechanical Eng.
E. Pedersen	Management Engineer	Licensee Qual.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

APPENDIX G

May 11, 1982

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE WOLF CREEK GENERATING STATION, UNIT NO. 1

During its 265th meeting, May 6-8, 1982, the Advisory Committee on Reactor Safeguards reviewed the application of Kansas Gas and Electric Company (KG&E), Kansas City Power and Light Co. and Kansas Electric Power Cooperative, Inc. (Applicants) for a license to operate the Wolf Creek Generating Station, Unit No. 1. The Station is to be operated by KG&E. A Subcommittee meeting was held in Emporia, Kansas, on April 21-22, 1982, to consider this project. A tour of the facility was made by members of the Subcommittee on April 21, 1982. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicants, Westinghouse Electric Corporation, Bechtel Power Corporation, the Nuclear Regulatory Commission (NRC) Staff, and with members of the public. The Committee also had the benefit of the documents listed below. The Committee commented on the construction permit application for this plant in its report dated October 16, 1975.

The Wolf Creek Generating Station is located in Hampdon Township, Coffey County, Kansas. The site is in eastern Kansas approximately 53 miles south of Topeka, and 100 miles east-northeast of Wichita. The nearest population center is Emporia, Kansas, 28 miles west-northwest of the site (estimated 1980 population of 25,019).

The Wolf Creek Generating Station will be the first commercial nuclear power plant in the state of Kansas. It should be assured that state and local agencies are qualified to respond to possible emergency situations associated with the operation of the Wolf Creek Generating Station.

The Station will use a Westinghouse, four-loop, pressurized water, nuclear steam supply system having a rated power level of 3425 MWt. Unit 1 employs a cylindrical, steel-lined, reinforced, post-tensioned concrete containment structure with a free volume of 2.5 million cubic feet. The Wolf Creek Generating Station uses the Standardized Nuclear Unit Power Plant System (SNUPPS) design. It is one of two plants built to this design. The Committee reported on the operating license application of the other plant (Callaway Plant Unit No. 1) in its November 17, 1981 report to you.

The Wolf Creek Generating Station is the first nuclear power plant to be operated by KG&E. The Committee reviewed KG&E's management organization, experience, and training programs. We were favorably impressed by the general competence and attitude of KG&E's personnel. Nevertheless, we wish to emphasize the importance of KG&E's building a strong in-house capability for analyzing and understanding the nuclear-thermal-hydraulic behavior and systems performance of this plant.

To strengthen the shift structure during the initial period of operation, KG&E plans to augment each shift with a consultant who is an experienced, previously licensed PWR operator. These consultants will serve for a period of one year after startup. In addition, KG&E has retained the services of a consultant with considerable commercial nuclear experience to act as a technical assistant to the Plant Superintendent through the initial loading of fuel. We believe the technical assistant to the Plant Superintendent and the "experienced operator consultants" should be retained until the operating organization has developed an experience base involving those operational duties of importance to public safety. This experience base should be defined by the NRC Staff in consultation with operational experts and incorporated into the regulatory requirements instead of using arbitrary operating time periods as a basis for measuring skill. We encourage the practice of assigning the Senior Reactor Operator (SRO) candidates to extended tours of service at operating nuclear power plants, and recommend that others in the operations staff participate in such a program to the extent practical.

KG&E has proposed, as an alternative to a Shift Technical Advisor (STA), that at least one SRO on each shift have the training and background required for an STA. This approach appears to us to meet the need which originally led to the requirement of an STA. However, it is not clear that the level of training given to the SROs will correspond to that intended for STAs, and we recommend that the Staff review this matter carefully.

The site-specific portions of the plant, including vital aspects of the ultimate heat sink and associated systems, were designed for a 0.12 g earthquake, and are being reanalyzed for an earthquake represented by site-specific response spectra that are encompassed by Regulatory Guide 1.60 spectra anchored at a zero-period acceleration of 0.15 g. The standard portion of the plant, on the other hand, was designed for a 0.20 g earthquake with the usual margins of safety and thus would be expected to withstand a considerably larger earthquake without failing in such a manner as to cause a severe accident.

We do not have confidence that all vital aspects of the ultimate heat sink and associated systems have margins sufficient to provide an appropriate level of resistance to a lower probability, more severe earthquake. We recommend therefore that the seismic margins inherent in the components of the ultimate heat sink and associated systems be investigated further and that any needed modifications be made before the plant resumes operation after the second refueling.

Other issues have been identified as Outstanding Issues, License Conditions, and Confirmatory Issues in the Staff's Safety Evaluation Report dated April 1982; these include some TMI Action Plan requirements. Except as noted above, we believe these issues can be resolved in a manner satisfactory to the NRC Staff and recommend that this be done.

We believe that, if due consideration is given to the recommendations above, and subject to satisfactory completion of construction, staffing, training, and preoperational testing, there is reasonable assurance that the Wolf Creek Generating Station, Unit No. 1 can be operated at power levels up to 3425 MWt without undue risk to the health and safety of the public.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References:

1. "Final Safety Analysis Report for Standardized Nuclear Unit Power Plant System," with Revisions 1-8.
2. "Final Safety Analysis Report, Wolf Creek Generating Station Unit No. 1," with Revisions 1-8.
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Wolf Creek Generating Station, Unit No. 1," NUREG-0881, dated April 1982.
4. Written statement by John M. Simpson, Attorney for Intervenors, Re: Emergency Planning Procedures and Plans - Wolf Creek Plant, dated April 22, 1982.

APPENDIX H
ERRATA TO WOLF CREEK SAFETY EVALUATION REPORT

Page	Line	
1-6	15	Change "Quadrex Corporation" to "Phoenix Power Services"
2-11	4	Change "32.8°C (27°F)" to "-32.8°C (-27°F)"
2-14	43	Change "410 acre-ft" to "442 acre-ft"
2-26	7	Insert the following paragraph in place of the sentence.

(3) The applicant's SSE for the Wolf Creek site is a peak horizontal acceleration of 0.12g for those seismic Category I structures outside the standard plant portion of the facility. The SSE value for the standard (SNUPPS) portion of the Wolf Creek facility is 0.20g. The Operating Basis Earthquake (OBE) acceleration values are 0.06g for the nonstandard portion of the facility and 0.12g for the standard portion. These acceleration values are used as high frequency inputs to Regulatory Guide (RG) 1.60 response spectra. Current staff practice has been to request the applicant to calculate appropriately derived site-specific response spectra from accelerograms of similar controlling earthquake size and epicentral distance and local site conditions. The staff seismology consultants (LLNL) have made independent estimates of site-specific spectra and seismic hazard for the Wolf Creek site. It is the staff's position that the 84th percentile spectrum represents an appropriately conservative representation of the site-specific earthquake. The 84th percentile site-specific spectrum calculated by LLNL for a magnitude 5.25 local earthquake exceeds the 0.12g RG 1.60 balance-of-plant SSE spectrum above about 3 hz (see Figure 2.6). The staff finds the LLNL 84th percentile spectrum is appropriate for describing ground motion to be used in evaluating the effects of the maximum local event (magnitude 5.25). LLNL found and the staff agrees that a 0.12g RG 1.60 spectrum is, however, appropriate for describing ground motion to be used in evaluating the effects at the site of the maximum event associated with the Nemaha Uplift (magnitude 5.75). Site-specific spectra calculated by LLNL do not exceed the SSE for the SNUPPS portion of the facility.

Page	Line	
8-2	20	Change "loss of both units themselves" to "loss of the unit".

8-3	12	Change "battery charger alarm" to "345 kV battery trouble alarm" and change "battery voltage alarm" to "69 kV general trouble alarm"
9-6	2	Change first sentence to read: "...consists of two independent subsystems, one for each diesel generator room."
9-8	16	Change "supervisor" to "supervision" and change "start" to "strict"
10-3	16	Change "cooling lake screenhouse" to "circulating water screenhouse"
11-7	--	In Table 11.5, change the capacity of the primary spent resin storage tank from "350 gal" to "350 ft^3"
13-4	35	Delete "and inservice"
13-17	12	Change "Both of these latter individuals" to "The Operations Coordinator"
13-17	--	In Section 13.1.3.4, delete the second paragraph.
13-19	22	Change "160" to "60"
13-19	33	Change "five of seven" to "four of seven"
13-20	--	In the fifth line of the first paragraph change "licensed" to "certified"
13-20	--	In the last line on the page, change "three operators and two security persons" to "a minimum of five persons, no more than two of which may be security personnel"
13-29	7	Change "six" to "five"

NRC FORM 335
(7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)*
NUREG-0881
Supplement No. 1

2. *(Leave blank)*

3. RECIPIENT'S ACCESSION NO.

4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*
Safety Evaluation Report Related to the operation of Wolf Creek Generating Station, Unit No. 1

5. DATE REPORT COMPLETED

MONTH	YEAR
AUGUST	1982

DATE REPORT ISSUED

MONTH	YEAR
AUGUST	1982

6. *(Leave blank)*

7. AUTHOR(S)

8. *(Leave blank)*

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*
Same as 9 above

13. TYPE OF REPORT
Technical

PERIOD COVERED *(Inclusive dates)*
April 1982 - August 1982

14. *(Leave blank)*

15. SUPPLEMENTARY NOTES
Docket No. STN 50-482

16. ABSTRACT *(200 words or less)*

Supplement No. 1 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. This supplement also addresses the ACRS Report issued May 11, 1982.

The Safety Evaluation Report and its supplement pertain to the application for a license to operate the Wolf Creek Generating Station, Unit No. 1 filed by the 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.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
UNLIMITED

19. SECURITY CLASS *(This report)*
UNCLASSIFIED

20. SECURITY CLASS *(This page)*
UNCLASSIFIED

21. NO. OF PAGES

22. PRICE
$

NRC FORM 335 (7-77)

FIRST CLASS MAIL
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PERMIT No G67

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