NUREG-0881
Supplement No. 5

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

March 1985

UNITED STATES NUCLEAR REGULATORY COMMISSION

— NOTICE —

THE ATTACHED FILES ARE OFFICIAL RECORDS OF THE DIVISION OF DOCUMENT CONTROL. THEY HAVE BEEN CHARGED TO YOU FOR A LIMITED TIME PERIOD AND MUST BE RETURNED TO THE RECORDS FACILITY BRANCH 016. PLEASE DO NOT SEND DOCUMENTS CHARGED OUT THROUGH THE MAIL. REMOVAL OF ANY PAGE(S) FROM DOCUMENT FOR REPRODUCTION MUST BE REFERRED TO FILE PERSONNEL.

DEADLINE RETURN DATE 50-482

3/85

RECORDS FACILITY BRANCH

8504030055

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission,
   Washington, DC 20555
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations,* and *Nuclear Regulatory Commission Issuances.*

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

GPO Printed copy price: $6.50

NUREG-0881
Supplement No. 5

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

March 1985

ABSTRACT

This report supplements the Safety Evaluation Report (SER) 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). The facility is located in Coffey County, Kansas. This supplement has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission and provides recent information regarding resolution of the open items identified in the SER. Because of the favorable resolution of the items discussed in this report, the staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

APPENDICES

LIST OF FIGURES

LIST OF TABLES

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 (OL) for the Wolf Creek Generating Station, Unit 1 (Docket No. STN 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 OLs 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 (10 CFR 50). The other SNUPPS OL application submitted for review was that submitted by Union Electric Company (UE) for the Callaway Plant (Docket No. STN 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. Supplement 1 to the SER (SSER 1) was issued in August 1982, Supplement 2 (SSER 2) was issued in June 1983, Supplement 3 (SSER 3) was issued in August 1983, and Supplement 4 (SSER 4) was issued in December 1983. These documents contained a number of items that were not resolved with the applicant. These items were categorized as

(1) Outstanding items that needed to be resolved before the issuance of an operating license.

(2) Items for which the staff had completed its review and had determined positions about which there appeared to be no significant disagreement between the applicant and the staff. However, further information was needed 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 OL. These would be conditions to the OL.

The purpose of this fifth supplement (SSER 5) is to provide the staff evaluation of the items that have been resolved and to address changes to the SER that resulted from the receipt of additional information. Each of the following sections of this supplement is numbered the same as the section of the SER and its supplements that is being updated and, unless otherwise noted, the discussions are supplementary to and not in lieu of the previous discussions. Appendix A to this supplement is a continuation of the chronology. Appendix B, References, lists material used in preparing this supplement. Appendix C continues the discussion of how the Unresolved Safety Issues relate to the Wolf Creek application. Appendix D is a list of principal contributors to this supplement. Appendix J is a preservice inspection relief request evaluation prepared with the techincal assistance of Department of Energy contractors from the Idaho National Engineering Laboratory. Appendices K and L are two reports prepared by EG&G Idaho, Inc., on the control of heavy loads, indicating Wolf Creek's degree of compliance with the guidelines of NUREG-0612. Appendix M is the staff's evaluation of the materials properties data of Westinghouse Report WCAP-10456.

Copies of this supplement are available for inspection at the NRC Public Document Room, 1717 H Street N.W., Washington, D.C., at the William Allen White Library, Emporia State University; and at the Washburn University School of Law Library, Topeka, 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. Paul W. O'Connor. Mr. O'Connor may be contacted by calling (301) 492-4708 or writing

Paul W. O'Connor
U.S. Nuclear Regulatory Commission
Division of Licensing
Washington, D.C. 20555

1.7 Summary of Outstanding Items

Listed below is an update of all of the outstanding items that require resolution before the operating license is issued. The status of each of these items is given.

Part A*

A(1) Seismic and dynamic qualification of seismic Category I mechanical and electrical equipment** (closed in SSER 5)

A(2) Environmental qualification of safety-related electrical equipment** (closed in SSER 5)

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

- I.A.1.1 Shift Technical Advisor (closed in SSER 5)
- I.D.1 Control room design review (closed in SSER 5)
- III.A.1.2 Upgrade emergency support facilities (removed in SSER 4)

A(4) Onsite emergency preparedness (removed in SSER 4)

Part B*

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

B(2) Pump and valve operability assurance program (closed in SSER 5)

*Part A lists the site-specific items; Part B contains the SNUPPS items that are common to both Wolf Creek and its sister plant Callaway.
**This item includes both plant-specific and duplicate-plant information.

B(3) Fire protection program - alternate shutdown panel (closed in SSER 3)

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

I.C.1 Guidance for evaluation and development of procedures for transients and accidents (closed in SSER 5)

I.C.8 Pilot monitoring of selected emergency procedures for near-term operating license applications (closed in SSER 5)

II.B.2 Plant shielding to provide access to vital areas and protect safety equipment for postaccident operation (closed in 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 revisions to the Final Safety Analysis Report, additional information provided via letters, and the submittal of the draft Technical Specifications, several confirmatory items have been resolved. These items are noted below.

Part A*

A(1) UHS dam dispersiveness (closed in SSER 1)

A(2) Main dam seepage (closed in SSER 5)

A(3) Site-specific seismic structural analysis (closed in SSER 1)

A(4) Identification of base metal and heat-affected zone surveillance material (closed in SSER 2)

A(5) Pressure-temperature limits (closed in SSER 2)

A(6) Fire protection site visit (closed in SSER 5)

A(7) Security plan (closed in SSER 1)

A(8) TMI Action Plan (closed in SSER 4)

II.K.1 IE bulletin on measures to mitigate small break LOCAs and loss-of-feedwater accidents (closed in SSER 4)

III.A.2 Improving licensee emergency preparedness--long-term (closed in SSER 4)

A(9) Onsite emergency preparedness (closed in SSER 5)

*Part A lists the site-specific items.

Part B*

B(1) Additional seismic instrumentation and control room indication (closed in SSER 1)

B(2) Analysis of steam generator tube plugging (closed in SSER 4)

B(3) Testing of pressure isolation valves (closed in SSER 2)

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

B(5) Preservice inspection testing program (closed in SSER 5)

B(6) Steam generator inservice inspection (closed in SSER 4)

B(7) ECCS analysis (closed in SSER 1)

B(8) Steam generator level control and protection (closed in SSER 4)

B(9) Capability for safe shutdown following loss of a bus supplying power to instruments and controls (closed in SSER 5)

B(10) Operator actions required to maintain safe shutdown from outside control room (closed in SSER 5)

B(11) Reactor coolant temperature indicators on the auxiliary shutdown panel (closed in SSER 5)

B(12) Volume control tank level control and protection interaction (closed in SSER 4)

B(13) Boron dilution control (closed in SSER 5)

B(14) Environmental qualification of control systems (closed in SSER 5)

B(15) Circuitry for automatic transfer of diesel generator from test to auto control mode (closed in SSER 3)

B(16) Diesel generator reliability qualification testing (closed in SSER 3)

B(17) Circuitry for bypass of protective circuitry (closed in SSER 3)

B(18) Circuitry for inservice testing per Regulatory Guide 1.108 (closed in SSER 3)

B(19) Low and or degraded grid voltage (closed in SSER 5)

B(20) Use of regulatory-type transformer as isolation device (closed in SSER 3)

B(21) Isolation of control room and remote circuits (closed in SSER 5)

*Part B contains the SNUPPS items that are common to both Wolf Creek and its sister plant Callaway.

B(22) Sequencing of loads on the offsite power system (closed in SSER 5)

B(23) Submerged electrical equipment (closed in SSER 3)

B(24) Separation between redundant safety-related cables inside control panels (closed in SSER 3)

B(25) Compliance with position 1 of Regulatory Guide 1.63 (closed in SSER 5)

B(26) Monitoring of rocker arm lube oil system temperature for diesel generators (closed in SSER 4)

B(27) Reactor coolant pump locked rotor accident (closed in SSER 1)

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

- II.D.1 Performance testing of BWR and PWR relief and safety valves (closed in SSER 5)
- II.E.1.1 Recommendation GS-2, physical locking of isolation valve. (closed in SSER 4)
- II.E.4.2 Containment isolation dependability (closed in SSER 4)
- II.F.1 Additional accident monitoring instrumentation, Attachments 1, 2, and 3 (Attachment 3, closed in SSER 2; Attachments 1 and 2, closed in SSER 4)
- II.K.2.13 Thermal mechanical report--effect of high-pressure injection on vessel integrity for small-break LOCA with no auxiliary feedwater (closed in SSER 3)
- II.K.3.2 Report on overall safety effect of PORV isolation system (closed in SSER 2)
- II.K.3.11 Justification of use of certain PORVs (closed in SSER 4)
- III.D.1.1 Integrity of systems outside containment likely to contain radioactive material (closed in SSER 5)

B(29) Test of engineered safeguards P-4 interlock (closed in SSER 4)

B(30) Automatic indication of block of signals initiating auxiliary feedwater following trip of main feedwater pumps (closed in SSER 4)

B(31) Actuation of valve component level windows on the bypassed and inoperable status panel (closed in SSER 4)

B(32) Postaccident monitoring (closed in SSER 5)

B(33) Indicators, alarms, and test features provided for instrumentation used for safety functions (closed in SSER 4)

B(34) Interlocks for reactor coolant system pressure control during low-temperature operative (closed in SSER 4)

B(35) Capacity and capability of offsite circuits (closed in SSER 5)

1.9 License Conditions

The following is an update of each of the license conditions described in Section 1.9 of the SER. License Condition B(18) has been removed based on a review of Revision 12 to the SNUPPS FSAR.

Part A*

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

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 (removed in SSER 5)

B(4) Sensor time response testing (removed in SSER 5)

B(5) Tests of engineered safeguards P-4 interlocks (removed in SSER 1)

B(6) Automatic indication of block of signals initiating auxiliary feedwater following trip of the main feedwater pumps (removed in SSER 1)

B(7) Steam generator level control and protection (removed in SSER 1)

B(8) Indicator, alarms, and test features provided for instrumentation used for safety functions (removed in SSER 3)

B(9) Reactor coolant temperature indications on the auxiliary shutdown panel (removed in SSER 3)

B(10) Actuation of valve component level windows on the bypassed and inoperable status panel (removed in SSER 1)

B(11) Postaccident monitoring (removed in SSER 2)

---

*Part A lists the site-specific items; Part B contains the SNUPPS items that are common to both Wolf Creek and its sister plant Callaway.
**This item includes both plant-specific and duplicate-plant information.

B(12) Interlocks for reactor coolant system (RCS) pressure control during low-temperature operation (removed in SSER 3)

B(13) Volume control tank level control and protection interaction (removed in SSER 3)

B(14) Boron dilution control (removed in SSER 3)

B(15) Bypass of protective trips on diesel generator (removed in SSER 3)

B(16) Installation of battery discharge alarm (removed in SSER 3)

B(17) TMI Action Plan (SER and SSER 4 Section 22)

II.B.3 Postaccident sampling capability (removed in SSER 5)

B(18) Operation restriction above 90% of full power (removed in SSER 4)

B(19) Experienced PWR operator or startup engineer required onshift for one year or until sufficient operating experience is acquired (closed in SSER 5, Section 13.1.2.3)

1.7 Nuclear Waste Policy Act of 1982

Section 302(b) of the Nuclear Waste Policy Act of 1982 states that NRC shall not issue or renew a license for a nuclear power reactor unless the utility has signed a contract with the Department of Energy for disposal services. The Kansas Gas and Electric Company has signed a contractual agreement with the Department of Energy dated October 10, 1984. This agreement is applicable to Wolf Creek Generating Station, Unit 1.

2 SITE CHARACTERISTICS

2.5 Geology and Seismology

2.5.6 Dams

2.5.6.7 Seepage

Main Dam Seepage

The main dam of the cooling lake is the major water-retaining structure for the Wolf Creek Generating Station. This earthen dam is not part of the ultimate heat sink; therefore, it is not a seismic Category I structure. Failure of this main dam, however, would present a challenge to the ultimate heat sink reservoir. During a site visit, the staff noticed minor seepage at the toe of the downstream slope of the main dam and requested the applicant to evaluate the cause and significance of this seepage. The staff's evaluation of the safety of the main dam is presented in SER Section 2.5.6.

In response to the staff request, the applicant conducted a detailed study of this seepage (letters dated Nov. 28, 1983, and March 16, 1984), which included

- identifying the location of seepage
- constructing a weir to measure the seepage flow
- recording precipitation and lake level data
- estimating the expected seepage through the dam
- analyzing the data

The seepage was exiting from the gravel toe drain of the main dam at station 58 + 50 (see SER Figure 2.10 for cross section). Review of the longitudinal profile of the main dam foundation indicated that seepage from approximately 3000 ft of the main dam, from station 58 + 50 to station 88 + 85, drained to this location. The estimated (calculated) seepage through 3000 ft of main dam is 0.018 cfs. The applicant constructed a weir to measure the flow at this location. The flow measured at the weir from September 1983 to February 1984 ranged from 0.003 cfs to a maximum of 0.35 cfs. During this period, the lake level was at elevation 1087.0 ft (normal pool elevation) and the daily precipitation ranged from 0 to a maximum of 1.5 in. Figure 241.15-5 included in the applicant's response to Confirmatory Item A(2) (March 16, 1984) shows a plot of daily precipitation, lake level, and seepage flow from September 1983 to February 1984. The measured flow shows a trend of steady-state flow combined with intermittent peak flows. These data indicate that the peak flows correspond with the precipitation in the area; increased flow is the surface runoff from the slope of the dam draining to this location. The flow measured (0.003 cfs) in the absence of precipitation was the seepage through the dam, which was less than the estimated seepage through 3000 ft of the main dam (0.018 cfs). The applicant concluded that, because the seepage through the dam was less than that estimated in the design of this dam, the seepage does not pose any threat to the safety of this dam. The staff concurs with the applicant's reasoning and conclusions.

The applicant has committed (April 30, 1984) to continue to monitor the seepage flow for 1 year to cover one annual weather cycle, and to inspect the main dam in accordance with the provisions of Regulatory Guide (RG) 1.127; this is satisfactory to the staff. On the basis of its findings, the staff considers Confirmatory Item A(2) resolved.

Conclusions

The evaluation presented in this supplement and in SER Section 2.5.6 completes the staff safety evaluation of the main dam. The staff concludes that the information, including analysis and substantiation presented by the applicant, is sufficient to demonstrate that the main dam will remain functional under both static and dynamic (operating basis earthquake) loading conditions, and meets these requirements in accord with 10 CFR 50, Appendix A. On the basis of its evaluation the staff considers Confirmatory Item A(2) now resolved.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.6 Protection Against Effects Associated With the Postulated Rupture of Piping

3.6.1 Inside Containment

3.6.1.1 Reactor Coolant System Main Loop Piping

The construction permits issued for constructing the facility provide, in pertinent part, that the facility units are subject to all rules, regulations, and Orders of the Commission. This includes General Design Criterion (GDC) 4 of Appendix A to 10 CFR 50. GDC 4 requires that structures, systems, and components important to safety shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with the normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, discharging fluids that may result from equipment failures, and from events and conditions outside the nuclear power unit.

By letter dated May 31, 1984, the applicants for Wolf Creek Unit 1 and Callaway Unit 1, submitted a report (Westinghouse, WCAP-10500) on the technical bases for eliminating large primary loop piping ruptures as a structural design basis. This submittal was made in support of a request for an exemption to GDC 4 in regard to the replacement of flexible water bag neutron shielding by rigid neutron shields. The flexible shielding was previously designed to prevent the shielding from acting as a missile in the event of pressurization caused by postulated pipe breaks. After meeting with another applicant (Georgia Power Company) and Westinghouse, the staff responded by letter dated March 19, 1984, to transmit the staff's comments and questions on the Georgia Power submittal. The response to the staff's concerns resulted in a revision to the SNUPPS report (Westinghouse, WCAP-10691) which was submitted to the NRC on October 26, 1984. By means of deterministic fracture mechanics analyses, the applicant contends that postulated double-ended guillotine breaks (DEGBs) of the primary loop reactor coolant piping will not occur in Wolf Creek and Callaway units and, therefore, need not be considered as a design basis for installing protective devices such as flexible water bag neutron shielding to guard against the dynamic effects associated with such postulated breaks. No other changes in design requirements are addressed within the scope of the referenced reports; e.g., no changes to the definition of a LOCA nor its relationship to the regulations addressing design requirements for emergency core cooling system (ECCS) (10 CFR 50.46), containment (GDC 16 and 50), other engineered safety features, and the conditions for environmental qualification of equipment (10 CFR 50.49).

The Commission's regulations require that applicants provide protective measures against the dynamic effects of postulated pipe breaks in high energy fluid system piping. Protective measures include physical isolation from postulated pipe rupture locations, if feasible, or the installation of pipe whip restraints, jet impingement shields, or compartments, and the installation of water bags for neutron shielding. In 1975, concerns arose as to the asymmetric loads on pressurized water reactor (PWR) vessels and their internals which could result from these large postulated breaks at discrete locations in the main primary coolant loop piping. This led to the establishment of Unresolved Safety Issue (USI) A-2, "Asymmetric Blowdown Loads on PWR Primary Systems."

The NRC staff, after several review meetings with the Advisory Committee on Reactor Safeguards (ACRS) and a meeting with the NRC Committee to Review Generic Requirements (CRGR), concluded that an exemption from the regulations would be acceptable as an alternative for resolution of USI A-2 for 16 facilities owned by 11 licensees in the Westinghouse Owner's Group (one of these facilities, Fort Calhoun has a Combustion Engineering nuclear steam supply system). This NRC staff position was stated in Generic Letter 84-04, published on February 1, 1984. The generic letter stated that the affected licensees must justify an exemption to GDC 4 on a plant-specific basis. Other PWR applicants or licensees may request similar exemptions from the requirements of GDC 4 provided that they submit an acceptable technical basis for eliminating the need to postulated pipe breaks.

The acceptance of an exemption permitting the replacement of flexible water bags by rigid shielding was made possible by the development of advanced fracture mechanics technology. These advanced fracture mechanics techniques deal with relatively small flaws in piping components (either postulated or real) and examine their behavior under various pipe loads. The objective is to demonstrate by deterministic analyses that the detection of small flaws by either inservice inspection or leakage monitoring systems is assured long before the flaws can grow to critical or unstable sizes which could lead to large break areas such as the DEGB or its equivalent. The concept underlying such analyses is referred to as leak-before-break (LBB). There is no implication that piping failures cannot occur, but rather that improved knowledge of the failure modes of piping systems and the application of appropriate remedial measures, if indicated, can reduce to insignificant values the probability of catastrophic failure.

Advanced fracture mechanics technology was applied in topical reports (WCAP-9558, WCAP-9787, Letter Report NS-EPR-2519) submitted to the staff by Westinghouse on behalf of the licensees belonging to the USI A-2 Owners Group. Although the topical reports were intended to resolve the issue of asymmetric blowdown loads that resulted from a limited number of discrete break locations, the technology advanced in these topical reports demonstrated that the probability of breaks occurring in the primary coolant system main loop piping is sufficiently low that these breaks need not be considered as a design basis for requiring installation of pipe whip restraints, jet impingement shields, or flexible neutron shielding. The staff's Topical Report Evaluation is attached as Enclosure 1 to Generic Letter 84-04, February 1, 1984.

Probabilistic fracture mechanics studies conducted by the Lawrence Livermore National Laboratories (LLNL) on both Westinghouse and Combustion Engineering nuclear steam supply system (NSSS) main loop piping (UCRL-86249) confirm that both the probability of leakage (e.g., undetected flaw growth through the pipe wall by fatigue) and the probability of a DEGB are very low. The results given in this LLNL report are that the best-estimate leak probabilities for Westinghouse NSSS main loop piping range from 1.2×10^{-8} to 1.5×10^{-7} per plant year and the best-estimate DEGB probabilities range from 1×10^{-12} to 7×10^{-12} per plant year. Similarly, the best-estimate leak probabilities for Combustion Engineering NSSS main loop piping range from 1×10^{-8} per plant year to 3×10^{-8} per plant year, and the best-estimate DEGB probabilities range from 5×10^{-14} to 5×10^{-13} per plant year. These results do not affect core melt probabilities in any significant way.

During the past few years it has also become apparent that the requirement for installation of large, massive pipe whip restraints, jet impingement shields, and flexible neutron shielding is not necessarily the most cost-effective way to achieve the desired level of safety, as indicated in Enclosure 2, "Regulatory Analysis," to Generic Letter 84-04. Even for new plants, these devices tend to restrict access for future inservice inspection of piping; or if they are removed and reinstalled for inspection, there is a potential risk of damaging the piping and other safety-related components in this process. If installed in operating plants, high occupational radiation exposure (ORE) would be incurred and public risk reduction would be very low. Removal and reinstallation for inservice inspection also entail significant ORE over the life of a plant.

Parameters Evaluated by the Staff

The primary coolant system of Wolf Creek Unit 1 and Callaway Unit 1, described in WCAP-10691, have four main loops each comprising a 33.9-in.-diameter hot leg, a 36.2-in.-diameter crossover leg, and 32.2-in.-diameter cold-leg piping. The material in the primary loop piping is cast stainless steel (SA 351 CF8A). In its review of WCAP-10691, the staff evaluated the Westinghouse analysis with regard to

- the location of maximum stresses in the piping, associated with the combined loads from normal operation and the safe shutdown earthquake (SSE)
- potential cracking mechanisms
- size of through-wall cracks that would leak a detectable amount under normal loads and pressure
- stability of a "leakage-size crack" under normal plus SSE loads and the expected margin in terms of load
- margin based on crack size
- the fracture toughness properties of thermally aged cast stainless steel piping and weld material.

Staff Criteria Used in the Evaluation

The NRC staff's criteria for evaluating the above parameters are delineated in its Topical Report Evaluation, Enclosure 1 to Generic Letter 84-04, Section 4.1, "NRC Evaluation Criteria," and are as follows:

(1) The loading conditions should include the static forces and moments (pressure, deadweight, and thermal expansion) that result from normal operation, and the forces and moments associated with the SSE. These forces and moments should be located where the highest stresses, coincident with the poorest material properties, are induced for base materials, weldments, and safe-ends.

(2) For the piping run/systems under evaluation, all pertinent information which demonstrates that degradation or failure of the piping resulting from stress corrosion cracking, fatigue, or water hammer is not likely, should be provided. Relevant operating history should be cited, which includes system operational procedures; system or component modification; water chemistry parameters, limits and controls; resistance of material to various forms of stress corrosion, and performance under cyclic loadings.

(3) A through-wall crack should be postulated at the highest stressed locations determined from item 1 above. The size of the crack should be large enough so that the leakage is assured of detection with adequate margin, using the minimum installed leak detection capability when the pipe is subjected to normal operational loads.

(4) It should be demonstrated that the postulated leakage crack is stable under normal plus SSE loads for long periods of time; that is, crack growth, if any, is minimal during an earthquake. This margin, in terms of applied loads, should be determined by a crack stability analysis, i.e., that the leakage-size crack will not experience unstable crack growth even if larger loads (larger than design loads) are applied. This analysis should demonstrate that crack growth is stable and the final crack size is limited, so that a double-ended pipe break will not occur.

(5) The crack size should be determined by comparing leakage-size crack to critical-size cracks. Under normal plus SSE loads, it should be demonstrated that there is adequate margin between the leakage-size crack and the critical-size crack to account for the uncertainties inherent in the analyses, and leakage detection capability. A limit-load analysis may suffice for this purpose; however, an elastic-plastic fracture mechanics (tearing instability) analysis is preferable.

(6) The materials data provided should include types of materials and materials specifications used for base metal, weldments, and safe-ends; the materials properties including the J-R curve used in the analyses; and long-term effects such as thermal aging and other limitations to valid data (e.g., J maximum, maximum crack growth).

Staff Evaluations and Conclusions

On the basis of its evaluation of the analysis contained in Westinghouse Report WCAP-10691, the staff finds that the applicant has presented an acceptable technical justification, addressing the above criteria, for not installing protective devices to deal with the dynamic effects of large pipe ruptures in the main loop primary coolant system piping of Wolf Creek Unit 1. This finding is predicated on the fact that each of the parameters evaluated for Wolf Creek is enveloped by the generic analysis performed by Westinghouse in WCAP-9558, and accepted by the staff in Enclosure 1 to Generic Letter 84-04. Specifically

(1) The loads associated with the highest stressed location in the main loop primary system piping are 1857 kips (axial), 36,397 in.-kips (bending moment), and result in maximum stresses of about 80% of the bounding stresses used by Westinghouse in WCAP-9558. Further, these loads are approximately 74% of those established by the staff as limits (e.g., a moment of 42,000 in.-kips in Enclosure 1 to Generic Letter 84-04).

(2) For Westinghouse plants, there is no history of cracking failure in reactor primary coolant system loop piping. The Westinghouse reactor coolant system primary loop has an operating history that demonstrates its inherent stability. This includes a low susceptibility to cracking failure from the effects of corrosion (e.g., intergranular stress corrosion cracking), water hammer, or fatigue (low and high cycle). This operating history totals more than 400 reactor-years, including five plants, each having 15 years of operation, and 15 other plants, each having more than 10 years of operation.

(3) The leak-rate calculations performed for the SNUPPS units, using an initial through-wall crack of 7.5 in. are identical to those of Enclosure 1 to Generic Letter 84-04. The Wolf Creek/Callaway plants have a reactor coolant system (RCS) pressure boundary leak detection system which is consistent with the guidelines of Regulatory Guide (RG) 1.45, and that system can detect leakage of 1 gpm in 1 hour. The calculated leak rate through the postulated flaw results in a factor of at least 10 relative to the sensitivity of the Wolf Creek/Callaway leak detection systems.

(4) The margin in terms of load of the SNUPPS units based on fracture mechanics analyses for the leakage-size crack under normal plus SSE loads is within the bounds calculated by the staff in Section 4.2.3 of Enclosure 1 to Generic Letter 84-04. On the basis of a limit-load analysis, the load margin is about 2.2, and on the basis of the J limit discussed in item 6 below, the margin is at least 1.1 to 1.2.

(5) The margin between the leakage-size crack and the critical-size crack was calculated by a limit load analysis. Again, the results demonstrated that a margin of at least 3 exists and is within the bounds of Section 4.2.3 of Enclosure 1 to Generic Letter 84-04.

(6) As an integral part of its review, the staff's evaluation of the material properties data of WCAP-10456 is enclosed as Appendix M to this SER supplement. In WCAP-10456, data for 10 plants, including Wolf Creek, are presented, and lower bound or "worst case" materials properties were identified and used in the analysis performed in the Westinghouse report WCAP-10691. The applied J for Wolf Creek in WCAP-10691 was less than 3,000 in.-lb/in.^{2} and hence the staff's upper bound on the applied J (refer to Appendix M, page 6) was not exceeded.

Radiological Protection Consideration

The reactor cavity shield design, which presently uses water bags to reduce the neutron flux that streams into containment from the annulus between the reactor pressure vessel and biological shield, will be replaced by a rigid shield.

This change should not reduce the radiation protection criteria for limited access into containment during reactor operations.

The rigid shield will present no fire or smoke hazard and will have an equivalent factor of reduction of the neutron dose equivalent rate (mrem/hr) as compared to the water bags. Additionally, activation of the stainless steel container of the rigid shielding material, should not significantly increase the risk of exposure to personnel during handling and storage of the shield at refueling intervals.

On the basis of the applicant's response to staff concerns relevant to the rigid shield characteristics as described above, the staff finds the rigid shield an acceptable replacement for the water bags to reduce the dose rate from neutron and gamma radiation streaming into containment.

In view of the staff's evaluation findings, conclusions, and recommendations above, the Commission has determined that, pursuant to 10 CFR 50.12(a), this exemption is authorized by law and will not endanger life or property or the common defense and security, and is otherwise in the public interest. The Commission hereby approves the limited exemption from GDC 4 of Appendix A to 10 CFR 50, to permit the applicant to replace the flexible water bag neutron shielding in the reactor cavity with rigid shielding.

3.8 Design of Seismic Category I Structures

3.8.3 Other Seismic Category I Structures

By letter dated December 3, 1984, the applicant proposed an addition to Section 3.8.3.5.3.3 of the SNUPPS FSAR in forthcoming revision which is applicable to Wolf Creek, as follows: "(2) Fillet welds need not satisfy the convexity limitations of Section 3.6.1 provided that all other parameters of acceptable weld profile are maintained."

The applicant justified the change on the basis that convexity has no detrimental effect on the structural capacity of the weld if other weld parameters are maintained. The staff has reviewed the change and has the following observation. It is well known that a convex fillet weld has less tendency to crack as a result of shrinking while cooling, and it is relatively free from undercut. Even though such a weld may result in excess weld metal, it will not affect the allowable strength of weld determined on the basis of the weld leg size.

As a result of the above observation, the staff concluded that the proposed change to the SNUPPS FSAR Section 3.8.3.6.3.3, applicable to Wolf Creek, is acceptable.

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

3.10.1 Seismic and Dynamic Qualification

The staff evaluation of the applicant's program for qualification of safety-related electrical and mechanical equipment for seismic and dynamic loads consists of (1) a determination of the acceptability of the procedures used, standards followed, and the completeness of the program in general; and (2) an audit of selected equipment items to develop the basis for the staff judgment on the completeness and adequacy of the implementation of the entire seismic and dynamic qualification program. The seismic qualification review team (SQRT) consists of engineers from the NRC staff and the Idaho National Engineering Laboratory (INEL, EG&G).

The SQRT has reviewed the equipment dynamic qualification information in FSAR Sections 3.9.2 and 3.10 and visited the plant site December 8 and 9, 1983, to determine the extent to which the qualification of equipment as installed at Wolf Creek meets the current licensing criteria in RGs 1.100 and 1.92, SRP 3.10, and Standard 344-1975 of the Institute of Electrical and Electronics Engineers (IEEE). The applicant must conform to these criteria to satisfy the applicable portions of GDC 1, 2, 4, 14, and 30 of Appendix A to 10 CFR 50; Appendix B to 10 CFR 50; and Appendix A to 10 CFR 100. The SQRT audited a representative sample of safety-related electrical and mechanical equipment as well as instrumentation included in both the nuclear steam supply system (NSSS) and the balance-of-plant (BOP) systems. The site visit included field observations of the actual, final equipment configuration and its installation, and a review of the corresponding test and/or analysis documents in the applicant's central files. The SQRT observed the field installation of the equipment to verify and validate equipment modeling employed in the qualification program.

On the basis of the audit and review of the applicant submittals, it is the staff's opinion that the SNUPPS seismic and dynamic qualification program of equipment has been satisfactorily defined and implemented for Wolf Creek Unit 1 to the extent that a low-power license may be issued. The staff's findings are summarized in Sections 3.10.1.1, 3.10.1.2, and 3.10.1.3 of this report, and a summary of the staff's evaluation of the applicant's program is provided in Section 3.10.1.4.

3.10.1.1 Generic Issues

Surveillance and Maintenance Programs

The applicant must confirm that all safety-related Class 1E and age-sensitive mechanical components in both harsh and mild environments are covered in the surveillance and maintenance programs and also that the qualified life of these equipment items is estimated. The applicant must also provide for staff review the description of the surveillance and maintenance program for a sample equipment item.

The SNUPPS independent review of the environmental qualification of safety-related equipment according to NUREG-0588 addressed only equipment in harsh environment areas. However, by letter dated February 2, 1984, the applicant* states that the maintenance and surveillance programs at the SNUPPS plants are applicable to all safety-related equipment, whether it is located in harsh or mild environment areas. The applicant noted that, to clarify this point, Section 8.7 of the NUREG-0588 submittal was revised. This is acceptable to the staff.

*SNUPPS speaks for the applicant.

3.10.1.2 Specific Issues

Power-Operated Relief Valve

There was a difference between the tested and the field configuration of the power-operated relief valve. The position indication device and terminal strip within the electrical compartment of the valve were tested in a sealed configuration, not allowing the environment to enter the compartment. The field configuration did not have this attribute. The applicant has modified the field configuration to provide a sealed installation. This is acceptable to the staff.

Pressurizer Safety Valve

The operability of the SNUPPS safety valve is addressed under TMI Action Plan Item II.D.1 in Section 22 of this supplement. The applicant must provide qualification documentation for the position indication device. The device is environmentally qualified under Westinghouse Equipment Qualification Data Package HE-7, which is scheduled to be submitted to the NRC by November 30, 1985. The staff has reviewed the applicant's justification for interim operation for this equipment submitted by letter dated June 29, 1984, and found it acceptable. This conclusion is based on the fact that the device has successfully passed plant-induced-vibration aging and seismic testing conducted according to IEEE 344-1975 (see Section 3.10.1.3, below).

Main Steam Isolation Valve, Loop 2

The applicant was required to submit the data on the qualified life of the nonmetallic parts in this equipment item, once it was established.

The applicant revised the SNUPPS NUREG-0588 submittal (Revision 2, Vol. 2, Specification M-628) to include this information. The staff has reviewed M-628 and found that a maximum qualified life of 5 years is specified for a number of nonmetallic subcomponents. This is acceptable to the staff.

Medium Voltage Metal Clad Switchgear

The original switchgear fuses in the field were not seismically qualified. In a letter dated March 16, 1984, the applicant stated that the original fuses in 4.16-kV switchgear have been replaced with fuses that have been seismically qualified under Specification E-018, "Motor Control Centers." The staff finds this change acceptable and will ensure that the change is completed before fuel load.

Motor Control Center

The applicant was required to submit data on the qualified life of the motor control center components. The staff has reviewed Specification E-018 and found that a qualified life of 12 years has been calculated for the ground fault sensor. The rest of the components have a qualified life of 40 years. This information is acceptable to the staff.

3.10.1.3 Justification for Interim Operation

Twelve categories of equipment for which qualification documentation is not expected to be fully complete by fuel loading were not specifically included among the items reviewed by the SQRT. The applicant has, however, provided justification for interim operation (JIO) that was accepted by the staff in NUREG-0830, Supplement 3, as adequate for 5% power operation. The basis of the staff conclusion was (1) evidence of partial qualification results or (2) insignificance of impact to the safety function of the system.

Subsequently, SNUPPS submitted additional JIOs, in its letters of June 29 and July 16, 1984, to its request for full-power operation while the qualification program for some of these equipment items is in progress. As is indicated, the applicant has, in some cases, incorporated the JIO in the equipment qualification documentation. In all such cases, the applicant has stated that testing has been sucessfully completed according to the staff licensing criteria. When completed equipment qualification documentation is available, it will be substituted for the JIO in the documentation file to ensure uniformity of the file.

The staff reviewed the additional information provided in the above SNUPPS letters and found that some of the previously unqualified equipment items have now been completely qualified for SNUPPS application, and that the associated JIO as discussed in NUREG-0830, Supplement 3, should be terminated. The staff has also found that the additional JIOs, as presented for other equipment items, are acceptable to the staff for supporting low-power operation of Wolf Creek Unit 1. Discussion for each individual equipment item follows.

Crosby Position Indication Device (HE-7)

On the basis of previous tests, the failure mechanism of the position indication device (PID) had been concluded to be the moisture/chemical spray inwicking along the lead wires that damaged reed switches and degraded electrical performance of the switches. The applicant committed to have the connection sealed with seismically and environmentally qualified Conax connectors. In addition, previous seismic testing has provided acceptable evidence that the PID is seismically qualified. On the basis of the above and the fact that the complete qualification test of the assembly of the individually qualified items is in process and will be completed by November 30, 1985, the staff concludes that the SNUPPS JIO is acceptable for full-power operation. The applicant should provide a written confirmation that the qualification test, when completed, will meet the regulatory requirements.

7300 Process Protection System (ESE-13)

In a recent seismic and environmental test, the NPC (potentiometer), NCH (function generator), NSE (signal converter), and NTD (tracking driven) cards and cabinet power supply exhibited errors. Westinghouse is evaluating these errors to determine their exact effect. The initial Westinghouse evaluation indicates that there are adequate margins for both generic and SNUPPS applications. The evaluation also indicates that NTD card contact bounce has an insignificant impact on system operation.

The NTC (temperature channel test) card exhibited contact bounce during the testing, and the applicant evaluated and reported the results to the NRC as a 10 CFR 50.55(e) issue on June 2, 1983 (see letter dated July 21, 1983). The SNUPPS design has this NTC card in the overtemperature and overpower ΔT channels. This intermittent signal may cause saturation of downstream RTD (resistance temperature detector) amplifier (NRA) cards and could prevent a trip from occurring on demand. The applicant has issued a field change notice that will provide a method of bypassing these relays during normal operation until a permanent resolution is prepared. Because these relays are used only to ease periodic testing of the channels, this interim modification will not interfere with plant operation.

The applicant was committed to complete the qualification program for this item before exceeding 5% power operation. Review of the JIO for this system has led the staff to conclude that the equipment in the Wolf Creek plant is seismically qualified. The JIO will be used as supporting documentation for seismic qualification until the final documentation is finished. Final documentation is required to ensure uniformity of data in the equipment qualification document files and will be completed by November 30, 1985. The applicant should provide a written confirmation that the qualification program, when completed, will meet the regulatory requirements.

Boron Dilution Protection System (ESE-47)

The operational concern on the source-range preamplifier, a part of the system, leads to a new preamplifier (model MK II) to replace the old one (model MK I). Functionally, testing has proven this to be a superior design. However, the new redesigned triaxial connector, which was in the field, failed during the seismic test. The old style connector was then installed and subsequent seismic test results on the preamplifier were satisfactory. Furthermore, the applicant has already placed these old connectors in the field.

On the basis of the above, the seismic qualification of the ESE-47 equipment has been demonstrated for SNUPPS application, and the staff is in agreement that the JIO of this equipment should be terminated.

Thermocouple/Core Cooling Monitor System (ESE-56A)

The failure of the plasma display during safe shutdown earthquake (SSE) testing in positions 3 and 4 is attributed to fretting of the edge connector contacts and board edge fingers which produces microscopic particles of oxidized material that act as an insulator causing intermittent open circuits. On the basis of the symmetry of construction, the direction of excitation is determined to be an insignificant factor in fretting. It is, therefore, concluded that the unit is adequate for one SSE. To provide additional margin, however, the manufacturer was developing a lubricant/oxidation inhibitor which would be applied at SNUPPS. As a result of a later decision by the applicant, the inhibitor will not be applied.

The problem of intermittent output from the PS-2 power supply during seismic testing was attributed to temperatures greater than or equal to 138°F. This was confirmed when performance resumed after the temperature was reduced. For SNUPPS, this system is located in the control room which has Class 1E heating, ventilation, and air conditioning (HVAC) and will not likely experience abnormal temperature. However, further testing is scheduled to qualify the PS-2 power supply for different, harsh environment applications. The TC/CCM system has been successfully seismically tested after certain hardware modifications. The seismic qualification of this system will be considered demonstrated when Westinghouse Field Change Notice (FCN) SAPM-10627 has been completed for Wolf Creek. By letter dated February 22, 1985, the applicant has documented the completion of the FCN. The staff finds the JIO acceptable and it will serve as documentation of qualification for the system until complete equipment qualification documentation, scheduled for completion by November 1985, is available. The staff will ensure that the field modifications are completed. In the meantime, the applicant should provide a written confirmation that the qualification program, when completed, will meet the regulatory requirements.

International Instruments Model 1151 Indicators (J-110)

Adequate seismic testing has been performed for SNUPPS by American Environments and witnessed by Bechtel Power Corporation. Minor anomalies which occurred were judged to be insignificant. The qualification program, including full documentation, has been completed. The staff agrees that the JIO of this equipment should be terminated.

AWV Model 7401 Damper (M-627A)

A seismic test to verify the acceptability of the modified dampers for SNUPPS was completed in February 1984. The results were determined to be satisfactory. Test reports have been reviewed and approved. The dampers, therefore, have been fully qualified for SNUPPS applications. The staff agrees that the JIO of this equipment should be terminated.

Operator Interface Module (OIM)

During seismic testing, all four current meters were well within accuracy requirements, and the meters are qualified with no anomalies of the associated switches observed. One of the three Brush recorders for the 500-ohm potentiometer, however, indicated momentary interruptions of the signal. Such anomaly is not significant since, as stated above, the potentiometer is not required to function during the event.

The anomaly of the current meters which was observed during abnormal environment testing at high temperatures is not applicable to the SNUPPS plants because of the SNUPPS Class 1E control room HVAC system.

The current meters included in the seismic test were the modified meters of the abnormal environment test. Unmodified meters are considered to be represented by this test since the addition of the temperature-compensating resistor has no impact on the seismic capability of the meter. Thus, these test results are applicable to unmodified meters.

In the recent letter of October 9, 1984 from SNUPPS, it was revealed that the OIMs installed at the SNUPPS plants were fabricated before the vendor had established a subcomponent material certification program. However, Westinghouse has subsequently completed a review of material traceability and concluded that the materials used in the SNUPPS meters are similar to those used in the qualification program. This is acceptable to the staff.

In addition, the above letter indicates that the SNUPPS meters are susceptible to inaccuracies because of the initial calibration method employed. This may result in exceeding allowable accuracy values during a seismic event. Also the SNUPPS OIMs were shipped without circuit board mounting clips which were used on the tested equipment. The clips are required to ensure seismic qualification of the meters and controls of the OIM. By letter dated February 15, 1985, the applicant confirmed that the recalibration and installation of the mounting clips committed to in the applicant's December 14, 1984, letter have been completed.

The OIMs are used to control the following six valves in the SNUPPS plants:

(1) BB-HV-8157 A and B - excess letdown to pressurizer relief tank (PRT)
(2) BG-HV-8357 A and B - alternate reactor coolant pump seal injection
(3) EM-HV-8837 A and B - boron injection line bypass

BB-HV-8157 A and B are used to regulate letdown flow to the PRT. BG-HV-8357 A and B are used to provide a safety-grade, controlled charging flowpath to the reactor coolant pump (RCP) seals. EM-HV-8837 A and B are used to supply metered charging flow to the reactor coolant system so that charging and letdown flow rates can be matched.

For all three types of valves, redundant flow indications are available for operator use in establishing or terminating the flows: BB-FI-138 A and B for excess letdown to PRT flow, BG-FI-215 A and B for seal injection flow, and EM-FI-917 A and B for boron injection line flow. Loss of function of the above valves has been evaluated in connection with the availability of redundant systems. The function of BB-HV-8157 A and B can be performed by the alternate method utilizing emergency letdown to the PRT via the fully qualified pressurizer power-operated relief valves. Loss of function of BG-HV-8357 A and B is not critical because the primary seal injection flowpath to the RCP seals is expected to be operable. This flowpath provides seal injection flow via locked open manual valves and an air-operated flow control valve (BG-FCV-121) which fails in the open position on loss of air or electrical power. The function of EM-HV-8837 A and B can be performed by manually throttling the motor-operated valves, EM-HV-8803 A and B, in the boron injection line. In addition, spurious opening of the valves as a result of OIM failure can also be mitigated by operation of some designated safety-grade equipment, as was also indicated by the applicant. Furthermore, the applicant has provided written instructions to the operators regarding the potential for the OIM to produce unreliable readings, as is evidenced by the applicant's letter of October 12, 1984.

On the basis of the above information presented by the applicant as well as the evidence of the seismic testing already conducted, the staff concludes that the operator interface modules have been seismically qualified and the JIO should be terminated.

<u>Cutler Hammer Series E-30 Pushbutton Assemblies (E-028, J-200, J-201)</u>

The seismic testing of E-30 pushbutton assemblies has been completed at Wyle Laboratories. Testing was performed to the requirements of IEEE 323-1974 and

IEEE 344-1975. The qualification program, therefore, has been completed. The staff agrees that the JIO of this equipment should be terminated.

Head Vent System Control Module (HE-10B)

A seismic test of this module, utilizing multiaxis, multifrequency input, has been performed which met or exceeded the prescribed requirements; no failures were detected. The qualification program, including full documentation, has been completed. The staff agrees that the corresponding JIO should be terminated.

Incore Thermocouples, Connectors, Adapters, and Reference Junction Box--Core Cooling Monitor System (ESE-43 and ESE-44)

The JIO was based on a nearly completed qualification test series with evidence that the series could be successfully completed. The testing of the junction box for postaccident radiation exposure needed repeating because of a loss of seal on the original loss-of-coolant accident (LOCA) testing. This did not affect the seismic qualification of the box because the occurrence of a seismic event following a design-basis accident has not been defined as a credible event. The JIO which describes seismic testing is considered acceptable for the SNUPPS equipment. Complete equipment qualification documentation is scheduled to replace the JIO in the documentation file by November 30, 1985. At that time, the applicant should provide a written confirmation that the qualification program, when completed, will meet the regulatory requirements.

Barton Differential Pressure Indicating Switches (ESE-40) Model Nos. 288A and 581A

The JIO was based on previously completed testing and an analysis indicating that seismically induced chatter, shown to be possible by the testing, will not degrade the performance of the systems in which the switches are installed to unacceptable levels. A change in switch setpoint in the field is required to ensure this. By letter dated February 22, 1985, the applicant has confirmed the implementation of the switch setpoint change. This JIO is acceptable to the staff and will serve as an interim documentation. Full documentation will be completed by November 1985. At that time, the applicant should provide a written confirmation that the qualification program, when completed, will meet the regulatory requirements.

Rosemount Pressure and Differential Pressure Transmitter (ESE-2D and ESE-4D)

Rosemount transmitters are being used as replacements for Barton transmitters in selected, outside-containment locations at Wolf Creek Generating Station.

Two types of Rosemount transmitters have been provided by Westinghouse for use at Wolf Creek. Model 1153B and 1153D transmitters were qualified in separate programs of qualification testing performed by Wyle Laboratories. The qualification program addressed the requirements of IEEE 323-1974 for harsh environment qualification. The Rosemount model 1153B and 1153D transmitters are used in SNUPPS balance-of-plant (BOP) applications under Specification J-301. On the basis of the SNUPPS review of the Rosemount reports supporting J-301, these transmitters were determined to be fully qualified for their applications in accordance with NUREG-0588 requirements as documented in the SNUPPS NUREG-0588 submittal.

Westinghouse evaluated the methodology and results of the portions of the test sequence which support qualification for Group B applications and noted differences. There are considerable differences in the seismic test methodology of Wyle and that of Westinghouse described in WCAP-9714. Westinghouse reviewed the Wyle methodology, test plan, and results and compared them to Westinghouse test conditions. SSE and OBE response spectra were established in this evaluation. The response spectra represent the envelope of qualification after reconciliation of methodology differences. The spectra envelope the plant-specific required response spectrum for Wolf Creek. Seismic qualification consistent with the WCAP-8587 program is demonstrated.

Therefore, on the basis of the reconciliation of differences between the qualification methodology used by Wyle and that of Westinghouse, the staff concludes that the JIO is acceptable for full-power operation.

3.10.1.4 Summary

On the basis of SQRT audit findings as well as on the review of subsequent submittals including the JIO, the staff concludes that an appropriate seismic and dynamic qualification program has been defined and implemented which provides adequate assurance that such equipment should function properly during and after the excitation from vibratory forces imposed by the safe shutdown earthquake. The staff finds the SNUPPS seismic and dynamic qualification program for Wolf Creek Unit 1 to be acceptable and to meet all the applicable portions of General Design Criteria previously stated, pending completion of all the confirmatory items as stated in Sections 3.10.1.2 and 3.10.1.3. The staff recommends, therefore, low-power operation for Wolf Creek Unit 1.

3.10.2 Operability Qualification of Pumps and Valves

To ensure that the applicant has provided an adequate program for qualifying safety-related pumps and valves to operate under normal and accident conditions, the staff performs a two-step review. In the first step, the description of the applicant's pump and valve operability assurance program in FSAR Section 3.9.3.2 is reviewed and compared to Section 3.10 of the NRC Standard Review Plan (SRP, NUREG-0800). The information in the FSAR, however, is general in nature and not sufficient by itself to ensure the adequacy of the applicant's overall program for pump and valve operability qualification. To provide this assurance, the pump and valve operability review team (PVORT) conducts an onsite audit of a small representative sample of safety-related pumps and valves and their supporting documentation.

The onsite audit includes a plant inspection of the as-built configuration and installation of the equipment, a discussion of the normal and accident conditions under which the equipment and system must operate, and a review of the qualification documentation (stress reports, test reports, etc.).

The two-step review is performed to determine the extent to which the qualification of equipment, as installed, meets the current licensing criteria in

SRP 3.10. Conformance with these criteria provides an acceptable way of meeting GDC 1, 2, 4, 14, and 30, in Appendix A to 10 CFR 50 as well as Appendix B to 10 CFR 50.

A pump and valve operability review was conducted simultaneously for the Callaway and Wolf Creek generating stations. Both plants were designed and constructed in accordance with the SNUPPS design. The SNUPPS FSAR covers the equipment common to both plants; plant-specific equipment, such as the essential service water system, is addressed in plant-specific addenda. The PVORT review of the FSAR included the SNUPPS FSAR and the plant-specific addenda.

The onsite audits were conducted at the Callaway and Wolf Creek plants to observe the as-built installation of the equipment, to perform a detailed review of the qualification program for selected equipment, and to compare plant-specific features. A walkdown was performed at each plant to examine the selected equipment. Whenever possible, the plant engineers described the features and procedures unique to that plant.

The qualification documents for the selected equipment were reviewed at Callaway only, because most of the equipment and documentation for both plants is identical. Plant-specific equipment, such as the essential service water pump, was reviewed to the extent necessary to determine its qualification. In this case, information that distinguished the pump design at each plant with respect to analysis, testing, and installation was included in the review. The PVORT found that the comments generated from the document review apply to both Callaway and Wolf Creek, unless otherwise stated.

The onsite audit for Callaway and Wolf Creek was performed December 5 to 9, 1983. A representative sample of three pumps and seven valves was chosen for the review. The sample included both NSSS and BOP equipment. Nine of the ten sample equipment items were identical for both plants. The essential service water pump was chosen to investigate its plant-specific application. At Wolf Creek, the ultimate heat sink is a cooling lake. All ten components were examined during separate walkdowns at each plant.

During the PVORT review, a number of concerns were raised. The applicant resolved all of these concerns during the audit by either supplying additional information or by demonstrating that the appropriate commitments are already addressed by administrative controls. The discussion below shows how the applicant addressed generic and specific issues at the Wolf Creek plant.

3.10.2.1 Generic Findings

In preparation for the PVORT audit, the staff reviewed the SNUPPS FSAR Section 3.9.3.2 and the master list of seismic Category I equipment. The applicant provided enough information in these documents to allow the staff to conduct the onsite audit. This information was augmented by discussions with plant personnel during the audit.

The staff has determined that the SNUPPS FSAR Section 3.9.3.2 should be amended to provide a more current and detailed description of the pump and valve operability program. The applicant should provide separate FSAR tables listing the active safety-related balance-of-plant (BOP) pumps and valves. The BOP and

NSSS equipment items are listed in FSAR Table 3.11(B)-3. The new BOP pump and valve tables should be included in FSAR Section 3.9(B).3.2 and presented in a format that is consistent with the existing NSSS pump and valve tables [3.9(N)-10 and 3.9(N)-11] and the information in Table 3.2-1. Also, the FSAR should explain the criteria for determining which BOP and NSSS pump and valve accessory equipment are incorporated into the FSAR lists of active safety-related equipment. As an example, the applicant considers the turbine lube oil pump as a functional accessory to the auxiliary feedwater turbine. Consequently, the table of active pumps does not list the turbine lube oil pump. The applicant has committed to incorporate this listing in an FSAR revision. This commitment is acceptable to the staff.

No generic operability concerns resulted from the evaluation of the Wolf Creek qualification programs for pump and valve operability. A minor area of concern regarding the permanent tagging of plant equipment was discussed with plant personnel and satisfactorily resolved. The applicant demonstrated that the equipment could be positively identified by other cross-reference numbers should the plant identification tag be misplaced. A related concern was the number of quality hold and temporary modification tags that were attached to many pieces of installed equipment. The Wolf Creek systems engineers explained that the plant was currently undergoing preoperational tests. Test procedures were reviewed that specified test sequences, system lineup procedures, and temporary equipment changes. The applicant described a program of tracking the plant's operational state. The execution of this program satisfactorily addressed the concern of the operational status of plant equipment. Documentation of changes showed full accountability of equipment status and resolved the concerns raised by the staff.

The applicant presented orientation lectures on maintenance and surveillance, preoperational testing, and inservice tests. In addition, the PVORT made a limited review of the corresponding documentation. Wolf Creek has a computer-based maintenance program that appears to be very comprehensive. This program (1) incorporates all of the pertinent data provided by the vendors, such as aging information, spare parts, and schedules; (2) takes initial signatures of equipment to detect and analyze trends that may be indicative of degradation and to implement a vibration analysis program; (3) provides a closed-loop check by the quality assurance group to inspect and verify maintenance and other related activities performed on the equipment; and (4) analyzes equipment when it is removed to help determine changes in the inspection and replacement schedules. Wolf Creek also participates in the nuclear plant reliability data system (NPRDS). The staff concludes that qualification considerations have been incorporated into the maintenance, surveillance, preoperational testing, and inservice test programs and finds the applicant's methods acceptable.

Comprehensive startup and preoperational test programs are under way. The test program consists of (1) generic procedures for system lineup and component calibration and (2) specific procedures for equipment functional performance. A detailed program of documentation and administrative procedures addresses temporary alterations of equipment in preparation for testing, equipment status, consideration for retest, and test compliance. The staff concludes that the procedures, as they relate to equipment status and qualification, use the quality assurance criteria of Appendix B to 10 CFR 50 and are acceptable.

The Wolf Creek inservice test program has been submitted to the staff for review and approval. The applicant has committed to meet, through this program, the requirements of ASME Code Section XI, paragraphs IWV and IWP (1980 Edition, Winter 1981 Addenda). The inservice test program for Wolf Creek provides a series of closed-loop checks for (1) test execution and acceptance or (2) test rejection and procedures for retest and compliance. There are direct links with maintenance and quality assurance personnel to ensure that test inspection schedules are followed. The inservice test program satisfactorily addresses operability- and qualification-related concerns such as sensitive or potentially degradable subcomponents.

With respect to the long-term operability of deep draft pumps, the staff has reviewed the applicant's response to the NRC Office of Inspection and Enforcement (IE) Bulletin No. 79-15. The staff finds that the applicant's long-term operability verification program is in general agreement with the procedure accepted by the staff for detecting problems with deep draft pumps. The staff also finds this to be a satisfactory resolution to the concern regarding operability of deep draft pumps at Wolf Creek.

At the time of the PVORT audit, the applicant had not completed the qualification programs for a small percentage of components. By cross-referencing specific environmental and seismic qualification sections of the SNUPPS "Report of Independent Review of Environmental Qualification Programs to NUREG-0588" and the qualification program in SNUPPS FSAR Section 3.10, the applicant has listed several items whose complete qualification will not be confirmed until after fuel loading, and has provided justification for interim operation. The staff has reviewed this justification and finds it satisfactory because the applicant has

(1) presented a rigorous test program based on IEEE 323-1974 and 344-1975 and RGs 1.89, 1.100, and 1.73

(2) established administrative programs in conformance with RG 1.33 to ensure that equipment is maintained in a qualified status throughout the life of the plant

(3) committed to complete qualification no later than November 1985

The staff required that the SNUPPS FSAR Section 3.9.3.2 be amended to provide a more current and detailed description of the pump and valve operability program, including a description of the criteria for determining which BOP and NSSS pump and valve accessories are incorporated into the FSAR lists of active safety-related equipment. By letters dated March 16 and March 24, 1984, the applicant committed to comply with the staff request in a future revision of the SNUPPS FSAR. This update was done in Revision 15 to the SNUPPS FSAR and is acceptable to the staff.

The staff also required the licensee to verify that all safety-related equipment is fully qualified, and the licensee addressed this in letters dated March 16 and June 29, 1984. The staff has reviewed these responses and concludes that, except for equipment-specific issues which are discussed below in Section 3.10.2.2, all generic concerns are resolved.

3.10.2.2 Equipment-Specific Issues

A number of concerns noted during the walkdown were satisfactorily resolved during the audit. Many of these issues were attributed to the preoperational and hot functional test schedules in progress during the review; other concerns were satisfactorily addressed by administrative controls already in effect. In all cases, the applicant was able to explain and justify the presence of quality control tags, temporary equipment modifications, test preparations, and followup procedures. The PVORT made a check of the applicant's documentation system by requesting, at short notice, and reviewing in detail the appropriate quality tags, startup work requests, test reports, and related document controls.

The staff concludes that the applicant's procedures for tracking equipment status are followed in an orderly and disciplined manner and that immediate attention is given to identifying safety-related equipment. Thus, the procedures are acceptable.

Conclusion

The staff finds that the applicant is dealing with the equipment qualification issue in a positive manner. The staff has reached this conclusion because the applicant has (1) provided adequate documentation to demonstrate qualification of safety-related pumps and valves; (2) established administrative programs to determine, monitor, and maintain equipment operability for the lifetime of the plant; (3) demonstrated an adequate central file system by the timely retrieval of information requested by the staff during the audit; (4) communicated with Wolf Creek personnel to discuss and compare details of construction, utility policy, and plant operation; and (5) demonstrated overall accountability by committing the appropriate personnel to implement these programs.

On the bases of the results of the site review performed December 5 to 9, 1983, and the subsequent submittals by the applicant to resolve issues identified during the review, the staff concludes that the applicant has defined an acceptable pump and valve operability qualification program. The continuous implementation of this overall program will ensure that the safety-related functions will be performed as needed.

On June 29, 1984, the applicant provided additional information regarding its April 1984 justification for interim operation (JIO) in order to justify plant operation above the 5% power level. The staff reviewed the latest submittal. The staff's review and acceptance are based on the following reasons.

(1) JIOs HE-1, HE-9, HE-10A, and HE-10C were issued because the documentation has not been reviewed in accordance with the SNUPPS procedures described in the SNUPPS submittal for NUREG-0588. The JIOs, which are standard Westinghouse Equipment Qualification Data Packages (EQDPs), document the successful completion of rigorous testing programs to the requirements of IEEE 323-1974 and 344-1975.

(2) The applicant has stated that the above equipment does comply with the operability qualification provisions of the SNUPPS FSAR, although the

SNUPPS review of the documentation is not complete. The applicant has committed to complete its review by November 1985.

(3) The equipment accessories, whose qualification is incomplete, impact the safety function of the system minimally. JIO HE-7 addresses the qualification of a position-indication device, the design of which is such that it does not cause malfunction of the pressurizer safety valve. JIO J-601A addresses the qualification of the NAMCO limit switch for a design-basis accident (DBA) radiation. The associated containment isolation valve will perform its safety function within minutes of the beginning of the DBA. Any subsequent failure of the limit switch will not cause the valve to change position. By letter dated December 21, 1984, the applicant has committed to close out JIO HE-7 and J-601A by November 1985.

The staff, however, requires that the applicant, upon completion of the qualification program based on methodology accepted by the staff, confirms in writing that the program, including upgrading of equipment qualification files, is complete and that the governing qualification standards are met.

On the basis of the results of the site review performed for Callaway and Wolf Creek between December 5 and 9, 1983, and the subsequent submittal by the applicant to resolve issues identified from the site review, the staff has concluded that an appropriate pump and valve operability qualification program has been defined and implemented. The staff finds that the SNUPPS pump and valve operability assurance program is acceptable.

On the basis of the staff review and acceptance of the JIO and the requirement of written confirmation by the applicant of the completion of all items of the pump and valve operability qualification program in accordance with approved standards, the staff recommends full-power operation for Wolf Creek Unit 1.

3.11 Environmental Qualification of Safety-Related Electrical Equipment

10 CFR 50.49(b)(3) requires that all installed RG 1.97, Revision 2, Category 1 and 2 instrumentation located in a harsh environment be included in the equipment qualification program. The applicant has stated that all Category 1 instruments and all Category 2 instruments powered by a Class 1E power source are included in the equipment qualification program. The applicant has also provided a response to RG 1.97, Revision 2. This response is under review by the staff. The applicant has submitted an analysis to justify interim operation until the review is completed, based on the accomplishment of the function by designated alternative equipment for those items not presently included in the equipment qualification program. The staff has reviewed this analysis and finds it acceptable for interim operation of the plant.

With respect to TMI Action Plan Item II.F.1, Attachment 1, the staff has concluded that the type, make and model, calibration, and operation of the SNUPPS high-range postaccident effluent monitors are acceptable and that compliance with TMI Action Plan Item II.F.1, Attachment 1, is complete. The applicant has stated that this equipment is located in a mild environment and is, therefore, outside the scope of the environmental qualification program.

3.11.3 Staff Evaluation

3.11.3.2 Safety-Related Mechanical Equipment in a Harsh Environment

Qualification documentation has been submitted by the applicant and has been reviewed by the staff. The staff review has verified that the requirements for environmental qualification of safety-related mechanical equipment have been fully addressed.

3.11.3.3 Aging

The applicant has described a program that incorporates the guidelines discussed in Wolf Creek SSER 4. In SSER 4, the applicant was requested to provide information on the specific maintenance/surveillance activities to be performed on five specific equipment items.

The surveillance/maintenance program described by the applicant was found acceptable, and the applicant has committed to implement the program before fuel loading. The applicant has also described a program to detect age-related degradation of electrical cables inside containment that includes a periodic inspection of selected cables. The program described has been reviewed by the staff and is acceptable.

3.11.3.4 Outstanding Equipment

For safety-related items not having complete qualification documentation, the applicant has provided commitments for corrective action and schedules for completion. For items identified to date that will not have full qualification before an operating license is issued, analyses have been performed in accordance with 10 CFR 50.49(i) to ensure that the plant can be operated safely pending completion of environmental qualification. These analyses have been submitted for consideration. The staff has reviewed the JIOs and has concluded that reasonable assurance has been provided that the SNUPPS plants can be operated safely pending completion of environmental qualification.

As a result of recent inspections of the Rockbestos Company, the NRC staff has determined that there is doubt about the validity of the test reports referenced by the applicant to demonstrate qualification of Rockbestos electrical cables. However, on the basis of the results of review of the information available, the staff concludes at this time that no safety problem exists in the use of these cables. IE Information Notice 84-44 concerning this item has been issued by the NRC Office of Inspection and Enforcement. It is the applicant's responsibility to evaluate the content of this Information Notice for applicability to its facility and take appropriate action to ensure that Rockbestos electrical cable is validly qualified.

3.11.4 Qualification of Equipment

3.11.4.1 Electrical Equipment Important to Safety

The staff has separated the electrical equipment in a harsh environment into three categories: (1) equipment requiring replacement before plant startup, (2) equipment requiring additional qualification information or corrective action, and (3) equipment considered acceptable pending implementation of the maintenance and surveillance program.

3.11.4.1.1 Equipment Requiring Replacement Before Plant Startup

The applicant has committed to replace Marathon 1600 terminal blocks before fuel load. This is discussed in Section 3.11.4.3 below.

3.11.4.1.2 Equipment Requiring Additional Information and/or Corrective Action

Table 3.2 in SSER 4 identified equipment for which additional information or corrective action was required. The applicant has stated that all of the concerns identified have been reviewed, and all deficiencies identified either are not applicable or have been adequately resolved and are auditable. Acceptable JIOs have been submitted for equipment items not having complete qualification. The staff finds that this item has been adequately resolved and is acceptable.

3.11.4.2 Environmental Qualification Audit

On June 20 to 23, 1983, the staff, with assistance from EG&G Idaho, Inc., conducted an audit of the applicant's qualification documentation and equipment installed at the plant. Ten equipment items were reviewed to determine if the documents in the qualification files supported the qualification status determined by the applicant. The following observation was made during the audit and subsequently has been resolved or documented in further correspondence:

> In one file (Marathon terminal blocks), it was determined that insufficient attention had been given to the acceptance criteria for qualification tests and their applicability to plant-specific requirements. As a result, the applicability of all test data and acceptance criteria was reviewed by the applicant. The applicant has adequately resolved this concern by removing the terminal blocks in affected circuits and replacing them with qualified splices.

The staff finds that all concerns cited during the audit have been adequately and acceptably resolved.

3.11.5 Conclusions

The staff has reviewed the SNUPPS program for the environmental qualification of electrical equipment important to safety and safety-related mechanical equipment. The purpose of the review was to determine the adequacy of the program, including the systems selected for qualification, the environmental conditions resulting from design-basis accidents, and the methods used to demonstrate qualification. As detailed above, the staff concludes that all open items identified have been satisfactorily resolved.

The following license condition will be incorporated into the SNUPPS plants licenses: All electrical equipment within the scope of 10 CFR 50.49 shall be qualified by November 30, 1985.

On the basis of its review, the staff concludes that the applicant has demonstrated conformance with the requirements for environmental qualification as detailed in 10 CFR 50.49, the relevant parts of GDC 1 and 4, and Sections II, XI, and XVII of Appendix B to 10 CFR 50, and with the criteria specified in NUREG-0588.

4 REACTOR

4.4 Thermal-Hydraulic Design

4.4.4 Design Abnormalities

4.4.4.1 Fuel Rod Bowing

In the SER, the staff determined the magnitude of rod bow as a function of burnup using interim methods that had previously been accepted by the staff. In addition, the staff approved generic margins that could be used to offset the reduction in the departure from nucleate boiling ratio (DNBR) as a result of rod bowing.

Revision 14 to the SNUPPS FSAR provided an update to the rod bow penalty methodology. This revised methodology uses the analytical process described in WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," and the generic margins reported in the SER. For the worst case at an assembly average burnup of 33,000 megawatt days per metric ton (uranium) (MWd/MTU), the applicant calculated a penalty of less than 3%. Westinghouse-designed plants do not consider the effects of rod bow for an assembly average burnup greater than 33,000 MWd/MTU because beyond this burnup, the burndown effects preclude the fuel from achieving the limiting value of $F\Delta_H^N$. On the basis of its review of the information in the FSAR and the fact that the methods used have been previously approved by the staff, the staff has concluded that the proposed rod bow calculations are acceptable. The available thermal margin used to offset the rod bow penalty is in the bases of the Technical Specifications.

4.4.4.3 Flow Measurement Uncertainty

During reactor power operation, the reactor coolant system (RCS) flow must be verified periodically to ensure that it is no less than the acceptable limit. The RCS flow is verified through the measurement of flow through the elbow taps in the cold legs. The elbow tap flow measurements are normalized against a precision flow calorimetric measurement that will be performed at the beginning of each fuel cycle. Therefore, the overall uncertainty of the RCS flow measurement consists of the uncertainties associated with the precision flow calorimetric and the elbow tap measurements. By letter dated April 6, 1984, the applicant provided a detailed breakdown of the measurement component uncertainties associated with the flow calorimetric and the elbow tap measurements, as well as the statistical method of combining these uncertainties. The staff's review findings follow.

In determining the flow calorimetric uncertainty, several interdependent error components are combined statistically, and thus violate the independence requirement. For example, the venturi thermal expansion factor, feedwater density, and enthalpy are all dependent on the feedwater temperature; the feedwater density and steam enthalpy are both dependent on steamline pressure because the feedwater pressure is calculated from the steamline pressure; and the hot-leg and cold-leg enthalpies are both dependent on the pressurizer pressure. However, they are treated as independent quantities because the magnitudes of the uncertainties of these interdependent error components are so small (compared to the dominant error components such as the hot-leg temperature stratification uncertainty) that the use of the statistical treatment of these components has no significant effect on the final result. This conclusion has been demonstrated previously for the McGuire flow measurement uncertainty analysis (letter from E. Adensam, NRC, to Duke Power Co, Docket No. 50-369/50-370, June 28, 1983). In addition, the uncertainty values used in the analysis are the bounding conservative values that can offset the small error resulting from the statistical treatment of these interdependent error components.

The effects of drift of some of the measurement instrumentation are not included in the calorimetric measurement uncertainty analysis. In its submittal, the applicant committed to calibrate the instrumentation used in the performance of the precision heat balance 7 days before the calorimetric measurement. This calibration will be required in the Wolf Creek Technical Specifications. Therefore, neglecting the drift effect of this special test instrumentation is acceptable.

The effect of fouling as a result of crud buildup in the venturi is not taken into account in the feedwater flow measurement. The venturi fouling is a bias that will result in a higher measured feedwater flow, and, in turn, higher RCS flow than the actual values. Therefore, if the feedwater venturi is not cleaned, the effect of the fouling on the feedwater flow measurement and then on the steam generator power and RCS flow, should be determined and treated as a bias. That is to say, the error caused by venturi fouling should be added to the statistical sum of the rest of the measurement errors. However, the applicant has committed to a visual inspection of the feedwater venturi at every refueling to detect any buildup of fouling and to correct the situation before it can affect the feedwater flow measurement. The staff will ensure that this requirement is incorporated in the Wolf Creek Technical Specifications. Therefore, the staff concludes that neglecting the feedwater venturi fouling effect is acceptable.

The instrumentation uncertainties used in the analysis are based on the generic bounding values for the Westinghouse instrument. The applicant has identified the instrumentation that deviates from the Westinghouse instrumentation and provided the uncertainty value pertinent to these instruments.

The staff has performed an audit calculation based on these values of the component errors and concluded that the uncertainty associated with the precision flow calorimetric is 1.95%; the uncertainty associated with the elbow tap flow measurement is 0.74%; and the overall flow measurement uncertainty is 2.1% for the RCS flow measured by the elbow taps, which are normalized with the precision flow calorimetric. On the basis of its review, the staff has concluded that the methodology used to determine the flow measurement uncertainty for the Wolf Creek Generating Station is acceptable with the above exception. The overall RCS flow measurement uncertainty of 2.1%, rather than 2.0% as proposed by the applicant, will be incorporated in the Wolf Creek Technical Specification.

5 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.2 Overpressure Protection

Introduction

The presently approved low-temperature overprotection system (LTOPS) for Wolf Creek Unit 1 consists of two power-operated relief valves (PORVs) that have a temperature-dependent variable setpoint. In a letter dated July 20, 1984, the applicant proposed the use of the two safety-relief valves at the residual heat removal (RHR) pumps suction side as an alternative low temperature overpressure protection system.

Residual Heat Removal System (RHRS) Relief Valves for LTOPS

There is one spring-loaded relief valve at the suction side of each of the two RHRS trains. Each one of these valves has a relief capacity of 900 gpm of water at a set pressure of 450 psig. The applicant's letter of February 10, 1984, contained analyses showing that there is sufficient relief valve capacity at this setpoint to prevent exceeding 10 CFR 50 Appendix G limits in the event of an inadvertent loss of letdown flow when either one charging pump or one safety injection pump is operating at full flow. The analyses also show that the RHR relief valves will prevent exceeding the Appendix G limits in the event of a reactor coolant pump start with the steam generator secondary temperature no more than 50F° higher than the reactor coolant system (RCS) temperature.

Autoclosure Interlocks

Two motor-operated valves in series are located on the suction side of each train of the RHRS. These are operated by separate automatic interlocks to ensure both isolation valves are closed, when the RCS is operating at normal pressure, to preclude a LOCA outside the containment. The interlocks are discussed in Branch Technical Position (BTP) RSB 5-1, paragraph B.1.c (NUREG-0800), and in Section 7.6.1 of the SER. These interlocks are designed so that any single failure cannot prevent the closure of at least one of these valves in each RHRS train. However, as a result of the interlock configuration, a single failure in this automatic system can close one of these valves in each train and thereby isolate the RHRS from the RCS. Thus both the RHRS and LTOPS could be lost when they are needed.

To eliminate this possibility, the applicant proposed changing the operating procedures and Technical Specifications to require opening and removing power from one isolation valve in each train when the RHRS is in operation and being used for LTOPS. As described below, the applicant also proposed administrative controls to ensure that power is restored to the motor operators of these valves whenever the RHRS is isolated from the RCS.

Evaluation

The Wolf Creek Technical Specifications require that whenever the RCS temperature is below 368°F, one charging pump and both safety pumps be made inoperable by the removal of power. Also, the Technical Specifications require that when in this temperature range, a reactor coolant pump should not be started when its steam generator temperature is more than 50F° higher than the RCS temperature. These measures, together with the PORV setpoint, ensure that the Appendix G limits are not exceeded. On the basis of the applicant's letter of July 20, 1984, the staff agrees that with the plant in these conditions the RHRS relief valves have sufficient capacity for an LTOPS. However, the removal of power from a certain isolation valve in each RHRS train, which is necessary in order to meet the single-failure criteria for the LTOPS, violates the position stated in BTP RSB 5-1 for redundant autoclosure interlocks in the RHRS.

The staff discussed its concerns with the applicant on April 10, 1984, and in a meeting held on April 13, 1984. The applicant was asked if the isolation requirements specified in BTP RSB 5-1 would be met with the proposed removal of power. In response, the applicant stated that the staff position would probably not be met; however, supplemental administrative measures would be implemented to ensure that all RHRS suction isolation valves were closed before pressurizing the RCS above the RHRS design pressure. Furthermore, the applicant stated that long-term hardware modifications and administrative steps would be evaluated and proposed by June 1, 1985, to meet the RHRS isolation requirements of BTP RSB 5-1. If necessary, these hardware modifications would be implemented before startup, following the first refueling outage. Thus, the applicant's proposed reliance on the RHRS for LTOPS and the removal of power from a certain isolation valve in each RHRS train, which makes the RHRS suction non-isolable because of a single failure, is an interim proposal pending either equipment modifications or further justification.

By letters dated April 23 and 26, 1984, the applicant committed to the above actions, and described the supplementary administrative measures to be employed. The plant startup procedures require that at about 350°F and 400 psig, the RHRS is removed from service and placed in the emergency core cooling system (ECCS) alignment. Valve checklists require separate, independent verification that the RHRS suction isolation valves are closed and that power to the valves has been removed. Each verification requires a signature.

Conclusion

The staff concludes that with the removal of power from the RHRS isolation valves and strict administrative controls on the position of these valves, the RHRS is acceptable as an interim LTOPS. However, the proposed removal of power from the isolation valves is not acceptable as a long-term approach, and the applicant must either propose hardware modifications meeting the requirements of BTP RSB 5-1, or justify the acceptability of relying solely on administrative means throughout the life of the plant to prevent Event V concerns.

The license will be conditioned to require that the applicant submit by June 1, 1985, a description of equipment modifications to the RHRS suction isolation valves autoclosure circuitry that meet the applicable staff requirements. These modifications are to be implemented before startup, following the first refueling outage. Alternatively, by June 1, 1985, the applicant may provide acceptable justification for the reliance on administrative means alone to meet the staff's RHRS isolation requirements, or propose Technical Specification changes removing the reliance on the RHRS for LTOPS.

5.2.4 Preservice and Inservice Inspection and Testing of the Reactor Coolant Pressure Boundary

5.2.4.1 Evaluation of Compliance With 10 CFR 50.55a(g) for Wolf Creek Generating Station

The evaluation supplements conclusions in this section of NUREG-0881, which addressed the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g). The staff has reviewed the information presented in the FSAR through Revision 12 dated March 1984; the information obtained at a public meeting in Bethesda, Maryland, on November 9, 1983; letters from the applicant dated February 8 and 14, 1984; and the Preservice Inspection Program Plan through Revision 6, submitted August 28, 1984. The staff finds the selection of primary boundary welds subject to examination acceptable.

In letters dated September 6 and 19, 1984, the applicant requested relief from the ASME Code Section XI requirements that had been determined to be impractical to perform. These relief requests address the required volumetric examination of small-bore piping in the reactor coolant system, reactor pressure vessel examination, pressurizer examination, and random component and piping welds. The applicant provided supporting information pursuant to 10 CFR 50.55a(a)(2)(i). The staff evaluated the ASME Code-required examinations that the applicant determined to be impractical and, pursuant to 10 CFR 50.55a(a)(2), has allowed relief from the impractical requirements that, if implemented, would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. On the basis of the granting of relief from these specific preservice examination requirements, the staff concludes that the preservice inspection program for the Wolf Creek Generating Station meets the requirements of Section XI of the ASME Code, 1977 Edition, including addenda through Summer 1978, and, therefore, is in compliance with 10 CFR 50.55a(g)(3). The detailed evaluation supporting this conclusion is provided in Appendix J to this report.

The initial inservice inspection program has not been submitted. This program will be evaluated after the applicable ASME Code edition and addenda can be determined based on 10 CFR 50.55a(b), but before the first refueling outage, when inservice inspection begins.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

When the Wolf Creek SER was published in April 1982 (NUREG-0881) the staff determined that the requirements of the then current Appendix G of 10 CFR 50 had been met, except for the specific matter discussed in the SER. However, Appendix G was revised effective July 26, 1983, to require that the fracture toughness program meet the ASME Code edition and addenda, as permitted by 10 CFR 50.55a. Because the Wolf Creek reactor vessel was procured to the ASME Code edition and addenda as permitted by 10 CFR 50.55a, the licensee's fracture toughness program meets the requirements of the revised 10 CFR 50, Appendix G.

5.4 Component and Subsystem Design

5.4.2 Steam Generators

5.4.2.3 Secondary Water Chemistry

In the SER, the staff conditioned the Wolf Creek license to require that the secondary water chemistry monitoring and control program be carried out. Because this requirement is covered in the administration section of the Wolf Creek Technical Specifications, the staff is removing this as a license condition. This item has been added to the Technical Specification tracking list in Chapter 16.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.3 Containment Isolation System

At a March 15, 1984, meeting between the members of the staff and applicant representatives, the applicant provided information to justify operating the containment mini-purge system 5000 hours a year as opposed to the Standard Technical Specification value of 500 hours. In addition, the applicant requested that the allowable leakage rate for the mini-purge supply and exhaust isolation valves be increased.

At the meeting, the applicant presented the following points as justification for extending the mini-purge operating time limit:

(1) There is a negligible increase in risk to the public when the mini-purge system operates 5000 hours rather than 500 hours.

(2) Extended purging operation would assist in keeping occupational exposures as low as reasonably achievable.

(3) The outage cost would be approximately $2 to $3 million more for a 500-hour operation limit.

The applicant also discussed the request for an increase in the allowable leakage rate for the isolation valves and how this was tied to the Type C testing described in 10 CFR 50, Appendix J.

On the basis of its review of the information presented at the March 15, 1984, meeting and the fact that it is staff policy to limit containment purge operations to durations as brief as reasonably possible, the staff has concluded that the Technical Specification limitation on purge time should be 2000 hours a year.

For the leakage rate, the staff has concluded that the increase from 0.01 to 0.05 L_a is acceptable. This conclusion is based on the fact that the change represents a re-allocation of leakage rates, does not alter the overall allowable integrated leakage rate of 0.60 L_a, and is an acceptable criterion for assessing degradation of the resilient seals.

6.2.4 Combustible Gas Control System

The hydrogen monitoring system was originally designed to be normally open to the containment atmosphere and to automatically close on a containment isolation Phase A signal, with manual operation of the system beginning about 6 hours after the onset of a loss-of-coolant accident (LOCA). Now, however, the system is normally closed to the containment atmosphere and is designed with the capability to obtain an accurate sample 30 minutes after the initiation of safety injection. The staff has reviewed this design change and finds it acceptable because the hydrogen monitoring system will still meet the requirements of TMI Action Plan Items II.E.4.2 and II.F.1, Attachment 6.

6.2.6 Containment Leakage Testing

Containment Air Lock Surveillance

By letter dated November 6, 1984, the applicant requested an exemption from the requirement of paragraph III.D.2(b)(ii) of Appendix J to 10 CFR 50. Paragraph III.D.2(b)(ii) of Appendix J states: "Air locks opened during periods when containment integrity is not required by the plant's Technical Specifications shall be tested at the end of such periods at not less than P_a." The above Appendix J requirement would require a full-pressure air lock test after each and every shutdown regardless of the purpose of the shutdown. In place of this requirement, the applicant proposes to perform a full-pressure air lock test only when maintenance is performed on the air lock that could affect the air lock sealing capability. This proposed change requires an exemption from the requirements of Appendix J to 10 CFR 50. The staff's evaluation of this exemption request follows.

Whenever the plant is in cold shutdown (Mode 5) or refueling (Mode 6), containment integrity is not required. However, if an air lock is opened during Modes 5 and 6, paragraph III.D.2(b)(ii) of Appendix J requires that an overall air lock leakage test at not less than P_a be conducted before plant heatup and startup (i.e., entering Mode 4). The existing air lock doors are so designed that a full-pressure test (P_a, 48.0 psig) of an entire air lock can only be performed after strong backs (structural bracing) have been installed on the inner door. Strong backs are needed because the pressure exerted on the inner door during the test is in a direction opposite to that of the accident pressure direction. Installing strong backs, performing the test, and removing strong backs require at least 6 hours per air lock, during which access through the air lock is prohibited.

If the periodic 6-month test of paragraph III.D.2(b)(i) of Appendix J and the test required by paragraph III.D.2(b)(iii) of Appendix J are current, no maintance has been performed on the air lock, and the air lock is properly sealed, there should be no reason to expect the air lock to leak excessively just because it has been opened in Mode 5 or Mode 6.

Accordingly, the staff concludes that the applicant must still meet its proposed approach of substituting the seal leakage test of paragraph III.D.2(b)(iii) for the full-pressure test of paragraph III.D.2(b)(ii) of Appendix J.

Therefore, an exemption from this requirement [10 CFR 50, Appendix J, paragraph III.D.2(b)(ii)] is justified and acceptable for Wolf Creek Unit 1, and the applicant's proposed changes to the Technical Specifications concerning this subject are acceptable.

6.6 Inservice Inspection of Class 2 and 3 Components

This section was prepared with the technical assistance of Department of Energy (DOE) contractors from the Idaho National Engineering Laboratory (INEL).

6.6.1 Evaluation of Compliance With 10 CFR 50.55a(g)

This evaluation supplements conclusions in this section of NUREG-0881, which addressed the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g). The staff has reviewed the information presented in the FSAR through Revision 12, dated March 30, 1984; the information obtained at a public meeting at Bethesda, Maryland, on November 9, 1983; a letter from the applicant dated February 8, 1984; the Preservice Inspection Program Plan through Revision 6, submitted August 28, 1984, and finds the selection of Class 2 and 3 components subject to examination acceptable.

In letters dated September 6 and September 19, 1984, the applicant requested relief from the ASME Code Section XI requirements that were determined to be impractical. These relief requests address the required volumetric examination of random component and piping welds, steam generator welds, and the visual examination of ASME Class 3 supports. The applicant provided supporting information pursuant to 10 CFR 50.55a(a)(2)(i). The staff evaluated the ASME Code-required examinations that the applicant determined to be impractical and, pursuant to 10 CFR 50.55a(a)(2), has allowed relief from the impractical requirements that, if implemented, would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. On the basis of granting relief from these specific preservice examination requirements, the staff concludes that the preservice inspection program for Wolf Creek Generating Station meets the requirements of Section XI of the ASME Code, 1977 Edition, including addenda through Summer 1978, and, therefore, is in compliance with 10 CFR 50.55a(g)(3). The detailed evaluation supporting this conclusion is provided in Appendix J to this report.

The initial inservice inspection program has not been submitted. This program will be evaluated after the applicable ASME Code edition and addenda can be determined based on 10 CFR 50.55a(b), but before the first refueling outage when inservice inspection begins.

7 INSTRUMENTATION AND CONTROLS

7.2 Reactor Trip System

7.2.2 Resolution of Issues

7.2.2.1 Sensor Time Response Testing

The SNUPPS SER includes a license condition that requires the periodic submittal of sensor response time test results to determine the adequacy of the computer-based, process noise method of testing. Since the SER was issued, the applicant has submitted information by letters (February 1 and 23, 1984) stating that the noise method will not be the only means of sensor response time testing. The applicant is proposing to use (as the primary means) the direct methods endorsed by NUREG-0809 and Instrument Society of America (ISA) draft Standard 67.06, which have been proven through industry practice. Specifically, temperature sensors will be tested using the loop-current step-response method in which pressure sensors will be tested using simulated pressure ramps. The noise analysis method will be a backup only.

The staff finds this approach acceptable and concludes that, because the primary response time testing will be performed using direct methods, this item is considered resolved and License Condition B(4) is removed.

7.2.2.8 Environmental Errors for Reactor Trip Setpoints

By letter dated May 16, 1984, the staff requested that the applicant provide information before operation above 5% power or justify the omission of environmental errors for setpoint calculations related to the diverse trip functions or to incorporate appropriate environmental errors.

In Revision 15 to the SNUPPS FSAR, dated June 26, 1984, SNUPPS stated that for each event that could result in adverse environmental conditions, there is at least one actuation function available as a backup that is not located in the vicinity of the accident. Thus, it is not necessary to include environmental errors for setpoint calculations associated with such backup trips. The applicant did note, however, that if a trip function is diverse for one event but primary for another, the setpoint for both cases is based on the primary actuation function. Further, if a trip function is used in the safety analysis as a primary trip for an event, the actuation setpoint is based on the requirements of that event (i.e., if that event includes adverse environmental conditions in the vicinity of the sensor/transmitter, an environmental allowance is included). Also, the applicant reiterated that no credit is taken for the functioning of the diverse trip functions in the plant's FSAR accident analyses.

On the basis of the above discussion, the staff concludes that the applicant has provided sufficient information to justify the omission of environmental errors for setpoint calculations associated with the diverse trip functions. Thus, the staff concludes, with reasonable assurance, that the facility can be operated without undue risk to the health and safety of the public. This issue is considered resolved.

7.3 Engineered Safety Features Actuation Systems

7.3.1 Description

7.3.1.5 Feedwater Line Isolation Actuation

In Revision 14 to the SNUPPS FSAR, the applicant updated the Section 7.3 drawings to include main feedwater isolation on low-low level in any steam generator. The use of this signal for main feedwater isolation has been reviewed and approved by the staff as described in Section 10.4.7 of this SER supplement. The need for this design feature is also discussed in Section 10.4.7.

The steam generator low-low level signal is generated from the Westinghouse solid-state protection system and is the same signal that has been reviewed and approved by the staff for the initiation of reactor trip (SER Section 7.3.1.10). This signal is consistent with the other main feedwater isolation signals in that, upon its receipt, the main feedwater isolation valves and other valves associated with the main feedwater lines are closed.

On the basis of the above discussion, the staff finds this design function acceptable.

7.4 Systems Required for Safe Shutdown

7.4.2 Remote Shutdown Capability

During review of the SNUPPS plant design before the SER was issued, it was the staff's understanding that the auxiliary shutdown panel and all essential instrumentation (indicators) and controls mounted on it were redundant, safety related, and seismically qualified. After the SER was issued, the staff became aware that design changes for remote shutdown were being made as a result of the ongoing SNUPPS Appendix R (10 CFR 50) review. Thus, the staff forwarded letters (November 3, 1982, and October 5, 1983) to request additional information on this matter. The letters provided guidance on safe shutdown from outside the control room and requested that the applicant show how the SNUPPS design will comply with the staff's guidance after the Appendix R modifications are made.

The applicant responded by letters (December 23, 1982, and October 27, 1983) and confirmed that the SNUPPS design provides the capability to achieve and maintain hot standby conditions using only redundant safety-related controls and indication. This safety-related equipment is seismically and environmentally qualified for the conditions to which it will be exposed. The SNUPPS plants will have redundant safety-related capability to reach and maintain cold shutdown from outside the control room through the use of suitable procedures. Also, the applicant has verified that, in a nonaccident or noncontrol-room fire situation, the transfer of plant control to the auxiliary shutdown panel and local control stations outside the control room would not disable any automatic actuation of engineered safety features equipment or change the operating status of equipment.

On the basis of the above discussion and on a review of the applicant's additional information, the staff concludes that the SNUPPS design (after Appendix R modifications) complies with the staff's guidance related to remote shutdown from outside the control room. The acceptability of the remote shutdown station designs given a fire is determined as outlined in SRP 9.5.1 and is addressed in Section 9.5.1.5.

7.4.3 Resolution of Issues

7.4.3.1 Capability for Safe Shutdown Following Loss of a Bus Supplying Power to Instruments and Controls

The SNUPPS SER states that the applicant should provide confirmatory information related to IE Bulletin 79-27 guidelines. The applicant responded by letter (February 2, 1984). The staff review of this information and discussions with the applicant showed that all Class 1E and non-Class 1E ac and dc instrument buses that could affect the ability to achieve a cold shutdown condition followed the guidelines of IE Bulletin 79-27. The applicant stated that the SNUPPS design provides the capability to achieve cold shutdown using only Class 1E equipment. Because of this capability, the control room operators have the necessary redundant Class 1E instrumentation and control systems available to reach cold shutdown regardless of the loss of any Class 1E or non-Class 1E instrument bus. The applicant has stated that all buses in the SNUPPS plants that supply power to instrument systems are provided with alarms in the control room that indicate the loss of a particular bus. The applicant has developed procedural guidelines that incorporate required safe shutdown methods to address the situation in which the initiating event is a loss of instrument bus. During rereview of IE Circular 79-02, which included both Class 1E and non-Class 1E inverter-supplied instrumentation and control buses, the applicant concluded that no modifications were necessary.

The staff has concluded that the applicant's February 2, 1984, letter provides satisfactory confirmatory information. Therefore, Confirmatory Item B(9) is considered resolved.

7.4.3.2 Operator Actions Required To Maintain Safe Shutdown From Outside the Control Room

The SNUPPS SER notes that the applicant committed to revise the FSAR to confirm that the design provides for maintaining extended (greater than 24 hours) hot standby conditions from outside the control room.

In FSAR Revision 11 (March 1983), the applicant satisfactorily confirmed this capability of the design. To maintain hot standby after 24 hours, the addition of boron would be necessary. The FSAR states that boration can be accomplished from outside the control room using only redundant, safety-related equipment. This would include the operation of a centrifugal charging pump, taking suction from the refueling water storage tank (RWST), and charging into the reactor coolant system through either the normal charging path or the boron injection tank flow path. The FSAR describes the various local actions required after 24 hours.

On the basis of its review of the information provided in FSAR Revision 11, the staff considers Confirmatory Item B(10) resolved.

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 a confirmatory item, the SNUPPS SER required that the FSAR be revised to describe the reactor coolant temperature indication at the auxiliary shutdown panel. FSAR Revision 12 shows that both hot-leg and cold-leg reactor coolant temperatures will be indicated on the auxiliary shutdown panels (118A and 118B). Safety-related hot-leg and cold-leg temperature indication is provided on panel 118B; the redundant indication on panel 118A for these parameters is not safety related. The staff verified the installation of the remote shutdown panel indication and controls during a site visit in October 1983.

After the SNUPPS SER was issued, the staff developed guidance on remote shutdown from outside the control room. This guidance requires that the instrumentation and controls for remote hot and cold shutdown remain functional during and following a seismic event. At the request of the staff, the applicant has verified by letter (February 23, 1984) that the non-safety-related hot-leg and cold-leg indicators are identical to the safety-related indicators provided on panel 118B that are seismically and environmentally qualified. The circuitry for the safety-related and non-safety-related indicators will be routed in separate raceways (Class 1E and non-Class 1E, respectively). Therefore, adequate electrical and physical independence is maintained between the redundant indication circuits. The staff concludes that this satisfies the requirements for remote safe shutdown equipment.

FSAR Revision 10 verified that no single failure can inhibit the indication at the auxiliary shutdown panel for reactor coolant hot-leg and cold-leg temperatures associated with operable steam generators (i.e., having an auxiliary feedwater supply and an operable power-associated atmospheric dump valve).

On the basis of its review of the additional information, the staff considers Confirmatory Item B(11) resolved.

7.5.2.3 Postaccident Monitoring

SNUPPS SSER 2 states that a schedule for implementation of Regulatory Guide (RG) 1.97, Revision 2, would be established consistent with Generic Letter 82-33. In response to the generic letter, the applicant submitted information (April 15, 1983). This letter notes that Appendix 7A to the SNUPPS FSAR provides a comprehensive comparison of the SNUPPS design to the recommendations of RG 1.97, Revision 2. The applicant has committed, with the few exceptions described below, to install and have operable all of the instrumentation and design features described in FSAR Appendix 7A before fuel loading. The following will not be operable before fuel loading:

(1) source range instrumentation (qualified to postaccident environmental conditions)

(2) reactor vessel water level instrumentation

(3) subcooling monitor

(4) radiation monitors for releases from steam generator safety-relief valves or atmospheric dump valves

(5) auxiliary feedwater pump turbine exhaust monitor

Appendix 7A is currently under staff review to determine compliance of the SNUPPS design to the guidance of RG 1.97, Revision 2. Until the staff completes its review of the SNUPPS design for compliance to RG 1.97, Revision 2 recommendations, a license condition will be imposed requiring the satisfactory resolution of all the review findings and that the five exceptions described above be implemented before startup, after the first refueling outage.

7.6 Interlock Systems Important to Safety

7.6.7 Resolution of Issues

7.6.7.2 Volume Control Tank Level Control and Protection Interaction

The Wolf Creek SER stated that the volume control tank (VCT) level control function will be separated from the charging pump protection function so that a single random failure in the control system will not lead to a loss of redundancy in the high-head safety-injection system. The staff found this acceptable, but required that the applicant formally revise the FSAR to confirm the above design.

The applicant has provided a description of the design in SNUPPS FSAR Revision 10. The VCT supply line to the charging pumps contains two normally open, motor-operated valves with one powered from Train A and the other from Train B. Each valve has its own train-associated VCT level protection channel, and each receives a signal to close on a low-low level signal or a safety-injection (SI) signal. An interlock is provided to prevent the VCT supply line valve from closing in the event that its train-associated refueling water storage tank (RWST) valve is closed. The FSAR states that to avoid any interface, an independent, control-grade level channel is used for VCT level control and is electrically separated from the VCT level protection channels associated with the VCT supply valves. The control-grade level channel shares the lower level tap with one of the redundant VCT level protection channels. The FSAR states that a rupture of this tap would result in an "empty" indication by the affected protection channel and subsequent switchover to the RWST. There is complete physical separation among the upper taps. The staff finds this acceptable.

On the basis of its review of the FSAR information and the discussion above, the staff concludes that a single failure within the VCT level control system will not lead to a loss of redundancy within the protection system. Therefore, the staff finds the design acceptable and considers Confirmatory Item B(12) to be satisfactorily resolved.

7.6.7.3 Boron Dilution Control

SNUPPS SER Section 7.6.7.3 states that the applicant has committed to incorporate the same design for terminating boron dilution as implemented for the Comanche Peak plant (Docket No. 50-445). The staff found this acceptable but required that the applicant (1) provide a discussion in the FSAR to confirm the design associated with the automatic control of inadvertent boron dilution and (2) notify the staff of completion of installation of this design. By FSAR Revision 13, the applicant has provided information describing this design. FSAR Section 7.6.12 confirms that the design is identical to that reviewed and approved by the staff for Comanche Peak Units 1 and 2, including the seismic and environmental qualification of the instrumentation. The FSAR drawings have been revised to include the new design. The installation of this design will be verified by NRC Regional Office personnel as part of their pre-fuel-load inspection.

FSAR Section 15.4.6 was also revised (Revision 13) on the basis of the new automatic boron-dilution control system, and the staff's evaluation of it is discussed in Section 15.2.3.1 of this supplement.

On the basis of the above discussion, including the final review and acceptance of the analysis provided in Revision 13 of FSAR Section 15.4.6, the staff considers the proposed design to be adequately confirmed and hence this portion of Confirmatory Item B(13) is closed.

7.7 Control Systems

7.7.11 Resolution of Issues

7.7.11.3 Environmental Qualification of Control Systems

As required by the SER (Section 7.7.11.3), the applicant has submitted a plant-specific analysis by letters dated February 2 and October 27, 1983. This analysis confirms that the effects of a high-energy line break on the rod control system are appropriately bounded. The applicant's letters state that a steamline break in the turbine building will affect the turbine impulse stage transmitters that provide control signals to the reactor rod control system. This may cause the control rods to withdraw at the initiation of the transient, resulting in an increase in reactor power and core heat flux. This increase in core power will generate an overpower ΔT reactor trip signal, which will terminate the most adverse part of the transient.

In the letters, the applicant states that the rod cluster control assembly bank withdrawal at power and large main steamline break transients analyzed in FSAR Chapter 15 bound the consequences of the event postulated above. Therefore, on the basis of its review of the applicant's responses, the staff concludes that the applicant has adequately confirmed that high-energy line-break effects on the rod control system are appropriately bounded. Therefore, Confirmatory Item B(14) is resolved.

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 buses, the staff, as documented in the SER, reviewed and found acceptable the applicant's description and analysis of compliance to GDC 5, 17, and 18 that are included in FSAR Revision 7. The routing of offsite circuits also was reviewed as part of the confirmatory site visit during the week of April 5 to 8, 1983. As a result of this site visit, a new concern was identified and reported in Wolf Creek SSER 3.

Two engineered safety feature (ESF) transformers, XNB01 and XNB02, shown on Figure 8.3-1 of the FSAR, are separated by a 3-hour-rated fire wall, as documented in FSAR Revision 7. During the site visit, the staff expressed the concern that failure of one of the two transformers could cause failure of the other transformer and/or offsite circuit. For example, an oil fire at one transformer could overflow (with water sprinklers on) and damage or cause failure of the other division of offsite power. Other examples include (1) smoke and heat being drawn into the cooling system by fans on the transformer associated with the other offsite circuit and (2) heat causing failure of the other offsite circuit located above and 5 to 10 ft to the side of the transformer.

By letter dated August 10, 1983, the applicant responded to this staff concern. In summary, the applicant stated that the ESF transformers, which are not safety related and are not required to safely shut down the plant, are adequately designed in accordance with the requirements of GDC 17. However, the staff disagreed because the ESF transformers are considered important to safety, as defined by GDC 17.

GDC 17 requires, in part, that these two ESF transformers be located in a way that minimizes to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions.

The applicant presented additional information by letter dated February 2, 1984. On the basis of this additional information, the staff concludes that the physical separation, barriers, and transformer pit design, in conjunction with fire detection and suppression systems, provide reasonable assurance that both offsite circuits will not fail simultaneously. The design therefore, meets GDC 17 and is acceptable. This resolves Confirmatory Item B(35). Surveillance requirements for the operability of the fire detection and suppression systems will be included in the Technical Specifications.

8.2.2.3 Sequencing of Loads on the Offsite Power System

In the SER, the staff indicated that the SNUPPS reliability study for the solid-state load sequencer would be verified during a confirmatory site visit. At a September 30, 1982, meeting, the applicant presented two reports (J-104-0221-05 and J-104-0256-06) for staff review (Consolidated Controls Corporation, October 10,1978, and September 14, 1981). As reported in Wolf Creek SSER 3, on the basis of these reports, the staff was unable to conclude that the reliability of the offsite power to the Class 1E buses will not be compromised by using the same load sequencer to sequence loads on both the onsite and offsite power sources.

The applicant, by letter dated August 10, 1983, provided additional information and the results of analysis addressing the SNUPPS design for the load sequencer. On the basis of the additional information and the automatic and manual test capability of the sequencers, the staff considers Confirmatory Item B(22) acceptably resolved. Surveillance requirements for the operability of the load sequencer logic will be included in the Technical Specifications.

8.3 Onsite Emergency Power Systems

8.3.1 Onsite AC Power System Compliance With GDC 17

8.3.1.2 Low and/or Degraded Grid Voltage

In Wolf Creek SSER 3, the staff indicated that the Wolf Creek voltage drop analysis and testing would be verified. By letter dated January 31, 1984, the applicant provided a description of the methodology used for the analysis and its results. On the basis of the description and results, the staff concludes that for all plant loading configurations, the voltage rating and requirements for the Class 1E loads will not be exceeded; the design meets the guidelines of position 3 of the staff position defined in SER Section 8.3.1.2 and is acceptable. Therefore, Confirmatory Item B(19) is resolved.

Field measurements and confirmatory testing to verify the voltage drop analysis are being performed by the applicant. If any problems are identified during the field verification, the staff will ensure that corrective actions are taken; however, no problems are anticipated.

8.3.1.6 Electrical Independence Between Local and Control Room Panels

In the SER, the staff indicated that the SNUPPS design for isolation of diesel generator control circuits between the control room and remote panels would be verified during the staff's confirmatory site visit. The confirmatory site visit was held during the week of April 5 to 9, 1983. As a result of the site visit, the following new concerns were identified and reported in SSER 3:

(1) Redundant load sequencers are located in the same area and are subject to the common failure effects of the design-basis event exposure fire.

    In regard to this concern, 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 so that ac power cannot be re-established in a short period of time at the remote panel. The results of the staff evaluation for this item are reported in Section 9.5.1.5.

(2) The output relays of the redundant load sequencers are mounted back to back in a common panel.

In regard to this concern, it is the staff position, in accordance with Section 4.6 of IEEE 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.

The applicant responded to this concern by letter dated August 10, 1983.

The output relays of each load group load shedder emergency load sequencer (LSELS) are mounted on their own metal subpanel. Each subpanel is mounted on a C-channel that, in turn, is bolted to the sides of the panel bay. This arrangement provides an approximate 2-in. air space between the barriers and full side-to-side isolation of the two output relay sections. External cables for the two relay sections enter the bay from the top and bottom, respectively. In the area where the cables enter the cabinet, a 6-in. train separation is provided for the cables. No postulated failure in one output relay section can propagate to the other output relay section.

The staff agrees with the applicant's assessment and concludes that there is reasonable assurance that failure in one output relay section will not propagate to the other relay section.

8.3.3 Common Electrical Features and Requirements

8.3.3.1 Compliance With GDC 2 and 4

8.3.3.1.2 Thermal-Overload Protection Bypass

The staff indicated in the SER that the setpoints for the thermal overload protection, their required margin, and the frequency for periodic tests would be included in the Technical Specifications. By Revision 10 to the FSAR, the applicant indicated that the thermal overload relay trip contacts for all Class 1E valves will be permanently bypassed with jumpers before fuel loading. The staff concludes that the permanent bypass resolves the original SER concern relating to inadvertent operation of thermal overloads under accident conditions; thus, Technical Specifications for this item are no longer required.

The staff notes, however, that it is not the intent of RG 1.106 to totally eliminate the use of thermal overloads on motor-operated valves. RG 1.106 is intended to ensure that, under accident conditions, the valve will not be hindered from performing its safety function by a spurious trip of its thermal overload protective circuits. For the majority of valve operations such as during valve test or operation during nonaccident conditions, the use of thermal overload protective circuits is a prudent operational practice to minimize motor damage as a result of overload. This is only a staff observation, and it is not considered an open or confirmatory item.

8.3.3.6 Compliance With GDC 50

Compliance With Position 1 of Regulatory Guide 1.63

In the SER, the staff indicated that the results of an applicant re-evaluation program of electrical penetration assemblies would be reviewed as part of the staff confirmatory site visit to verify that they can withstand, without seal failure, the total range of available time-current characteristics, assuming a single failure of any overcurrent protective device. By letter dated January 31, 1984, the applicant provided the results of the re-evaluation program, which included representative time-current curves for the containment penetrations. On the basis of these results, the staff concludes that the design meets position 1 of RG 1.63 and the requirements of GDC 50. It is, therefore, acceptable. This resolves Confirmatory Item B(25).

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.4 Fuel Handling System

9.1.4.1 Introduction

As a result of Generic Task A-36, "Control of Heavy Loads Near Spent Fuel," NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," was developed. Following the issuance of NUREG-0612, a letter dated December 22, 1980, was sent to all licensees of operating plants, applicants for operating licenses, and holders of construction permits requesting that they prepare responses indicating the degree of compliance with the guidelines of NUREG-0612. The responses were to be made in two stages. The first response (Phase I) was to identify the load-handling equipment within the scope of NUREG-0612 and to describe the associated general load-handling operations such as safe load paths; procedures; operating training; special and general-purpose lifting devices; the maintenance, testing, and repair of equipment; and the handling equipment specifications. The second response (Phase II) was to show that either single-failure-proof equipment was not needed or that single-failure-proof equipment had been provided. In the following paragraphs, the staff presents its evaluation of both phases.

9.1.4.2 NRC Review and Evaluation

The staff and its consultant, EG&G Idaho, Inc. (EG&G), have reviewed the SNUPPS submittals for Wolf Creek. As a result of its review, EG&G has issued a Technical Evaluation Report (TER) for each phase; these reports are reproduced as Appendices K and L to this report. The staff has reviewed the TERs and concurs with EG&G's findings that the guidelines in NUREG-0612 Sections 5.1.1 and 5.3 for Phase I and Sections 5.1.2, 5.1.3, 5.1.5, and 5.1.6 for Phase II have been satisfied. For guidelines 2, 3, and 6 of Section 5.1.1, the procedures for load handling, operator training and testing, and maintenance and inspections of cranes will be implemented before initial fuel loading. Therefore, the staff concludes that Phases I and II are acceptable.

Because the completion of Phases I and II more than satisfies the interim protection measures described in Section 5.3 of NUREG-0612, the requirement in SER Section 9.1.4 that the interim protection measures should be satisfied before fuel loading is met.

9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems

9.4.1 Control Room Area Ventilation System (Control Building HVAC System)

The SER stated that redundant smoke detectors in the intake duct automatically isolated the control room ventilation system from the outside air intake. However, although the standard SNUPPS design provides for automatic isolation upon detection of high radiation or chlorine as evaluated in the SER, manual isolation is relied on for smoke detection. SRP 9.4.1 does not require automatic isolation of the control room air intakes on the detection of smoke for most sites. Only in special cases, such as high probability of aircraft crashes, is automatic isolation required for smoke detection. Because the latter is not the case for Wolf Creek, manual isolation of the air intake upon smoke detection is acceptable. Therefore the staff concludes that the control room area ventilation system is in accordance with SRP 9.4.1.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection

The staff has continued its review of the Wolf Creek fire protection program on the basis of a February 7, 1984, site audit and information provided in the applicant by letters dated February 1 and 24, 1984. The results of the review are discussed below. In addition to finding that the applicant's program conforms to the staff's guidelines, the staff also approved three deviations from the guidelines. These areas of approved deviation, which also are discussed below, are

(1) penetration seals acceptance criteria [Branch Technical Position (BTP) CMEB 9.5.1, Section C.5]

(2) unrated, missile-resistant doors (BTP CMEB 9.5.1, Section C.5.a)

(3) fire detection power supplies (BTP CMEB 9.5.1, Section C.6.a)

9.5.1.1 Fire Protection Systems Description and Evaluation

Water Supply System

The water supply system consists of two fire pumps separately connected to a buried, 12-in. pipe loop around the plant. There are two 100%-capacity fire pumps. One pump is driven by an electric motor and the other is driven by a diesel engine. The fire pumps are located in a circulating water screenhouse; the electric-motor-driven fire pump is separated by a fire-rated wall from the diesel pump. The fire pump and controllers are Underwriter's Laboratory listed. Controllers and pumps will be installed and tested in accordance with National Fire Protection Association Standard 20 (NFPA 20).

A separate jockey pump maintains the yard fire main pressure. If the fire main pressure drops, the electric-motor-driven fire pump will automatically start. The diesel-engine-driven fire pump will start automatically if the pressure drops to below the settings of the electric pump. Separate audible and visual alarms are provided in the control room for each pump to monitor pump operation, drive motor availability, power failure, and failure of a fire pump to start.

The pumps take suction from a common wet pit sump in the circulating water screenhouse. Two traveling water screens and bar grill are located at the inlet to the sump serving the fire pumps. The greatest water demand for the fixed fire suppression systems is 2300 gpm that, coupled with 1000 gpm for hose streams, creates a total water demand of 3300 gpm at a residual pressure of 80 psig. The staff finds that the water supply system can deliver the required water demand with one pump out of service.

By letter dated February 24, 1984, the applicant committed to either electrically supervise all essential valves in the fire protection water supply system or to lock them in the open position under a periodic visual supervision program conforming to the Standard Technical Specifications.

On the basis of this commitment, the staff concludes that the fire protection water supply system will meet Section C.5.a of BTP CMEB 9.5-1 (Section III.A of Appendix R to 10 CR 50) and is, therefore, acceptable.

Sprinkler and Standpipe System

The SER states that the automatic sprinkler systems would be designed to the recommendations of NFPA 13. During its site visit, the staff noted that in some corridor areas (e.g., auxiliary building corridor, elevation 1974 ft, west side), the sprinkler heads are located at the ceiling and there are a large number of cable trays, conduits, pipes, and vent ducts beneath the sprinkler heads. These obstructions may render the sprinkler system ineffective against a floor-level exposure fire, and are not in accordance with NFPA 13, which is recommended by Section C.6.c of BTP CMEB 9.5-1.

By letter dated February 24, 1984, the applicant committed to perform the following modifications by October 1984:

(1) Additional sprinkler heads will be added in the auxiliary building on the 2000-ft elevation, west corridor (three-tray area), and the 2026-ft elevation, north end of east corridor, to protect against postulated fires in transient combustibles.

(2) Sprinkler heads on the 1974-ft elevation of the auxiliary building west corridor that are partially obstructed by structural steel beams will be lowered to avoid spray obstructions.

On the basis of this commitment, the staff concludes that the sprinkler system will meet the guidelines in Section C.6.c of BTP CMEB 9.5-1, and is therefore, acceptable.

Fire Detection System

The SER states that the plant fire detection system is installed in accordance with NFPA 72D. During its site vist, the staff noted that the backup power supply may not meet the recommendations of NFPA 72D. The applicant was unable to show compliance, and verbally agreed to prepare an analysis showing how the existing primary/backup power supply circuitry compares with the requirements of NFPA 72D.

By letter dated February 1, 1984, the applicant provided the comparison. The applicant's comparison indicated that the primary and secondary power supplies comply with the provision of NFPA 72D. In the event of loss of power to the remote panels, loss of automatic activation of some pre-action sprinklers would occur. Because the pre-action systems are continuously supervised, any loss of power would be alarmed in the control room. The Plant Technical Specifications would then require the establishment of a continuous fire watch. Because of the fire watch and the fact that the sprinkler systems remain operable manually, the staff finds this to be an acceptable deviation from its guidelines. On the basis of its review, the staff concludes that the fire detection system power supply is an acceptable deviation from staff guidelines in Section C.6.a of BTP CMEB 9.5-1, and is, therefore, acceptable.

9.5.1.2 Other Items Related to Fire Protection Programs

Fire Barriers and Fire Barrier Penetrations

Where safe shutdown equipment is enclosed by a fire barrier, all walls, ceilings, floors, and associated penetrations that enclose the equipment have a minimum fire rating of 3 hours with the following exceptions: 1-1/2-hour elevator doors; pressure, watertight, and missile-resistant doors; and equipment hatches in the auxiliary building. For fire areas that do not have a 3-hour-fire-rated assembly because of the installation of these doors, each area was evaluated with respect to its fuel load, fire suppression and detection systems, and proximity to safe shutdown equipment to determine if the fire-rated assemblies provided are adequate for the areas affected and meet the guidelines in Section D.1.j of Appendix A to BTP ASB 9.5-1. On the basis of this evaluation, the staff finds the above fire barriers for these areas acceptable.

The applicant has agreed to provide 3-hour-rated designs for all fire penetration seals used in the penetration cable trays, conduits, and piping that pass the penetration qualification tests, including the time-temperature exposure fire curve specified by Standard E-119, "Fire Test of Building Construction and Materials," of the American Society for Testing and Materials (ASTM E-119).

By letter dated February 1, 1984, the applicant stated that the acceptance criterion for the penetration qualification test was in excess of the 325°F maximum temperature permitted on the unexposed side by ASTM E-119. The applicant stated that the acceptance criterion used was a maximum temperature rise on the unexposed surface of the fire stop of 325°F above ambient temperature. In addition, at no time during the test period did any visible flaming occur on the unexposed side of the test assembly, and no openings developed that permitted the hose stream test to penetrate the seals.

Although the penetration seals do not meet the specific ASTM E-119 temperature rise limitations, the test results showed that fire would not spread to the unexposed side of a protected fire barrier during a 3-hour test period. Few if any areas in the plant contain a 3-hour combustible loading. Therefore, the staff has reasonable assurance that the integrity and temperature transmission through the penetration assembly will not affect the capability to achieve and maintain safe shutdown considering the effects of a fire involving fixed and potential transient combustibles in the plant.

By letter dated February 24, 1984, the applicant committed to protect cable tray or conduit supports to achieve the same rating as the protected cable tray or conduit. The staff finds this acceptable.

On the basis of its evaluation, the staff concludes that the protection provided for fire barriers and fire barrier penetrations is an acceptable deviation from the guidelines in Section C.5 of BTP CMEB 9.5-1, and is, therefore, acceptable.

9.5.1.3 Fire Protection Systems Description and Evaluation

Gaseous Fire Suppression Systems

In the SER, the staff evaluated the Halon fire suppression system in the concealed trench in the control room floor. By letter dated January 13, 1985, the applicant requested approval for a design change for this system relating to the provision of an "extended discharge" capability for the system.

A Halon total flooding system is used as the primary extinguishing agent in the engineered safety feature (ESF) switchgear rooms, nonvital switchgear and transformer rooms, control room cable trenches, and switchgear rooms. The systems are designed to produce a 5 to 10% Halon concentration with a soaking time of 10 minutes. The systems are activated by cross-zoned ionization detectors.

By letter dated January 13, 1985, the applicant indicated that the results of preoperational testing of the Halon system in the control room cable trenches revealed that the design concentration of Halon gas was not being met for a full 10 minutes at the top of vertical cable chase extending a few feet below the floor of the upper cable spreading room.

The applicant modified the system in the control room so as to achieve an automatic timed initiation of the backup Halon storage bank after the discharge of the gas in the primary storage bank. This extends the soak time to about 7 minutes with concentration in excess of 5%. This modification provides reasonable assurance that a potential fire in this area will be controlled pending arrival of the plant fire brigade and eventual fire extinguishment using manual fire fighting equipment. This is an acceptable deviation from staff guidelines.

Except for the above deviation, the Halon suppression systems are all installed in accordance with the requirements of NFPA 12A, "Standard on Halogenated Fire Extinguishing Agent Systems - Halon 1301."

The staff has reviewed the concentration, soak times, and the design criteria for the Halon fire suppression systems. On the basis of its evaluation, the staff concludes that with the accepted deviation for the control room cable trench Halon systems, the gaseous fire suppression systems meet the guidelines.

9.5.1.4 Fire Protection for Specific Areas

Control Room

The SER states that ionization-type smoke detectors would be installed in all control room cabinets and consoles containing redundant equipment. During its site visit, the staff noted that smoke detectors are not provided for safety-related cabinets as stipulated by Section C.7.b of BTP CMEB 9.5-1.

By letter dated February 1, 1984, the applicant committed to provide detectors in the control room cabinets containing redundant safe-shutdown equipment by fuel load. The staff finds this acceptable.

In the rear of the control room complex, smoke detection is provided at the ceiling level. During its site visits, the staff was concerned that because of the ceiling height, there could be a substantial time delay in detecting an incipient fire.

By letter dated February 1, 1984, the applicant committed to provide a duct detector in the control room HVAC exhaust duct by October 1, 1984. The duct detector will provide enhanced detection capability and will compensate for the lack of low level detectors, because the HVAC exhaust inlets are near the floor level. On the basis of these commitments, the staff finds that the detection for the control room will meet the guidelines in Section C.7.a of BTP CMEB 9.5-1, and is, therefore, acceptable.

Component Cooling Pumps

The component cooling water pumps are located on the 2026-ft elevation of the auxiliary building. Partial sprinkler systems are provided for the corridor area around the pumps; however, there is an area between the pumps that does not have sprinkler protection and that contains intervening combustibles [balance of plant (BOP) cable trays]. This configuration is not in accordance with Section C.5.b of BTP CMEB 9.5-1. The applicant, by letters dated February 1 and 24, 1984, committed to provide fire stops in the intervening cable trays adjacent to one of the sprinklered zones by October 1, 1984. Because of the nature and configuration of combustibles in this area, the fire stops would effectively prevent a fire from spreading to redundant trains. On the basis of this commitment, the staff finds that the protection provided for the component cooling water pumps meets the guidelines in Section C.5.b of BTP CMEB 9.5-1, and is, therefore, acceptable.

Emergency Diesel Generator Rooms

The SER states that sprinkler systems would be installed in accordance with NFPA 13. During its site visit, the staff noted that a pre-action sprinkler system is provided for the protection of the diesel generators. A large vent duct passes directly beneath many of the sprinkler heads. The sprinkler piping arrangement is not in accordance with NFPA 13 and BTP CMEB 9.5-1, Section C.6.c.

By letter dated February 1, 1984, the applicant committed to change the layout of the sprinkler piping to bypass the HVAC duct work. On the basis of this commitment, the staff concludes that the sprinkler system in the diesel generator rooms will comply with the guidelines in Section C.6.c of BTP CMEB 9.5-1, and is, therefore, acceptable.

The diesel-fuel-oil day tanks are located in each diesel generator room. The SER states that a containment dike would be provided beneath each day tank to contain 100% of the fuel oil; however, during its visit, the staff noted that the top of the dike is beneath the tank. The staff was concerned that not all leaks would be contained by this configuration and that the applicant should modify the dike to provide a more positive collection ability (such as by completely surrounding the day tank) in accordance with Section C.7.l of BTP CMEB 9.5-1.

By letter dated February 1, 1984, the applicant indicated that the existing fuel tank and all piping are seismic Category I. The fuel-oil system is a gravity-feed-type system; therefore, no pressurized sprays will occur as a result of a leak. The floor area adjacent to the dike has floor drains. The day tank is provided with level indication that alarms in the control room if there are more than 3 gallons of leakage.

The applicant considers that the current design of the tank is adequate and, on the basis of the information provided, the staff agrees. If any leaks should occur, they would be detected promptly, and the floor drains would collect the majority of the leakage.

On the basis of its review, the staff concludes that the diesel-fuel day tank and dike assembly meets the guidelines in Section C.7.i of BTP CMEB 9.5-1, and is, therefore, acceptable.

Reactor Coolant Pumps

The reactor coolant pump (RCP) system is designed to collect and contain lubricating oil for each RCP. The collection systems are piped to two collection tanks. Each tank serves two RCPs, and each collection tank has a capacity of approximately 300 gallons. Each RCP motor contains approximately 265 gallons of oil. The collection tanks are provided with level indication and high level alarm in the control room.

Should leakage exceed the collection tank capacity before corrective actions are completed, the tank would overflow into the containment sumps. This oil would not come into contact with hot surfaces and would not pose a significant fire hazard.

The tanks are constructed to the requirements of ASME Code Section VIII and have flame arrestors on the vents. The drain piping meets American National Standards Institute Standard B31.1 (ANSI B31.1). The tanks and piping are seismically supported in accordance with the requirements of paragraph C.2 of RG 1.29.

By letter dated March 14, 1984, the applicant described how the oil collection system had been seismically analyzed and qualified to remain functional after the safe shutdown earthquake. On this basis, the staff concludes that the protection provided for the reactor coolant pumps meets the guidelines of Section C.7.a of BTP CMEB 9.5-1 and is, therefore, acceptable.

Hatchways

The auxiliary building is provided with two sets of equipment hatchways in the northern and southern ends of the auxiliary building corridors. A monorail hoist serves each set of hatchways to allow equipment to be moved from one location to another. Steel hatch covers and automatic sprinkler water curtains are provided for each hatchway at elevations 2000 ft, 2026 ft, and 2047 ft to separate the corridor fire areas.

At elevation 2000 ft in the center of the auxiliary building, two adjacent hatchways are provided above the residual heat removal and containment spray valve encapsulation tanks located at elevation 1988 ft. These two hatchways are covered with a 3-hour-rated material.

Because of the low fuel loading and configuration of equipment in these areas, the staff finds that the water curtains and steel covers provide a level of safety equivalent to the technical requirements of Section C.5.b of BTP CMEB 9.5-1.

Containment Penetrations

The reactor containment walls are penetrated by numerous mechanical and electrical penetrations, as well as by a personnel hatch and a fuel transfer tube.

The containment wall is 4-ft-thick reinforced concrete with a continuous 1/4-in.-thick steel liner. The construction is capable of withstanding a 60-psig overpressure without failure.

Because of the construction of the containment wall and the special nuclear safety-related purposes these penetrations serve, the staff considers that they provide a level of safety equivalent to the technical requirements of Section C.5.b of BTP CMEB 9.5-1.

Fuel Building Roof

No fireproofing is provided on the underside on the fuel building roof. The roof is missile proof, of 2-ft-thick reinforced concrete.

Because of the low fuel loading in this area is low, the staff finds the level of fire protection acceptable.

Trench Cover

In the fuel building Fire Area F-2, the floor is on grade, with the exception of a small pipe trench that opens into the room and connects with the radwaste tunnel. The trench opening in this room is closed by a heavy steel cover plate approximately 4 ft x 8 ft. Combustibles in this area are separated by more than 50 ft. Because of the separation distance and low combustible loading, the staff finds the level of protection acceptable.

Main Steam and Feedwater Valve Compartment

This fire area (Fire Area A-23) is separated from all adjoining areas and buildings by 3-hour-rated fire barriers. The fire area is divided into two compartments by a 2-ft-thick concrete wall. A 9-ft x 24-ft vent opening is located in the ceiling of each compartment. The barrier wall between the two compartments has a 27-ft-wide x 23-ft-high vent opening located approximately 34 ft above the floor. These vent openings are required to prevent overpressurization of the compartment in the event of a postulated break of main steam piping. Because of the vent opening, the barrier wall cannot be fire rated.

All other penetrations through the fire barriers are fitted with 3-hour-rated penetrations seals. Three-hour-rated fire dampers are installed in all HVAC ducts penetrating the fire barriers.

Because of the low combustible loading and configuration of valves in this area, the staff finds this level of protection acceptable.

Partial Suppression and Detection System

FSAR Tables 9.5B-3 and 9.5B-4 list the plant areas where automatic suppression and detection systems are not provided throughout the fire area.

The in situ and potential transient fire hazards for these areas of the plant have been assessed against the requirements for automatic sprinkler protection stipulated in Section C.5.b of BTP CMEB 9.5-1. The fire hazards in most of these areas are minimal, and partial suppression and detection systems are provided in areas where potential fire hazards exist.

Because of these conditions and the availability of manual fire fighting equipment, the staff concludes that additional automatic sprinker and detection systems are not necessary. The existing fire protection provides reasonable assurance that one shutdown-related division will remain free of fire damage. The systems are, therefore, acceptable.

On the basis of its review, the staff concludes that the fire protection provided for safe shutdown, with the approved deviations, meets the guidelines in Section C.5.b of BTP CMEB 9.5-1 and is, therefore, acceptable.

9.5.1.5 Alternate Shutdown

In Section 9.5.1.5 of SSER 3, the staff concluded that the alternative shutdown capability for the control room at the SNUPPS plants met the requirement of 10 CFR 50, Appendix R, Section III.L, for Wolf Creek and was, therefore, acceptable. BTP CMEB 9.5-1 and Section III.L are identical requirements with regard to alternative shutdown capability. For Wolf Creek, the correct reference should be SRP 9.5.1. The staff's conclusion was based on reviewing the SNUPPS FSAR and the SNUPPS control room fire hazard analysis dated November 15, 1982, and the understanding that all systems necessary to achieve and maintain hot shutdown could be isolated (which was assumed to include operability) from the control room following fire damage to any circuits in the control room by placing the isolation switches (outside the control room) to the isolated position.

A recent inspection of Wolf Creek revealed that for some systems necessary for hot shutdown (other than those on alternate shutdown panel B) to be isolated from control room fire damage and maintain operability without fuse replacement, isolation must take place before fire damage occurs. Although the present isolation switches at SNUPPS isolate the required equipment or components from the control room, it may be necessary to replace fuses as a result of control room fire damage to place the equipment/component in the desired mode of operation or position. The SNUPPS alternate shutdown procedures used at Wolf Creek and Callaway, are based on the assumption that the transfer switches will be placed in the isolated position before fire damage in the control room that could result in fuse failure in the control power circuit. For such a case the isolation switches would isolate the desired component/equipment from the control room and operability would be affected since the fuses would now be isolated from the control room circuitry. At this point any further fire damage (hot short, open, or short to ground) would not affect the component(s) in question.

However, the staff conclusions in SSER 3 were based on the understanding that replacement of fuses would not be necessary following the transfer switches being placed in the isolated position regardless of the time frame assumed for fire damage to the control room circuits. Following the inspection, the staff was made aware that the present SNUPPS design in combination with the alternate shutdown procedures did not meet staff requirements for alternative shutdown capability in the event of a control room fire.

As a result of meetings with the SNUPPS utilities on August 10, 14, 15, and 22, 1984, the staff determined that many of the concerns identified by the inspection could be taken care of by new procedures since local operation of breakers or valves could still be accomplished. In other cases it was determined that the replacement of fuses was acceptable because the components in question did not have an immediate effect on hot shutdown, and ample time was available for fuse replacement. However, there were four instances where SNUPPS identified isolation switches that required modifications and five instances where new isolation switches would have to be added. The new and modified isolation switches will have redundant fuses so that when placed in the isolation position, new fuses would be switched into the circuitry and the equipment would be isolated and immediately available.

By letter dated August 23, 1984, SNUPPS provided a detailed outline of new alternate shutdown procedures and identified where the new and modified switches were required. The proposed new procedures consist of six phases, A through F, which will be accomplished by four operators. The new procedures assume that evacuation of the control room takes place when the fire starts and operations outside the control room systematically bring all hot shutdown systems on the line and compensate for or prevent spurious operations that could affect achieving or maintaining hot shutdown.

Before an operator leaves the control room, the reactor is tripped and the main steam isolation valves (MSIVs) are closed if permitted by the fire. During Phase A, one operator establishes control at the alternate shutdown panel (ASP) using motor-driven pump B (after the diesel is running) and the atmospheric dump valves for steam generators B and D. The ASP operator also isolates the normal letdown path via an isolation switch on the ASP and closes the atmospheric dump valves for steam generators A and C. Meanwhile, other operators simulate a loss of offsite power (if not lost), strip the loads from the 4160-B bus that is isolated from the effects of a control room fire, and start the diesel generator and essential service water (ESW) flow to the diesel generator. Also during Phase A, an operator trips the reactor coolant pumps if running, and isolate the power-operated relief valves (PORVS) via a knife switch. To ensure that spurious operation of atmospheric dump valves for steam generators A and C does not affect hot shutdown, an operator (during Phase D) manually closes an isolation valve for each dump valve. New isolation switches, to ensure that essential service water (ESW) valves HV-26 and HV-38 are properly positioned, will be added. HV-26 isolates the ESW system from the service water system and HV-38 is the ESW return to the ultimate heat sink (UHS). Until these switches are installed, an operator will trip the valve breakers (motor-operated valves) and manually operate the valves if they need to be repositioned. Phase A will be completed within 5 minutes after control room evacuation and at its completion hot shutdown is being maintained at the ASP, diesel generator B is running with cooling water being supplied by ESW Train B, the reactor coolant pumps (RCPs) are secured to protect the seals, and some isolation of the primary and secondary systems has been accomplished (letdown, PORVs, and atmospheric dump valves). Although the turbine-driven auxiliary feedwater (AFW) pump is isolated, it will not be used until an operator has ensured a suction flow path is available in Phase D.

During Phase B, which will be accomplished within 10 minutes after control room evacuation, control is maintained at the ASP, the operators verify turbine trip, initiate room cooling for the ESW pump room and the diesel generator room, and the control building and auxiliary building air conditioning systems are started to ensure vital electrical areas are cooled. Also during Phase B, the isolation valves between the refueling water storage tank (RWST) and the residual heat removal (RHR) pump suctions are closed to preclude inadvertent draining of the RWST to the containment recirculation sump. New/modified isolation switches will be provided for the ESW and diesel generator ventilation inlet dampers and supply fans to ensure timely initiation of room cooling for these areas. In the interim, manual opening of the inlet dampers and fuse replacement for the supply fans may be required because of control room fire damage. A new isolation switch will also be installed for operation of HV-8812B, RWST-to-RHR-pump suction valve. Containment spray pump Train A is also tripped to prevent or stop its spurious operation. The Train B spray pump was isolated during Phase A when stripping the 4160-B bus.

During Phase C, which is completed within 20 minutes after control room evacuation, operators perform various operations to trip the valve breakers and verify the position of and manually operate, if necessary, various component cooling water (CCW) system valves to ensure proper CCW system lineup, then start CCW pumps B and D. A new isolation switch will be installed to ensure the closure of HV-70B, which is an air-operated solenoid-controlled CCW isolation valve, for the radwaste building. In the interim, a fuse can be pulled to stop dc power to the solenoid valve, causing the isolation valve to close.

During Phase D, the charging system is lined up using charging pump B, and RCP seal injection flow is initiated using the RWST as a source. If the MSIVs were not closed before evacuation of the control room, they will now be closed using a portable 125-V dc power source and wires will be cut to ensure that the MSIVs remain closed. Also during Phase D, operators ensure that the condensate storage tank (CST) is lined up to the turbine-driven AFW pump. At this time the operator at the ASP may use the turbine-driven pump instead of or in addition to the motor-driven B pump. Phase D is completed within 30 minutes after control room evacuation.

During Phase E (which will be completed 60 minutes after control room evacuation), the operators will ensure the availability/operability of systems and components required for long-term hot standby. These include containment air cooling, fuel-oil transfer system, and the isolation of minor potential blowdown paths such as the reactor head vents, steam generator blowdown system, excess letdown line and the MSIV bypass valves. During Phase E, the charging system is lined up to charge through the boron injection tank (BIT) to allow boration.

During Phase E, identified fuses are pulled to prevent spurious opening of the reactor head vent valves, excess letdown isolation valves, and the MSIV bypass valves. This is acceptable because the valves are all normally closed, fail-close valves and except for the bypass valves, require multiple hot shorts to result in a blowdown path since they have two isolation valves in series. Also, the blowdown paths are sized at 1-inch diameter, resulting in a limited rate of release. Regarding the MSIV bypass valves, additional downstream valves would have to spuriously open in order to result in steam releases. Also, if it was determined that these spurious operations might have occurred (by instrumentation on the ASP), these steps could be taken any time before reaching Phase E. Likewise, the steps to isolate the PORVs, atmospheric dump valves on steam generators A and C, or the steam generator blowdown system could be taken at any time if the instrumentation at the ASP indicated that isolation was necessary. These steps do not require pulling or replacing any fuses. Although it would take multiple hot shorts to cause spurious opening of the series RHR suction isolation valves, the breakers to one valve in each path will be tripped during normal operation to preclude a fire-induced LOCA.

The final phase, Phase F, includes operations to ensure the operability of the ESW system self-cleaning strainers. If necessary, the ESW system is lined up to the AFW system if the CST is depleted.

Many of the manual operations performed during Phases A through F are precautionary to prevent spurious operations of valves and/or pumps. It is not expected that all spurious operations will occur and in all likelihood many of the manual valve lineups described in the procedures for the cooling water systems would only be valve lineup checks. The actual valve manipulation may only be required if the valve spuriously moved to an undesired position before isolating control power from the control room, or if the valve's normal position was not that desired for the post-fire lineup.

On the basis of its review of the phased procedural approach outlined in the applicant's letter dated August 23, 1984, the staff concludes that the SNUPPS alternative shutdown capability is acceptable pending the following conditions:

(1) The applicant will revise the procedures for a fire in the control room in accordance with the SNUPPS letter of August 23, 1984, and will train operators to the revised procedures before the operating license is issued.

(2) The applicant will complete installation of the five new isolation switches and modification of the four existing isolation switches that were identified in the SNUPPS letter dated August 23, 1984, before fuel load. The staff will verify completion of the installation.

10 STEAM AND POWER CONVERSION SYSTEM

10.4 Other Features

10.4.7 Condensate and Feedwater System

The main feedwater system has been provided with an additional isolation signal that is generally not found in most Westinghouse designs. FSAR Revision 14 provided the electrical details of the additional isolation signal.

The original main feedwater system design had check valves in the main feedwater system piping just downstream of the main feedwater isolation valves (MFIVs) and upstream of the auxiliary feedwater (AFW) system connection to the main feedwater piping. This design produced excessive check valve slam as a result of high differential pressure across the check valve following AFW pump start. Therefore, the check valve in each line was moved to inside containment close to the steam generator. Although this resolved the check valve slam problem, under certain pipe break conditions, AFW flow would be fed back through the main feedwater system instead of to the steam generators because the check valves would no longer prevent backflow upstream of the AFW connection. To resolve this new concern, isolation signals derived from low-low level in any steam generator were provided to isolate the main feedwater system by closing the MFIVs and the main feedwater control valves. This isolation capability will provide a pressure boundary for the AFW system where it connects to the main feedwater system, thereby ensuring flow toward the steam generators. On the basis of its review of the main feedwater system isolation capabilities, the staff concludes that the additional isolation signal (low-low steam generator level) is necessary to ensure the capability of the AFW system to provide AFW flow to the minimum number of steam generators under all accident conditions. Therefore, the staff concludes that the design is acceptable and meets the applicable criteria in SRP 10.4.7.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure and Qualifications

Since the issuance of the SER, the applicant has made a number of organizational and personnel changes. The staff has reviewed these changes and those important to the operation of the Wolf Creek Generating Station are described below.

13.1.1 Management and Technical Resources

The licensee has restructured the corporate organization to reduce the number of subordinates who report directly to the President. The new organization is shown in Figure 13.1.

The Vice President - Nuclear now reports to the Group Vice President - Technical Services, but retains direct access to the President or Chairman of the Board on matters of an immediate nature. The Group Vice President - Technical Services has a strong nuclear background in design and operations and also oversees the non-nuclear-related engineering functions and operations of the non-nuclear plants.

The principal functions of the Vice President - Nuclear remain the same as previously reviewed by the staff with the following changes. The position of Manager Quality Assurance has been upgraded to Director Quality Assurance. The incumbent is currently located at the site. This individual possesses significant nuclear plant quality assurance (QA) experience. He is responsible for the Operating Quality Assurance Program. As noted, a corporate level Quality Assurance Committee oversees the QA program. Reporting to the Director Quality Assurance are the Manager Quality Assurance (WCGS), Manager Quality Assurance (Home Office), and the Superintendent Quality Control.

New functions under the Manager Management Systems have been added. These functions include document control and records management, as well as configuration management. This latter function provides for configuration identification, control, verification, and status accounting. Management control procedures for the Nuclear Department are developed under the direction of Manager Management Services.

To assist the Vice President - Nuclear in the day-to-day coordination of site construction activities, the applicant has hired a Project Director. This individual is located at the site and has direct responsibility for construction, engineering, and support for Wolf Creek Generating Station. This position will be eliminated following the startup. The nuclear operations organization is displayed in Figure 13.2.

There have been several personnel changes in the licensee's corporate organization. When the résumés of key persons in the organization were reviewed, these persons were determined to have the necessary experience and qualifications to fulfill their assigned duties.

The staff concludes that the applicant has an acceptable organization and adequate resources to provide offsite technical support for the operation of the Wolf Creek Generating Station.

13.1.2 Operating Organization

The operations organization under the Director Nuclear Operations remains nearly the same as previously reported with several minor exceptions. The onsite organization is shown in Figure 13.3. The position of Plant Superintendent has been elevated to Plant Manager. The Training Supervisor now reports to the Manager Nuclear Training, whereas the site training organization previously reported to the Plant Manager with only programmatic oversight by the Manager Nuclear Training.

The onsite training organization has been significantly upgraded; current plans call for a staffing level of 51. The applicant is developing a training program that is to be submitted to the Institute of Nuclear Power Operations (INPO) for full accreditation. The training organization consists of three sections. The first reports to the simulator supervisor and handles all simulator maintenance and all training using the simulator. The Supervisor Training Programs and Development heads the second section and is responsible for all training program development, job task analysis, and accreditation. The last section is under the Supervisor Training. This section is responsible for operator training (including nonlicensed operators), general employee training, and other technical and specialty training.

A recent Region IV inspection confirmed that the licensee is meeting FSAR commitments in training (except for one minor open item in the instrumentation and control (I&C) technician training that does not affect the substance of training). This inspection also confirmed that the licensee's program is proceeding on schedule and that the necessary elements of the training program will be completed before fuel loading. The results of this inspection are reported in NRC Inspection Report 50-482/84-29, dated October 9, 1984.

13.1.2.3 Operating Shift Crews

The Commission is concerned about the possible lack of hot operating experience among the operators on shift at newly licensed nuclear power plants. This has led to an evaluation of (1) the operating experience on shift proposed by the applicant and (2) the interim use of shift advisors to supplement the operating shift crews.

Operating Experience on Shift

Dialogue with the industry was begun in late 1983 to find a way of ensuring that each operating shift at a newly licensed plant had at least one senior operator with previous hot operating experience. On February 24, 1984, an Industry Working Group representing utilities with nuclear power plants under construction or ready for operation presented a proposal to the Commission on the amount of previous operating experience considered to be the minimum desirable on each shift and how that experience could be obtained. On June 14, 1984, the Commission accepted the industry proposal with certain clarifications. Information regarding the Commission action was forwarded to the industry as Generic Letter 84-16, dated June 27, 1984. The objective is that, at the time of fuel load, each operating shift will have at least one senior operator with a minimum of 6 months of hot operating experience on a similar type plant, including startup/shutdown experience and at least 6 weeks above 20% power. However, for plants in the late stages of licensing with insufficient time to meet the objective, the temporary use of experienced shift advisors is acceptable. The minimum qualifications for shift advisors are 4 years of power plant experience (including 2 years of nuclear power plant experience) and 1 year of hot operating experience as a senior reactor operator (or reactor operator, if found suitably qualified) on a large commercial nuclear power plant of the same type. All shift advisors are to be trained on the systems, procedures, and Technical Specifications of the plant for which they are to provide advice, and they are to be certified to the NRC as being qualified to act as shift advisors. Wolf Creek Unit 1 is in the group of plants electing to use shift advisors to supplement the experience of the operating shifts.

The applicant has selected six individuals to act as Wolf Creek shift advisors (the applicant uses the term "shift consultants"). These individuals have had commercial nuclear power plant experience ranging from ~2 years to ~10 years. Four have had U.S. Navy nuclear experience of 6 years or more, and two have had experience as shift advisors at a large pressurized-water reactor (PWR). Each of the proposed shift advisors has had at least 2 years of experience as a licensed senior operator and/or licensed operator at a large operating PWR.

The staff has reviewed the qualifications of the proposed shift advisors and the plant-specific training provided to them at Wolf Creek, and concludes that, subject to certain clarifications, they meet the guidelines for shift advisors as adopted by the Commission. These clarifications are discussed below in the section on the shift advisor program.

In addition, because Wolf Creek does not now have senior operators on each shift who meet the minimum guidelines for hot operating experience, the staff will condition the operating license to require shift advisors until such time as the requisite experience has been obtained.

Shift Advisor Program

By letters dated March 12, June 21, October 8, and October 11, 1984, the applicant has submitted information regarding its shift advisor program. The staff has reviewed this information for conformance to Generic Letter 84-16. In performing the review, the staff also used additional information regarding qualifications and training of shift advisors that was developed during its review of shift advisor programs at several other utilities.

The review of the Wolf Creek shift advisor program comprised four main areas: shift advisor qualifications, the training program for shift advisors (including written and oral examinations), the procedure used to define shift advisor duties and responsibilities, and any other guidance or requirements pertaining to the use of shift advisors.

(1) Shift Advisor Qualifications

One of the six shift advisors amply meets all the experience requirements of the Industry Working Group proposal of February 24, 1984, as clarified by the Commission on June 14, 1984.

Although three of the remaining five shift advisors are very close to meeting the requirements, each has slightly less than 1 year on shift as a senior reactor operator at an operating plant of the same type as Wolf Creek. However, if their reactor operator experience is added to their senior reactor operator experience, the periods of licensed operating experience range from 2 to 4 years. One of the individuals has approximately 2 years' experience at Arkansas Nuclear One, Unit 1, including supervisory experience obtained during daily operations and three refueling outages. Another has approximately 6 years' experience at the Trojan Nuclear Plant, including a period as Assistant Shift Supervisor. The third has approximately 6 years' experience at the Donald C. Cook Nuclear Power Plant, including experience at directing and performing operational evolutions during startup, physics testing, and power operation up to full power. Thus, the staff concludes that these three shift advisors have sufficient experience, including supervisory experience, to participate in the shift advisor program at Wolf Creek.

The two remaining shift advisors do not have time on shift as senior operators; however, each has at least 6 years of U.S. Navy nuclear experience and at least 9 years of commercial nuclear experience. One has 2 years and the other has 4 years of experience as a licensed operator at a large operating PWR. In addition, each has served as a shift advisor at the Sequoyah Nuclear Plant. By virtue of their extensive nuclear experience, and the fact that they also have supervisory experience as shift advisors, the staff concludes that these two shift advisors are qualified to participate in the shift advisor program at Wolf Creek.

(2) Shift Advisor Training Program

The Wolf Creek shift advisor training course is 6 weeks long, including 3 weeks of classroom training and 3 weeks of simulator training on the Wolf Creek site-specific simulator. Quizzes are given at the end of each classroom week. A 4-hour written examination and a simulator oral examination are conducted at the end of the course.

The classroom portion of the course consists of the following major areas:

- primary systems
- secondary systems
- reactor support systems
- plant support systems
- electrical systems
- instrumentation and control systems
- emergency core cooling systems
- theory review
- administrative review

The simulator portion of the course consists of fifteen 4-hour sessions. Each session stresses communications, Technical Specifications, and compliance with procedures. Simulator training covers control board familiarization, heatup, reactor startup, shutdown to hot standby, cooldown to cold shutdown, and power ramps associated with various major malfunctions (e.g., reactor trip, loss of coolant, steamline break, steam generator tube rupture, and loss of all ac power).

The Wolf Creek shift advisor training course meets the training guidelines of the Industry Working Group as clarified by the Commission on June 14, 1984. Before reaching initial criticality, the applicant should provide the staff with the names of advisors who have been examined and determined to be competent to provide advice to operating shifts. In addition, the remainder of the shift crews should be trained in the role of the advisors.

(3) Shift Advisor Procedure

The qualifications and responsibilities of the shift advisor are contained in Wolf Creek Generating Station Administrative Procedure ADM 02-012. This procedure establishes qualification criteria, normal responsibilities and authority, limitations, and record-keeping requirements associated with the shift advisor program.

The shift advisor functions at the Shift Supervisor level. He is assigned to assist the Shift Supervisor, but reports to the Superintendent of Operations and has direct access to plant management.

The shift advisor responsibilities given in Section 3.0 of ADM 02-012 include

- evaluating shift operating activities and providing appropriate recommendations concerning safe operation
- pursuing the resolution of disagreements that may affect safe operation
- assisting in a determination of the circumstances, an analysis of the cause, and a determination that operations can proceed following a trip or unplanned/unexplained power reduction
- assisting in on-shift training of operating personnel
- participating in recurrent training
- ensuring that the proper level of plant management has been informed of events that affect, or could affect, plant operations
- assisting in the review and modification of operations procedures
- assisting in the preparation and review of shift documentation and operating data

Procedure ADM 02-012 precludes the shift advisors from directly manipulating equipment or supervising operators in assignments that require a license. However, the version of the procedure reviewed by the staff (Rev. 0) also gives the shift advisors the authority to order the immediate trip or shutdown of the reactor and to order the immediate cessation of any activity in the plant that decreases station or personnel safety. The staff agrees with the limitations that restrict the shift advisor from manipulating equipment or directing licensed activities, but prefers that ADM 02-012 be revised so that the shift advisor can only make shutdown or stop-work recommendations. If the Shift Supervisor disagrees with a recommendation, the procedure allows the shift advisor to pursue the matter with higher plant management. The procedure should be revised, and the shift advisors and operating crews trained in the revised procedure before reaching initial criticality.

(4) Additional Shift Advisor Requirements

The applicant plans to evaluate shift advisor performance periodically to ensure effective participation in plant activities, effective communications, satisfactory personnel interactions, and a level of knowledge consistent with job responsibilities. The evaluations will be based on reports from plant management (Shift Supervisor through the Superintendent of Operations), peer reports from other members of the plant staff, and feedback from the shift advisors themselves regarding the effectiveness of the shift advisor position. The applicant plans to conduct the evaluations at any time, but at least annually. The staff concurs with the method of evaluation, but not necessarily with the timing. Since the shift advisor position is an interim position only, evaluations should be performed frequently (at least once per month).

The shift advisor is an important part of the shift crew, and shift interactions should be monitored closely to verify effectiveness.

The shift advisors will be given physical examinations that meet regulatory requirements pertaining to physical examinations for licensed operators. The examinations should be given before reaching initial criticality.

The applicant has not indicated whether the shift advisors will participate in the licensed operator requalification program. The staff (and industry reviewers at other plants) recommends that the Wolf Creek shift advisors be enrolled in the operator requalification program and, when possible, attend training sessions with their assigned crews.

By letter, dated December 7, 1984, the applicant committed to the following staff recommendations:

(1) Procedure ADM 02-012 will be revised to specifically preclude shift advisors from giving shutdown or stop-work orders. The revision will be complete before fuel loading.

(2) Monthly evaluations of shift advisors commenced in December 1984.

(3) Shift advisor physical examinations will be completed before reaching initial criticality.

(4) Shift advisors are participating in a shift advisor requalification program. Elements included in the program involve plant procedures, Technical Specifications, plant systems, and simulator experience.

On the basis of its evaluation and these commitments, the staff finds the applicant's shift advisor program acceptable.

13.3 Emergency Planning

13.3.1 Introduction

The applicant filed Revision 10 to the Wolf Creek Generating Station Radiological Emergency Response Plan in April 1983. The plan was reviewed against the requirements in 10 CFR 50.33 and 50.47, the requirements of Appendix E to 10 CFR 50, and the guidance criteria of NUREG-0654/FEMA-REP-1, Revision 1, entitled "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," November 1980. The criteria in NUREG-0654 have been endorsed by RG 1.101, Revision 2, dated October 1981. On the basis of a review of Revision 10 to the plan, the staff identified unresolved items requiring revision to the emergency plan or additional information in Supplement 4 to the Wolf Creek SER (SSER 4).

Since the publication of SSER 4, the staff has received revised plans through Revision 15, dated January 1985, and additional information from the applicant. The plan revisions and supporting information have been reviewed to evaluate the adequacy of emergency preparedness for the Wolf Creek Generating Station. The findings of the staff regarding the unresolved items identified in SSER 4 are presented in Section 13.3.2 of this supplement. In addition, the initial findings on offsite emergency preparedness by the Federal Emergency Management Agency (FEMA) are presented in Section 13.3.3 and the status of the Atomic Safety and Licensing Board (ASLB) conditions and confirmatory items is summarized in Section 13.3.4. The staff's conclusions on emergency preparedness for Wolf Creek are given in Section 13.3.5.

The onsite appraisal of the applicant's capability to implement the Wolf Creek emergency plan was conducted by NRC during the period September 17-28, 1984. The appraisal findings are contained in Inspection Report 50-482/84-25 dated December 21, 1984. A full participation exercise of the onsite and offsite emergency plans for Wolf Creek was held on November 7, 1984. The exercise was evaluated by both the NRC (on site) and the Federal Emergency Management Agency (off site).

13.3.2 Evaluation of Emergency Plan

This section of the supplement discusses each unresolved emergency planning issue and the specific information provided by the applicant to resolve the identified inadequacy.

13.3.2.1 Assignment of Responsibility (Organizational Control)

In SSER 4, the staff noted that an item requiring resolution involved specific written agreements for some offsite support organizations that still needed to be developed. In submittals dated February 16 and April 30, 1984, the applicant provided the additional agreement letters with offsite support organizations that were subsequently incorporated into Supplement CC of Revision 15 of the emergency plan. The staff finds that this item has been satisfactorily resolved.

13.3.2.2 Onsite Emergency Organization

In SSER 4, the applicant's approach to meeting the shift staffing guidance of Table 2 in Supplement 1 to NUREG-0737 was discussed. As a result of observations made during the onsite implementation appraisal and the emergency plan exercise, additional information on this subject was requested from the applicant and was subsequently provided in a submittal dated December 10, 1984. A review of this information shows that for each of the major functional areas (in Table 2 of the emergency plan) with a 30-minute staffing addition objective (a total of 11 emergency response positions), the applicant has available 2 or more trained personnel (34 total) living within 30 minutes of the Wolf Creek site. In the event of an emergency, the applicant states that an attempt will be made to notify all of those trained personnel living within 30 minutes of the site in an effort to meet the 30-minute staff augmentation goal of Supplement 1 to NUREG-0737, with a commitment to staff these functional areas in a maximum of 60 minutes as shown in Table 1.1-1 of the emergency plan.

The applicant has provided similar information for the position of Duty Emergency Manager (DEM), who fulfills the Table 2 position of Emergency Operations Facility (EOF) Director, and four other key positions in the EOF. This information shows that there are two trained DEMs living within 60 minutes of the site, and one other trained DEM living within a 90-minute distance. For each of the other key EOF positions, there are one or two trained personnel living within 60 minutes of the site and, for two of the positions, one or two others living within a 90-minute distance. In the event of an emergency requiring activation of the EOF, the applicant states that (1) an attempt will be made to notify all of those personnel trained for the DEM and for other key EOF positions, and (2) the first person arriving will assume the emergency role until he is relieved by a more senior person. Enhanced communications, including radio-telephone beepers, will be used to contact the on-call staff and others. The applicant also states that more personnel will be trained to fill these positions.

Revision 15 of the emergency plan reflects the information on shift staffing augmentation discussed above. On the basis of (1) information included in the December 10, 1984, submittal and in the emergency plan, and (2) the applicant's commitment to train more key response personnel, the staff finds that the applicant has made reasonable progress toward achieving the shift staffing augmentation goals of Table 2 in Supplement 1 to NUREG-0737, and the applicant's staffing for emergencies is adequate.

13.3.2.4 Emergency Classification System

In SSER 4 (on the basis of a review of information provided in an October 10, 1983, submittal), the staff found acceptable the applicant's responses to the staff's request for additional information concerning (1) the applicant's basis for recognizing damage to the fuel barrier in the Wolf Creek emergency classification system, (2) classification of the "challenge to a barrier" concept, and (3) the incorporation of fire and security events into the emergency classification system. The staff has verified that these concepts have been incorporated into the revised emergency plan and procedures.

13.3.2.5 Notification Methods and Procedures

SSER 4 referred to a commitment made by the applicant in a letter to the NRC dated October 21, 1983, to install and make operational the Wolf Creek alert and notification system before fuel loading. On the basis of information obtained by the NRC during the onsite appraisal and on observations made by the Federal Emergency Management Agency (FEMA) during the November 11, 1984, exercise, the staff has confirmed that the siren portion of the system has been installed and made operational and that the tone-alert radios have been distributed throughout the plume exposure pathway emergency planning zone (EPZ). FEMA has informed the staff that a remedial test was conducted in December 1984 that demonstrated correction of a siren deficiency identified by FEMA in the November 1984 exercise.

13.3.2.6 Public Information

In SSER 4, the staff identified the need for the public information brochure to be submitted for review and for the brochures to be distributed to the public before fuel loading. Public information brochures were obtained by the NRC during the onsite appraisal and exercise and a copy was formally submitted in a letter dated December 3, 1984. The brochure has been reviewed by the staff and found to be in conformance with the guidance of NUREG-0654. The staff has verified in the followup inspection conducted during the week of January 7, 1985 (Inspection Report 50-482/85-02) that the brochures have been distributed to the public in the plume exposure pathway EPZ.

SSER 4 made reference to facilities for the news media that had been established in New Strawn, Kansas. As shown in Section 4.1.2.3 of the emergency plan, Revision 15, the New Strawn facility has been deleted. The applicant's designated Media Release Center (MRC) is located in the Nickell Memorial Armory in Topeka, Kansas. This facility was examined during the implementation appraisal in September 1984 and found to be adequate. The MRC in Topeka was also used during the full participation exercise conducted on November 7, 1984. On the basis of these evaluations, the staff concludes that the facilities provided for the news media are adequate.

13.3.2.7 Emergency Facilities and Equipment

In SSER 4, the staff identified that the applicant had committed in a letter dated October 21, 1983, to provide permanent, fully completed, emergency response facilities and systems or adequate interim facilities and systems before fuel loading. On the basis of review of information in the emergency plan and procedures, the findings of the onsite appraisal, and observations made during the emergency plan exercise, the staff concludes that, on an interim basis, the applicant's emergency response facilities (i.e., the TSC, OSC, and EOF) are adequate to support an emergency response effort in the event of an emergency at Wolf Creek. As stated in SSER 4, the staff will conduct a postimplementation appraisal of the applicant's emergency response facilities in accordance with Supplement 1 to NUREG-0737 on a schedule to be developed by both the applicant and NRC.

The staff also identified in SSER 4 that the applicant had committed in a letter dated November 21, 1983, to have in place an agreement with a National Weather Service (NWS) first-order station (i.e., 24-hour-per-day coverage) to provide regional weather information. Revision 15 of the WCGS emergency plan contains in Supplement CC a memorandum of understanding with the National Weather Service Forecast Office at Topeka, Kansas, in which the NWS agrees to provide the applicant with current weather and forecast information for Coffey County, as requested. On the basis of this information, the staff finds that this issue has been resolved.

13.3.2.8 Protective Response

In SSER 4, the applicant's protective action decisionmaking process and interface with offsite authorities was described and reference was made to commitments contained in a November 21, 1983, letter to the NRC. In this submittal, the applicant stated that predetermined protective action recommendations keyed to each of the four emergency classes will be developed and these recommendations will be incorporated into the emergency plan and implementing procedures. Emergency plan implementing procedure EPP 01-10.1, Revision 1, "Protective Action Recommendations," provides the applicant's procedures for formulating and recommending protective action measures to appropriate State and County authorities. In accordance with the guidance of NUREG-0654 and IE Information Notice 83-28, EPP 01-10.1 clearly establishes that the applicant's protective action recommendations are based on an analysis of plant conditions as well as dose projections. The description of the development of protective action recommendations included in Chapter 3 of earlier revisions of the emergency plan, and in a response to the NRC dated December 10, 1984, was not entirely consistent with EPP 01-10.1. The emphasis in the emergency plan was on developing protective action recommendations based on dose projections and comparison with Protective Action Guides (PAGs) rather than indicating that protective action recommendations will also be based on plant/core conditions. The applicant has revised the emergency plan (Revision 15, dated January 1985) to reflect that the onsite protective action decisionmaking process is based on plant/core conditions as well as dose projections.

The staff concludes that on the basis of information in Revision 15 of the emergency plan and EPP 01-10.1, and on the applicant's demonstrated performance in the November 1984 exercise, the applicant is in conformance with the emergency planning standard for protective response. The staff recommends that the applicant continue to coordinate emergency planning efforts with offsite authorities to ensure that offsite plans fully reflect the onsite decisionmaking process for developing protective action recommendations.

13.3.3 FEMA Findings on Offsite Emergency Plans and Preparedness

On February 2, 1984, FEMA provided an interim finding report to the NRC based on a review of the State and County emergency plans. The report stated that the offsite plans for a radiological emergency at Wolf Creek were generally adequate in complying with the planning standards of NUREG-0654/FEMA-REP-1, Revision 1, and were capable of being implemented. The interim finding report also included comments on identified deficiencies in the offsite plans and FEMA noted that the State of Kansas had been requested to develop corrective actions in response to the FEMA comments. On September 28, 1984, FEMA provided an updated interim finding report in which the staff was informed that FEMA finds that all deficiencies in offsite planning either have been corrected or were scheduled for correction by January 1985. On November 7, 1984, a full participation exercise involving State and County response was conducted at the Wolf Creek site. FEMA's findings and determinations on the completion of the necessary corrective actions in the offsite plans and the results of FEMA's evaluation of the emergency plan exercise will be provided in a future supplement to the SER before authorization for power ascension above 5% of rated power.

13.3.4 Atomic Safety and Licensing Board Conditions

In an Initial Decision issued on July 2, 1984, the Atomic Safety and Licensing Board (ASLB) specified two conditions to be met before the operating license was issued. Subsequently, an Order was issued on July 26, 1984, that modified one of the conditions and clarified that the conditions were to be met before full-power operation (i.e., greater than 5% of rated power). The two conditions, as modified, are as follows:

(1) Letters of agreement shall be signed by Coffey County with hospitals and nursing homes in surrounding counties providing for the acceptance of patients from the Coffey County Hospital and the Golden Age Lodge Nursing Home in the event of an emergency evacuation occasioned by an accident at the Wolf Creek plant. These executed letters of agreement shall be submitted to the NRC staff and shall be included in the Coffey County Plan.

(2) Letters of agreement shall be signed by Coffey County with ambulance services and with funeral directors in surrounding counties to transport non-ambulatory patients from the Coffee County Hospital and from the Golden Age Lodge Nursing Home in the event of an emergency evacuation occasioned by an accident at the Wolf Creek plant. These executed letters of agreement shall be submitted to the NRC staff and shall be included in the Coffey County Plan.

In addition to the two license conditions, the Board, in the Initial Decision, requested that the staff confirm the following conditions:

- The tone alert radios have been installed, and the standard "fire" notification procedure has been set forth in the County Plan Implementing Procedures.
- A second telephone line has been installed in the County Engineer's Office.
- Radio equipment for the Sheriff has been installed.
- Additional sirens have been installed in the John Redmond Reservoir area.
- The County Plan Implementing Procedures have been amended to reflect a breakdown, by class and by number, of the County workers who will be furnished with dosimeters.
- The Implementing Procedures have been amended to specify where the dosimeters will be prepositioned or where the County workers in each class will be able to secure their dosimeters, and the number and types of such dosimeters.
- The County Plan and Implementing Procedures appropriately reflect the revisions describing the Joint Training Program.
- Radio equipment has been installed for the Coffey County fire departments and in vehicles of the Road Department.
- The U.S. Army Corps of Engineers will provide its own dosimeters or the applicant will provide them.

Since all of the license conditions and confirmatory items are related to offsite preparedness, the staff requested the assistance of FEMA in verifying that the license conditions have been met and the confirmatory items have been satisfactorily completed.

In a response dated November 21, 1984, FEMA reported that License Condition 1 had been met but that License Condition 2 was still open because of an incomplete tri-county mutual aid agreement to provide ambulance services for Coffee

County hospital and nursing home patients. The FEMA report included the letters of agreement with the hospitals and nursing homes referred to in License Condition 1. FEMA also reported that all of the confirmatory items, with the exception of the second (installation of a second telephone line), have been completed. Regarding that item, FEMA stated that it was still open because the second telephone line to the Coffee County Engineer's Office had not yet been installed. FEMA is expected to provide additional information on that remaining license condition and confirmatory item in a report to the NRC in January 1985. This information will be included in a future supplement to the SER before authorization for power operation above 5% of rated power.

13.3.5 Conclusion

On the basis of its review of the Wolf Creek Generating Station Radiological Emergency Response Plan, Revision 15, dated January 1985, and additional information submitted by the applicant, the staff concludes that the state of onsite emergency planning is in accordance with the requirements of 10 CFR 50.47 and Appendix E to 10 CFR 50, and all emergency planning requirements necessary for issuance of a license authorizing fuel loading and low-power (<5%) operation are satisfied. After a review of the findings and determinations of FEMA on the adequacy of State and local emergency preparedness, the staff will provide its conclusion on the overall state of onsite and offsite emergency preparedness for Wolf Creek in a future supplement to the SER before authorizing operation above 5% of rated power.

13.5 Plant Procedures

13.5.2 Operating and Maintenance Procedures

13.5.2.1 General

The staff guidance for upgrading Emergency Operating Procedures (EOPs) was provided in NUREG-0881, "SER Related to the Operation of Wolf Creek Generating Station, Unit 1," April 1982. The schedule and review requirements for TMI Action Plan (TAP) Item I.C.1 have been modified by Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability (Generic Letter No. 82-33)," dated December 17, 1982.

Supplement 1 to NUREG-0737 requires that technical guidelines be submitted to the NRC for review. For Wolf Creek Unit 1, this requirement is satisfied by the applicant's commitment in the Wolf Creek Generating Station Procedures Generation Package (PGP), submitted in letters dated November 28, 1983, and August 10, 1984, to implement upgraded EOPs based on the Westinghouse Owners Group Emergency Response Guidelines (ERGs), Revision 1, without any deviations. The NRC staff approved ERGs, Revision 1, for implementation in a letter dated December 27, 1984.

Supplement 1 to NUREG-0737 also requires that each licensee and applicant submit to the NRC a PGP at least three months before the date the utility plans to begin formal operator training on the upgraded EOPs. The PGP is to include

(1) plant-specific technical guidelines
(2) plant-specific writers's guide
(3) a description of the program for validation of EOPs
(4) a description of the training program for the upgraded EOPs

Criteria for the review of PGPs are not currently in the Standard Review Plan. Therefore, this review was not based on NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures," the reference document for the EOP upgrade portion of Supplement 1 to NUREG-0737 (Generic Letter 82-33).

13.5.2.3 Reanalysis of Transients and Accidents; Development of Emergency Operating Procedures

The staff has reviewed the Wolf Creek Unit 1 PGP, submitted by letter dated November 28, 1983, and additional information requested by the staff that was submitted in a revised PGP by letter dated April 4, 1984. The staff conducted its review to determine the adequacy of the applicant's program for preparing EOPs. The review consisted of an evaluation of

(1) the applicant's plant-specific technical guidelines, including

  (a) the planned method for developing plant-specific technical guidelines from approved generic technical guidelines

  (b) deviations from the approved generic technical guidelines and their technical justification

  (c) a description of the analysis of operator functions and tasks to identify operator information and control needs which either serve as a basis for deriving instrumentation and controls, or, for existing instrumentation and controls, against which the instrumentation and controls can be evaluated

(2) the applicant's plant-specific writer's guide, which details the specific methods to be used in preparing and modifying EOPs (These methods are to ensure that the EOPs are usable, accurate, complete, readable, convenient to use, and acceptable to the control room personnel.)

(3) the applicant's program for validating and verifying its EOPs

(4) the applicant's program for training operators on the upgraded EOPs

The applicant's PGP provides reasonable assurance, with the exception noted in the following paragraph, that the resulting EOPs will be based on approved technical guidelines that address a wide range of multiple and consequential failures, that the EOPs will be acceptable and usable by operators, that the accuracy and usability of the EOPs will be validated, and that the operators will be trained on the EOPs before the EOPs are implemented.

The exception is that the applicant needs to describe the process that was or will be used to derive the instrumentation and control characteristics from the information contained in Revision 1 of the generic guidelines and related background information. The function and task analysis, as required by Supplement 1 to NUREG-0737, applies to both the upgrade of EOPs and the Detailed Control Room Design Review (DCRDR) (TMI Action Plan Item I.D.1). Instrument characteristics include parameter, parameter type, dynamic range, setpoints, resolution/accuracy, speed of response, units, and the need for trending. Control characteristics include type (discrete or continuous), discrete functions (e.g., On, Off, Auto), rate, gain, response requirements, transfer function, criticality, and frequency of use. For the characteristics that cannot be derived from the ERGs and background documentation, the applicant needs to describe the process that was or will be used to generate information and control needs (from transient and accident analysis and associated information) that will then be used to derive instrumentation and control characteristics. Where the applicant deviates from the generic instrumentation and control characteristics, or develops its own needed characteristics, it needs to identify the deviations as part of the plant-specific deviations from the generic guidelines in the plant-specific technical guidelines portion of the PGP. This analysis, and associated documentation, may take place after the issuance of a full-power license. If the function and task analysis is not completed before fuel loading, the operating license will be conditioned to require the applicant to comply with the requirements of Supplement 1 to NUREG-0737 as applied to the DCRDR and the upgrade of EOPs on a schedule approved by the staff.

The staff has evaluated the impact of not completing the function and task analysis before plant operation. The Westinghouse Owners Group ERGs identify the existing control room instrumentation and controls to be used during implementation of the EOPs for an assumed reference plant design. The instrumentation and controls do not appear to be derived from operator information and control needs. Revision 1 to the Westinghouse Owners Group ERGs does, however, identify the basic information and control needs that the existing instrumentation and controls can meet. With this information, each licensee and applicant can evaluate the adequacy of installed instrumentation and controls to meet the operators' information and control needs, and determine needed instrument and control chaaracteristics to the level of information provided in the generic guidelines. Because the reference plant for the Westinghouse Owners Group ERGs was a SNUPPS design, and because Wolf Creek is a SNUPPS-designed plant, the staff has reasonable assurance that the Wolf Creek EOP upgrade effort will result in adequate EOPs. However, the staff finds that the applicant needs to complete the task analysis efforts, as required by Supplement 1 to NUREG-0737, to further reduce the risk to public health and safety.

13.6 Industrial Security

The applicant originally filed with the NRC the following security program plans for the Wolf Creek Generating Station, which have since been revised and amended:

- "Wolf Creek Generating Station Physical Security Plan and Safeguards Contingency Plan," Revision 0, transmitted by letter dated February 8, 1980.
- "Security Training and Qualification Plan," Revision 0, transmitted by letter dated July 30, 1981.

This section summarizes how the applicant has provided for meeting the requirements of 10 CFR 73. It includes a basic analysis that is available for public review and a protected appendix. On the basis of its review of the documents cited above and visits to the site, the staff has concluded that the protection provided by the applicant against radiological sabotage at the Wolf Creek Generating Station meets the requirements of 10 CFR 73. Accordingly, the staff finds that the protection will ensure that the health and safety of the public will not be endangered.

13.6.1 Physical Security Organization

To satisfy the requirements of 10 CFR 73.55(b), the applicant has provided a physical security organization that includes a Security Shift Lieutenant on the site at all times and who has the authority to direct the physical protection activities. To implement the commitments made in the physical security, training and qualification, and the safeguards contingency plans, written security procedures specifying the duties of the security organization members have been developed and are available for inspection. The training program and critical security tasks and duties for the security organization personnel are defined in the "Wolf Creek Generating Station Security Training and Qualification Plan," which meets the requirements of 10 CFR 73, Appendix B, for the training, equipping, and requalifying of security organization members. The physical security plan and the training program provide commitments that preclude the assignment of any individual to a security-related duty or task before that individual is trained, equipped, and qualified to perform the assigned duty in accordance with the approved guard training and qualification plan.

13.6.2 Physical Barriers

In meeting the requirements of 10 CFR 73.55(c), the applicant has provided a protected area barrier that meets the definition in 10 CFR 73.2(f)(1). A 20-ft-wide isolation zone, to permit observation of activities at the perimeter, is provided (except for the locations listed in the protected appendix) along both sides of barrier.

The staff has reviewed those locations and determined that the security measures in place are satisfactory and continue to meet the requirements of 10 CFR 73.55(c).

Illumination of 0.2 foot-candle is maintained for the isolation zones, protected area barriers, and external portions of the protected area.

13.6.3 Identification of Vital Areas

The protected appendix contains a discussion of the applicant's vital area program and identifies those areas and items of equipment determined to be vital for protection purposes. Vital equipment is located within vital areas that are located within the protected area; passage through at least two barriers, as defined in 10 CFR 73.2(f)(1) and (2), is required to gain access to the vital equipment. Vital area barriers are separated from the protected area barrier.

The control room and central alarm station are provided with bullet-resistant walls, doors, ceilings, floors, and windows.

On the basis of these findings and the analysis in paragraph C of the protected appendix, the staff has concluded that the applicant's program for identification and protection of vital equipment satisfies the regulatory intent. However, this program is subject to onsite validation by the staff and to subsequent changes, if any are found necessary.

13.6.4 Access Requirements

In accordance with 10 CFR 73.55(d), all points of personnel and vehicle access to the protected area are controlled. The individual responsible for controlling the final point of access into the protected area is stationed in a bullet-resistant structure. As part of the access control program, vehicles (except under emergency conditions), personnel, packages, and materials entering the protected area are searched for explosives, firearms, and incendiary devices by electronic search equipment and/or physical search.

Except for applicant-designated vehicles, vehicles operating in the protected area are controlled by escorts. Applicant-designated vehicles are limited to onsite station functions, and they remain in the protected area except for operational maintenance, repair, security, and emergency purposes. Positive control over the vehicles is maintained by personnel authorized to use the vehicles or by the escort personnel. A photo-badge/key-card system, utilizing encoded information, identifies individuals who are authorized unescorted access to protected and vital areas, and is used to control access to these areas. Individuals not authorized unescorted access are issued badges without photographs that indicate an escort is required. Access authorizations are limited to those individuals who need access to perform their duties.

Unoccupied vital areas are locked and alarmed. During periods of refueling or major maintenance, access to the reactor containment(s) is positively controlled by a member of the security organization to ensure that only authorized individuals and materials are permitted to enter. In addition, all doors and personnel/equipment hatches into the reactor containment(s) are locked and alarmed. Keys, locks, combinations, and related equipment are changed on an annual basis. In addition, when an individual's access authorization has been terminated because of a lack of reliability or trustworthiness or for poor work performance, the keys, locks, combinations, and related equipment to which that person had access are changed.

13.6.5 Detection Aids

In satisfying the requirements of 10 CFR 73.55(e), the applicant has installed intrusion detection systems at the protected area barrier, at entrances to vital areas, and at all emergency exits. Alarms from the intrusion detection system annunciate within the continuously manned central alarm station and at a secondary alarm station located within the protected area. The central alarm station is located so that the interior of the station is not visible from outside the perimeter of the protected area. In addition, the central alarm station is constructed so that walls, floors, ceilings, doors, and windows are bullet resistant. The alarm stations are located and designed so a single act cannot interdict the capability of calling for assistance or responding to alarms. The central alarm station contains no other functions or duties that would interfere with its alarm response function. The intrusion detection system transmission lines and associated alarm annunciation hardware are self checking and tamper indicating. When activated, alarm annunciators indicate the type of alarm and its location. An automatic indication of when the alarm system is on standby power is provided in the central alarm station.

13.6.6 Communications

As required in 10 CFR 73.55(f), the applicant has provided for the capability of continuous communications between the central and secondary alarm station operators, guards, watchmen, and armed response personnel through the use of a conventional telephone system and a security radio system. In addition, direct communication with the local law enforcement authorities is maintained through the use of a conventional telephone system and two-way FM radio links. All nonportable communication links, except the conventional telephone system, are provided with an uninterruptible emergency power source.

13.6.7 Test and Maintenance Requirements

In meeting the requirements of 10 CFR 73.55(g), the applicant has established a program for testing and maintaining all intrusion alarms, emergency alarms, communication equipment, physical barriers, and other security-related devices and equipment. Equipment or devices not meeting the design performance criteria or that failed to operate otherwise will be compensated for by appropriate measures as defined in the "Wolf Creek Generating Station Physical Security Plan" and in site procedures. The compensatory measures defined in these plans will ensure that the effectiveness of the security system is not reduced by failures or other contingencies affecting the operation of the security-related equipment or structures. Intrusion detection systems are tested for proper performance at the beginning and end of any period in which they are used for security. Such testing will be conducted at least once every 7 days.

Communication systems for onsite communications are tested at the beginning of each security shift. Offsite communications are tested at least once each day. Audits of the security program are conducted once every 12 months by personnel independent of site security management and supervision. The audits, focusing on the effectiveness of the physical protection provided by the onsite security organization implementing the approved security program plans, include but are not limited to (1) a review of the security procedures and practices, (2) system testing and maintenance programs, and (3) local law enforcement assistance agreements. A report is prepared documenting audit findings and recommendations and is submitted to the plant management.

13.6.8 Response Requirements

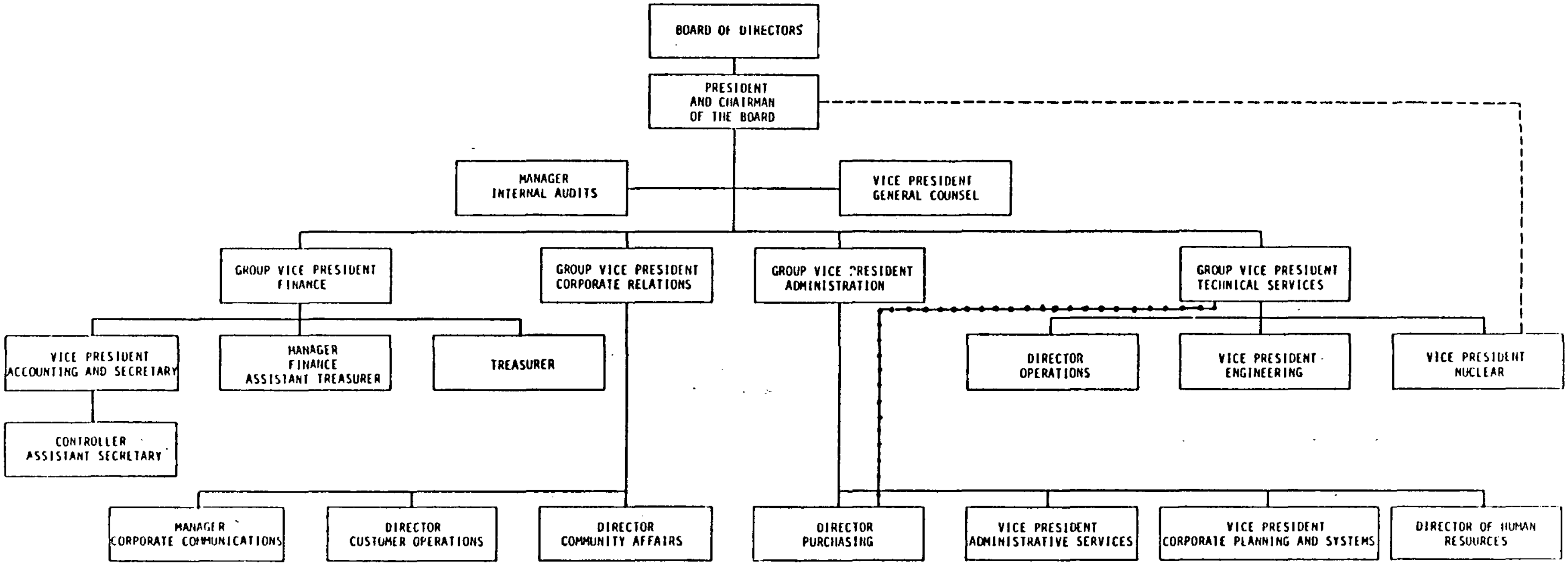
In meeting the requirements of 10 CFR 73.55(h), the applicant has provided for armed responders who are immediately available for response duties on all shifts, consistent with the requirements of the regulations. In addition, liaison with local law enforcement authorities to provide additional support to respond to security events has been established and documented.

The applicant's safeguards contingency plan for dealing with thefts, threats, and radiological sabotage events satisfies the requirements of 10 CFR 73, Appendix C. The plan identifies appropriate security events that could initiate a radiological sabotage event and identifies the applicant's preplanning, response resources, safeguards contingency participants, and coordination activities for each identified event. Through this plan, when abnormal presence or activity is detected within the protected or vital areas, response activities using the available resources would be initiated. The response activities and objectives include neutralizing the existing threat by requiring the response force members to interpose themselves between the adversary and the objective, instructing the response force members to use force commensurate with that used by the adversary, and authorizing the response force members to request sufficient assistance from the local law enforcement authorities to maintain control over the situation.

To assist in the assessment/response activities, the security organization has a closed circuit television system that has the capability to observe the entire protected area perimeter, isolation zones, and a majority of the protected area.

13.6.9 Employee Screening Program

In meeting the requirements of 10 CFR 73.55(a) to protect against the design-basis threat as stated in 10 CFR 73.1(a)(1)(ii), the applicant has provided an employee screening program. Personnel who successfully complete the employee screening program or its equivalent may be granted unescorted access to protected and vital areas at the Wolf Creek site. All other personnel requiring access to the site are escorted by persons authorized and trained for escort duties, who have successfully completed the employee screening program. The employee screening program is based on accepted industry standards and includes a background investigation, a psychological evaluation, and a continuing observation program. In addition, the applicant may recognize the screening program of other nuclear utilities or contractors based on a comparability review conducted by the applicant. The plan also provides for a "grandfather clause" exclusion that allows recognition of a certain period of trustworthy service with the applicant or contractor as being equivalent to the overall employee screening program. The staff has reviewed the applicant's screening program against the accepted industry standards (ANSI N18.17-1973) and has determined that the program is acceptable.



KEY:

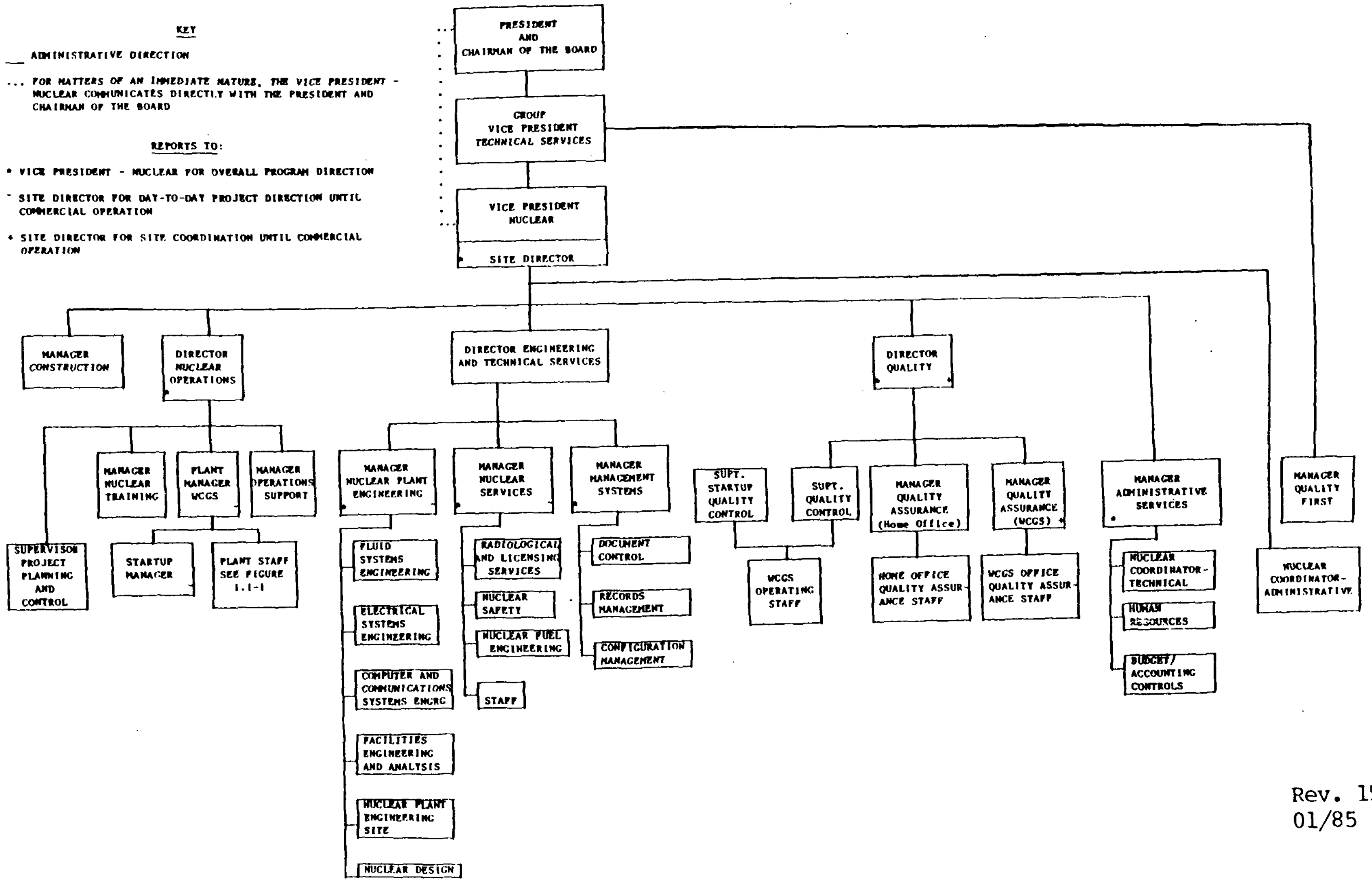
— Administrative direction

--- For matters of an immediate nature, the Vice President-Nuclear communicates directly with the President/Chairman of the Board

-•- For matters relating to nuclear procurement the Director Purchasing reports to the Group Vice President - Technical Services

Rev. 10
4/83

Figure 13.1 Kansas Gas and Electric Company corporate organization
Source: FSAR Figure 17.2-1



Rev. 15
01/85

Figure 13.2 Kansas Gas and Electric Company organization for nuclear operations
Source: FSAR Figure 17.2-2

PLANT MANAGER

- Supt of Maint
  - Maint Svcs Supvr
    - Maint Engr
    - Mat Cont Supvr
      - Mgr Coord Supvr
      - Storeroom Supvr
      - Warehouse Attd
    - Utility Supvr
      - Utility Mech
      - Bldg Serviceman
  - Maint Supp Supvr
    - Mech Supvr
      - Welder
      - Machinist
      - Mechanic
      - Utility Helper
    - Electrical Supvr
      - Electrician
      - Utility Helper
- Supt of Operations
  - Ops Coord-Operations
    - Shift Supvr
      - Supervising Opr
      - Reactor Operator
      - Station Operator
      - Util Helper/Cadet Tech
  - Ops Coord-Planning and Projects
    - Engineer/Specialist
    - Surveillance Coord
      - Systems Analyst
- Supt of Tech Supp
  - Reactor Eng Supv
    - Reactor Eng
  - I&C Supvr
    - Eng/Spec
    - I&C Coordinator
      - I&C Spec
      - I&C Tech
      - Utility Helper
    - Lead Computer Eng
      - Comp Eng/Spec
      - App Prog Train
  - Site Chemist
    - Chemistry Supv
      - HP/Chem Tech
      - Utility Helper
  - Site Health Phys*
    - Health Physics Supvr
      - HP/Chem Tech
      - Utility Helper
- Supt of Plant Supp
  - Fire Prot Spec*
  - Chief of Security*
    - Security Staff
  - Results Eng Supvr
    - Results Eng (M)
    - Results Eng (E)
  - Data Management System
    - Data Base Analyst
    - Asst Supv Comp Ops
      - Technician
      - Computer Opr
    - Asst Supv Syst Sftwr
    - Data Comm Engr Spec
      - Comp Syst Tech
- Supt of Regulatory Quality and Admin Services
  - Site Emerg Planning Administrator
  - Safety Specialist
  - Docu Cont Supvr
    - Docu Cont Clerk
    - Anal & Film Opr
  - Admin Supvr
  - Nuc Medical Spec - PA
  - Eng Spec

* For technical matters of an immediate nature, the respective individual reports directly to the Plant Manager.

Rev. 15
01/85

Figure 13.3 Wolf Creek Generating Station organization
Source: FSAR Figure 13.1-1

15 ACCIDENT ANALYSIS

15.2 Moderate Frequency Transients

15.2.1 Anticipated Transients Without Scram

SER Section 15.3.8 notes that the applicant is required to have procedures for mitigating anticipated transient without scram (ATWS) events. As reported in Section 22.2, Item I.C.1, of this supplement, the staff has approved the Westinghouse Owners Group Emergency Response Guidelines, and the applicant has committed to implement a program of emergency operating procedure (EOP) development based on these guidelines. Because the guidelines include appropriate actions for mitigating an ATWS, the staff concludes that the applicant has adequately responded to the guidance of NUREG-0460 for having EOPs for mitigating an ATWS.

15.2.3 Increased Core Reactivity Transients

15.2.3.1 Boron-Dilution Events

As stated in the SER, the applicant committed to implement certain design changes to ensure against inadvertent boron dilution. The applicant submitted these design changes in FSAR Revision 13. The SER also stated that the staff will conduct a confirmatory review of the boron-dilution-related design changes and transient analysis. This review is discussed below.

An inadvertent boron dilution can be caused by the opening of the primary water makeup control valve and failure of the blend system, either by controller or mechanical failure. The chemical volume and control system (CVCS) and reactor makeup water storage (RMWS) system are designed to limit, even under various postulated failure modes, the potential rate of dilution to values that, with indication by alarms and instrumentation, will allow sufficient time for automatic or operator response (depending on the mode of operation) to terminate the dilution. An inadvertent dilution from the RMWS may be terminated by closing the primary water makeup control valve. All expected sources of dilution may be terminated by closing isolation valves in the CVCS (valves BG-LCV-112B and C). The lost shutdown margin (SDM) may be regained by opening isolation valves to the refueling water storage tank (RWST) (valves BN-LCV-112D and E), thus allowing the addition of 2000-ppm borated water to the reactor coolant system (RCS).

The boron-dilution-related design changes consist of a safety-grade flux doubling detection system (environmentally qualified, seismic Category I, single-failure proof, and powered by a Class 1E supplier). When the system detects a neutron flux doubling, it will sound an alarm in the control room and initiate valve movements to both terminate the boron dilution and start boration. The system function is accomplished by closing valves BG-LCV-112B and C and opening valves BN-LCV-112D and E.

The status of the RCS makeup is continuously available to the operator by

(1) indication of the boric acid and blended flow rates

(2) CVCS and RMWS pump status lights

(3) deviation alarms, if the boric acid or blended flow rates deviate by more than 10% from the preset values

(4) when the reactor is subcritical, source range neutron flux indications of

  (a) high flux at shutdown alarm
  (b) indicated source range neutron flux count rate
  (c) audible source range neutron flux count rate
  (d) source range neutron flux doubling alarm

(5) when the reactor is critical

  (a) axial flux difference alarm (reactor power $\geq$50% rated thermal power)
  (b) control rod insertion limit low and low-low alarms
  (c) overtemperature ΔT alarm (at power)
  (d) overtemperature ΔT turbine runback (at power)
  (e) overtemperature ΔT reactor trip
  (f) power range neutron flux: high, both high and low setpoint reactor trips

The applicant has evaluated the inadvertent boron-dilution event during all modes of operation.

Dilution During Refueling

Inadvertent boron dilution is prevented during this mode of operation by administrative controls that isolate the unborated water sources by locking closed CVCS valves V-178 and V-601. These valves block all flow paths that could allow unborated water to reach the reactor coolant system.

Dilution During Subcritical Operation

The applicant has analyzed the boron-dilution transient during the cold shutdown, the hot shutdown, and the hot standby modes of operation. The applicant used conservative values for the analysis parameters (i.e., low initial boron concentrations (minimum shutdown margins), high critical boron concentration, high boron worths, small reactor coolant volumes, and high dilution rates). Using these assumptions resulted in conservatively short times to reach criticality. However, the boron-dilution mitigation system automatic function can be successfully completed despite a single active failure to terminate the event before the shutdown margin is exhausted.

Dilution During Critical Operation

The applicant has analyzed the boron-dilution transient during the startup and power operation modes. In these modes, the transient is delayed by any of the several trips discussed above. The operator is alerted by these automatic trips or by the several alarms that precede the trip signals (overtemperature ΔT approach alarm, control rod insertion low limit alarm, and low-low limit alarm when the control rods are operated in the automatic mode).

Using conservative assumptions, the applicant calculated that after the operator has been alerted to the transient by the above alarms and/or trips, 40 minutes will elapse before return to criticality, assuming no operator action. The staff finds that this time is adequate for the operator to diagnose and terminate the dilution event.

Conclusion

The staff concludes that Wolf Creek is adequately protected against an inadvertent boron-dilution transient. Therefore, the boron-dilution issue, Confirmatory Item B(13), is now resolved.

15.2.3.3 Rod Cluster Control Assembly Malfunction

The Wolf Creek SER did not indicate any problem for the dropped control rod(s) event (FSAR Section 15.4.3). However, since the SER was issued, a generic problem relating to the event has been recognized, and interim operating restrictions affecting automatic control or control rod insertion above 90% power were developed for operating reactors. (This was reported in SSER 1.) Westinghouse has developed a solution for the problem via methodology for analyzing the event and has described it in Topical Report WCAP-10297P. This report and its methodology have been evaluated by the staff and approved. The staff evaluation was included with the memorandum to F. Miraglia from L. Rubenstein, dated March 2, 1983, "Review of the Westinghouse Report 'Dropped Rod Methodology for Negative Flux Rate Trip Plants.'" The solution requires a reactor-cycle-specific analysis showing that departure from nucleate boiling (DNB) limits will not be exceeded. FSAR Revision 12 included a discussion of this analysis, and the results for cycle 1 operation indicate that DNB limits will be met for this cycle.

Thus, an operating restriction above 90% full power will not be necessary for cycle 1; therefore, License Condition B(18) is removed. Each future reload cycle will require similar cycle-specific analysis as part of the normal reload analysis.

15.3 Infrequent Transients and Postulated Accidents

15.3.2 Steamline Rupture

In FSAR Revision 14, the applicant proposed removing the boron concentration requirement of 20,000 ppm in the boron injection tank (BIT). Also, the applicant, in a letter dated March 20, 1984, proposed changes that would delete the Technical Specifications that address the boron concentration and heat tracing for the BIT.

The BIT was incorporated in the plant design primarily to mitigate the consequences of postulated steamline break events. During these events, the high-head safety-injection (HHSI) pumps automatically align to discharge through the BIT, which contains highly concentrated boric acid solution (20,000 ppm). This solution is then flushed into the primary system to ensure adequate shutdown reactivity.

In the proposed design change, the HHSI pumps would take suction from the refueling water storage tank, which contains borated water at a concentration of 2000 ppm, then discharge into the primary system via the BIT, which is now assumed to contain a fluid with 0 ppm boron concentration. To justify diluting the boron concentration, the applicant reanalyzed the steamline break accident using the same computer codes and assumptions as used in the previous analysis. The results indicate that the reactor returns to power with a maximum heat flux of approximately 18% of the design value and a corresponding reactor coolant system pressure of approximately 700 psia, which is well below 110% design pressure. The minimum departure from nucleate boiling ratio remains above the 1.30 limit.

Although limited clad perforation following a steamline break event is permitted by the SRP, the applicant has demonstrated that no cladding perforation is calculated to occur. Therefore, on the basis of its review of the applicant's evaluation, the staff concludes that there is no significant change in the safety margin.

With regard to the proposed changes of the Technical Specifications and heat tracing for the BIT, the applicant stated that the current requirement was the result of high boron concentration in the BIT and associated piping. Reducing the boron concentration to 0 ppm would eliminate all Technical Specifications concerning BIT boron concentration, temperatures, and associated surveillance including heat tracing, because heat tracing would be required only for boron concentrations above 4 weight percent, corresponding to approximately 7000 ppm.

The staff expressed concern that, during emergency boration, boron solution of a concentration higher than 2000 ppm may be discharged through the BIT and the associated piping. With heat tracing removed, boron precipitation could occur and block or disable the flow path through the BIT. The applicant's letter dated April 17, 1984, indicated that the emergency boration procedures would be modified to contain steps to require flushing of the BIT and the associated piping with water from the refueling water storage tank immediately after emergency boration. This would ensure that precipitation would not occur, because precipitation for a 2000-ppm solution will not begin until the temperature decreases to 33°F. The applicant has indicated that the auxiliary building design temperature is 65°F. The staff finds the applicant's response acceptable.

On the basis of its review of the applicant's evaluation, the staff concludes that there is no significant reduction in the safety margin and that SRP 15.1.5 is met. Thus, the staff finds the applicant's proposal acceptable.

15.3.7 Loss-of-Coolant Accident

By letter dated April 17, 1984, the applicant provided an evaluation of delaying the diesel generator start time by 2 seconds and the resultant consequences on the Chapter 15 accident analyses. The only accidents potentially affected by the delay are the large-break loss-of-coolant accident (LOCA) and the main steamline break (MSLB), which require safety injection. The applicant has concluded, and the staff agrees, that the effect of the 2-second delay on the steamline break is insignificant.

For the LOCA, the applicant reanalyzed the worst break using the 1978 Westinghouse evaluation model with some additional changes (WCAP-9220-P-A) and the 2-second increase in diesel generator start time. As noted in WCAP-9220-P-A, the 1978 model is no longer an approved model, and it is not acceptable to include selected modifications to a model on an ad hoc basis. The applicant also noted that if the currently approved 1981 model was used, significant margin would exist. A review of comparative calculations for other plants has shown that this is the case. However, since the analysis of record for SNUPPS was performed with the 1978 model, the staff requested additional documentation to ensure that the concerns leading to the formulation of the 1981 model were properly addressed by SNUPPS. From January 1980 to April 1982, the staff accepted analyses for Westinghouse-designed plants using the 1978 model and an interim assessment (December 7, 1979) of other model changes including cladding swelling and rupture. The applicant has provided the assessment (May 2, 1984) using the methodology outlined by Westinghouse (December 7, 1979). Although the staff SER for the 1981 model does not approve use of this methodology after April 1982, the staff finds that it does meet the requirements of 10 CFR 50, Appendix K, and therefore, concludes that the analysis and the assessment presented by the applicant (April 17 and May 2, 1984) are acceptable on an interim basis. The peak cladding temperature calculated for the worst break (April 17, 1984), including the 2-second increase in diesel start time, is 2174°F. The applicant's May 2, 1984, submittal shows that the penalty associated with incorporating the modified cladding models is more than offset by the benefits of a more accurate thermal-hydraulic blowdown model (upper-head injection (UHI) technology). In an April 23, 1984, letter, the applicant provided additional sensitivity studies on plants very similar to SNUPPS plants. These studies showed that the benefit afforded by UHI technology's approved 1981 model is expected to be more than that allowed by the interim assessment. Therefore, the staff concludes that the applicant meets the requirements of 10 CFR 50.46, subject to confirmation, using the staff-approved model described in WCAP-9220-P-A.

Therefore, the following condition will be placed on the Wolf Creek license: Before startup following the first refueling outage, the applicant shall perform a reanalysis for the worst large break using an approved evaluation model. At this time that model is the 1981 Westinghouse model with consideration of maximum safety injection as the worst single-failure condition. A modified version of the 1981 model that includes the BART computer code, which has been approved by the staff, may be used.

15.4 Radiological Consequences of Design-Basis Accidents

15.4.4 Steam Generator Tube Rupture

Before the Ginna steam generator tube rupture (SGTR) in January 1982, the staff review of SGTR covered only radiological consequences. In the FSAR, the applicant made general, unverified assumptions concerning system performance following a complete severance of a single steam generator (SG) tube. In addition, the FSAR assumed that the break flow is terminated within 30 minutes of the event by operator actions to equalize the primary and secondary pressures. the SER, the staff addressed the accident, including the sequence of events and the radiological consequences, and found them acceptable. However, the SGTR at Ginna indicated that more than 30 minutes is required for pressure equalization.

As a result of the Ginna event, the staff required additional information including an evaluation of operator response time, whether liquid can enter the steamlines, and what the effects would be on the integrity of steam piping and supports. The staff also required verification that the components that are credited in the analysis to mitigate the consequences of the SGTR are classified as safety related. This information was to be provided for both the "offsite power available" and "loss of offsite power" (LOOP) cases.

In its submittal of March 16, 1984, the applicant provided a preliminary response to these questions. The response indicates there will be an equalization of RCS and faulted SG pressure (leak termination) in 40 minutes, and termination of safety injection (SI) in 45 minutes after the event, based on initial operator action terminating auxiliary feedwater (AFW) to the faulted SG in 16 minutes. RCS cooldown would be initiated in 32 minutes; the reactor coolant pumps (RCPs) would be kept in operation, based on the premise that offsite power is retained. The analysis justifies the time for initial operator action based on (1) SGTR simulator data and (2) the fact that control of SG level is a common operation. The applicant's analysis assumes that RCS and SG pressures equalize 8 minutes after the initiation of RCS cooldown.

With regard to the question of whether liquid can enter the main steamlines as a result of the SGTR, the applicant has performed an analysis that includes the effects of single failure and has identified failure of the AFW valve in the open position as having the potential of causing overfill. The applicant also indicates that (1) the SNUPPS steamlines and supports have been designed for the case in which the steamlines filled with water, and (2) the possibility of damaging water hammer is extremely remote. The applicant should address the question of whether the safety valves would function properly if they are actuated with liquid in the steamlines.

With regard to the safety classification of components that are credited to mitigate the consequences of SGTR, the applicant states that all of the components are safety related.

The staff has reviewed this information and has a number of questions that the applicant is being requested to address. For example, the staff is not convinced that SNUPPS has determined the most limiting single failure with respect to SG overfill. Also, although the staff agrees that control of the SG level is a common operation, the staff cannot agree that level control following a ruptured SG tube is routine and straightforward. These and other questions are being forwarded to the applicant. This issue will be resolved before startup following the first refueling outage.

As to whether there is adequate assurance that the Wolf Creek Generating Station can operate safely for one cycle of operation, until the SGTR issue is satisfactorily resolved, the staff notes the following: (1) All components necessary for mitigation of the design-basis SGTR are safety related, (2) the Wolf Creek steamlines and supports are designed for the situation in which the streamlines are filled with water, and (3) there is a low probability of a design-basis SGTR in the first cycle of operation. The staff concludes that the applicant's submittal provides adequate assurance that the Wolf Creek Generating Station can operate safely for one cycle until this issue is resolved. The staff will condition the operating license to require satisfactory resolution of this issue before startup following the first refueling outage.

Additional information has become available regarding operator actions and the associated times to mitigate the consequences of SGTR. On the basis of recent plant simulator runs and preliminary thermal hydraulic calculations performed by Westinghouse, operator action can be expected within a time frame compatible with mitigation of SGTR consequences. Thus, termination of primary to secondary leakage by pressure equalization can be expected within a time frame necessary to prevent steam generator overfill. The staff continues to believe that the consequences of an SGTR at Wolf Creek can be adequately controlled by limiting the primary and secondary coolant system radioactivity concentrations by Technical Specifications and by proper operator actions. The recent information regarding operator actions, delay times, and time to overfill indicates that sufficient time is available for power operator actions to maintain the offsite radiological consequences below the staff's acceptance criteria. The staff further concludes that, subject to the receipt of the confirmatory information, the Wolf Creek SGTR analysis is acceptable.

16 TECHNICAL SPECIFICATIONS

The Technical Specifications of a license define certain features, characteristics, and conditions governing operation of the facility that cannot be changed without prior approval of the staff. During its review of the Wolf Creek application, the staff identified certain issues that must be included in the Technical Specifications as a condition of staff acceptance. These 18 issues are delineated in the SER. As a result of the staff's review of additional information received after the SER was issued, eight new items that must be included in the Technical Specifications. These items and the sections of this supplement in which they are addressed are as follows:

prt(19) Analysis of steam generator (Section 3.9.3.1)

(20) Calibration of special test instrumentation used in the calorimetric measurement (Section 4.4.4.3)

(21) Inspection and cleaning of feedwater venturi (Section 4.4.4.3)

(22) Secondary water chemistry (Section 5.4.2.3)

(23) Operation of containment purge system and leakage rate (Section 6.2.3)

(24) RG 1.97 exceptions (Section 7.5.2.3)

(25) Surveillance requirements for operability of the fire detection and suppression system (Section 8.2.2.1)

(26) Surveillance requirements for the operability of the load sequencer (Section 8.2.2.3)

17 QUALITY ASSURANCE

17.1 General

By Revisions 12, 13, 14, and 15 to the Wolf Creek Addendum to the SNUPPS FSAR, the applicant reported changes to its organization and program for quality assurance (QA) during the operations phase of the Wolf Creek Generating Station (WCGS). The former Quality Assurance Division under a Manager Quality Assurance has been renamed the Quality Branch under a Director Quality. The Quality Control organization is now part of the Quality Branch rather than part of the plant staff. The organizations of the Director Nuclear Operations and the Manager Nuclear Plant Engineering have also been changed.

17.2 Organization for the QA Program

The changed Kansas Gas and Electric Company (KG&E) organization for nuclear operations, which includes the Project Director as noted in SSER 4 is shown in Figure 13.2. The President and Chairman of the Board is reponsible for promulgating quality program requirements for WCGS and endorses the KG&E QA policy statement. The Vice President - Nuclear under the Group Vice President Technical Services is responsible for implementing the KG&E QA policy and the QA program resulting from the policy and has corporate responsibility for the operations phase of Wolf Creek. He and the Project Director who reports to him direct activities of those involved in the operations phase of the WCGS.

The Director Nuclear Operations reports to the Vice President - Nuclear and is responsible for the operations, training, and startup departments for Wolf Creek. The Plant Manager reports to the Director Nuclear Operations for overall program direction and he receives day-to-day project direction from the Wolf Creek Project Director. The Plant Manager is responsible for the safe operation of the plant, and the plant staff reports to him.

The Director Quality reports directly to the Vice President - Nuclear and is responsible for developing the operating quality program for Wolf Creek and assuring its implementation. He is responsible for staffing the Quality Branch and for ensuring that QA and quality control (QC) personnel are adequately trained and experienced to perform their assigned tasks. The Superintendent Quality Control reports to the Director Quality and is responsible for the conduct of operating QC activities and personnel at Wolf Creek.

The Manager Quality Assurance (Home Office) reports to the Director Quality and is responsible for verifying that an adequate QA program is developed and implemented for safety-related Wolf Creek activities that occur at the corporate office and at other locations remote from Wolf Creek. He is responsible for establishing and implementing a comprehensive audit program for offsite activities of KG&E and its suppliers, consultants, and agents.

The Manager Quality Assurance at Wolf Creek reports to the Director Quality and is responsible for verifying that an adequate QA program is developed and implemented for safety-related activities that occur at Wolf Creek. He is responsible for establishing and implementing a comprehensive plant site audit program.

The Manager Nuclear Plant Engineering reports to the Vice President - Nuclear He is responsible for station modifications, additions, engineering studies, and design reviews that are conducted at the general office or subcontracted by the general office to an outside organization.

The Manager Nuclear Services reports to the Vice President - Nuclear. He is responsible for providing services in the areas of licensing, fuels management, fuel procurement, and safety analysis. He is responsible for home office support of the plant in nuclear engineering chemistry, health physics, and environmental areas. He is also responsible for processing and maintaining records for the KG&E general office.

The Manager Management Systems reports directly to the Vice President - Nuclear and is responsible for configuration control, document control, and records management programs. Through these programs, he (1) defines and documents the plant configuration, (2) provides evaluation of changes and implementation of changes to the plant configuration, (3) verifies incorporation of approved changes, and (4) provides configuration records to support plant operation.

The Manager Nuclear Administrative Services reports directly to the Vice President - Nuclear and is responsible for providing administrative assistance to him.

The Director Quality and his staff have the authority, delineated in writing to (1) identify QA problems; (2) initiate, recommend, or provide solutions through designated channels; and (3) verify implementation of solutions. Disputes arising between QA/QC personnel and personnel in other applicant organizations that cannot be resolved shall be referred to the next appropriate higher level of management for resolution. Disputes that cannot be resolved through lower levels shall ultimately be resolved by the Vice President - Nuclear.

The KG&E organization for nuclear QA as provided in FSAR Revision 15 is shown in Figure 13.2. Three positions are shown on that figure which were not shown previously. These positions and their responsibilities are given below,

(1) Manager Construction - provides construction support at the Wolf Creek Generating Station during the operations phase.

(2) Director of Engineering and Technical Services - provides overall program direction to the Manager Nuclear Services; the Manager Nuclear Plant Engineering; and the Manager Management Systems.

(3) Manager Quality First - directs KG&E's "quality first" program which investigates safety and quality concerns.

17.3 Quality Assurance Program

The QA program for the operations phase of Wolf Creek is developed in accordance with the Wolf Creek Project Policy Manual that includes, among other things, (1) a policy statement by the President and Chairman of the Board describing the QA efforts to be applied at Wolf Creek and (2) QA directives by the Vice

President - Nuclear that specify responsibilities and authority of each organization under the Vice President - Nuclear and provide uniform direction to these organizations. The policy and directives are implemented through procedures including QA and QC procedures in the Quality Program Manual. These documents present the detailed techniques and methods by which the requirements of Appendix B to 10 CFR 50 and the provisions of the NRC regulatory guides regarding QA are satisfied (see SER Table 17.1).

The QA program requires that QA documents encompass detailed controls for (1) translating codes, standards, and regulatory requirements into specifications, procedures, and instructions; (2) developing, reviewing, and approving procurement documents, including changes; (3) prescribing all quality-affecting activities by documented instructions, procedures, or drawings; (4) issuing and distributing approved documents; (5) purchasing items and services; (6) identifying materials, parts, and components; (7) performing special processes; (8) inspecting and/or testing material, equipment, processes, or services; (9) calibrating and maintaining measuring and test equipment; (10) handling, storing, and shipping of items; (11) identifying the inspection, test, and operating status of safety-related items; (12) identifying and dispositioning nonconforming items; (13) correcting conditions adverse to quality; (14) preparing and maintaining QA records; and (15) auditing activities that affect quality.

An indoctrination and training program is conducted under the auspices of the Manager Nuclear Training to ensure that persons involved in quality-related activities are knowledgeable in QA instructions and implementing procedures and that proficiency in performing these activities is maintained.

Quality is verified through surveillance, inspection, testing, checking, and audit of work activities. The QA program requires that quality verification and inspections be performed by qualified QC inspectors who are not directly responsible for performing the actual work activity. Inspections are performed with procedures, instructions, and/or checklists by inspectors who have been qualified and certified in accordance with codes, standards, or company training programs.

The Managers Quality Assurance are responsible for establishing and implementing the audit program. Audits are performed, with written procedures or checklists, by qualified personnel not having direct responsibility in the areas being audited. The QA program establishes a comprehensive audit system to ensure that the QA program requirements and related supporting procedures are effective and properly implemented during operations. Audits will incude an objective evaluation of QA practices, procedures, and instructions; work areas, activities, processes, and items; and the effectiveness of implementation of the QA program and conformance with policy directives.

The QA program requires documentation of audit results and review by management personnel having responsibility in the area audited to determine and take corrective action as required. Reaudits are performed to determine that nonconformances are effectively corrected and that the corrective action precludes repetitive occurrences. Audit findings, which indicate quality trends and the effectiveness of the QA program, are reviewed by the Quality Branch and are reported to appropriate management on a regular basis.

17.4 Conclusions

The staff conclusions stated in the SER regarding KG&E's QA program description for the operations phase of Wolf Creek are unchanged.

Therefore, the staff concludes that the applicant's description of the QA program is in compliance with applicable NRC regulations.

17.5 Additional Assurance Associated With the Design Process Used at Wolf Creek

17.5.1 Background

By letter dated January 4, 1984, the NRC staff stated that it had been seeking additional assurances from applicants for operating licenses that the design process used in constructing their plant had fully complied with NRC regulations and licensing commitments. The staff noted that an integrated design inspection (IDI) was performed on Callaway Unit 1 by the NRC Office of Inspection and Enforcement. Because Wolf Creek Unit 1 was of similar design to Callaway Unit 1, the staff stated it was considering whether this and possibly other factors may support a conclusion that the design process for Wolf Creek Unit 1 had met NRC regulations and licensing commitments. To assist the NRC staff in making its decision with respect to Wolf Creek Unit 1, the applicant was requested to provide the following:

(1) a summary of the differences in the design process between Wolf Creek Unit 1 and Callaway Unit 1 (both plants are standard SNUPPS designs)

(2) a discussion of the effect of these differences (item 1) on the applicant's confidence that the design process for Wolf Creek is at least equivalent to that for Callaway Unit 1

(3) a discussion of how applicable Callaway Unit 1 integrated design inspection report findings had been addressed for Wolf Creek Unit 1

(4) a discussion of the quality assurance program related to design that assured that the applicable design commitments were implemented at Wolf Creek Unit 1

(5) any other information that would support the conclusion that the design process for Wolf Creek Unit 1 has been properly implemented.

A meeting was held on February 6, 1984, at Bethesda, Maryland, between the NRC staff, KG&E, Bechtel, and SNUPPS to discuss the applicant's design process used at Wolf Creek Unit 1.

The applicant's letters dated March 9 and August 15, 1984, responded to NRC's request for additional information regarding the design process for Wolf Creek Unit 1. These letters in combination with the Callaway IDI provide the basis for the staff's assessment.

17.5.2 Licensee-Furnished Information

17.5.2.1 SNUPPS Standard Design

The SNUPPS standard design process involves a single generic design applicable to both Callaway and Wolf Creek units. The SNUPPS standard design concept applies to the reactor (containment), auxiliary, turbine, diesel generator, fuel, control and radwaste buildings, several external storage tanks, transformers, and vaults, and is referred to as the "power block." Design of the power block is based on meteorological, hydrological, geotechnical, and seismological characteristics that envelope the two sites. Responsibility for the standard plant design has been assigned by Kansas Gas and Electric Company and Union Electric Company to the lead architect/engineer (AE), Bechtel Power Corporation (Bechtel). The two units use identical nuclear steam supply systems (NSSSs) and turbine-generator systems furnished by Westinghouse and General Electric, respectively. The design of the standard power block, including related stress and seismic analyses, is by the lead AE, Bechtel. Bechtel is also responsible for integrating design of the nuclear steam supply and turbine generator systems, thus providing a single point of interface control between the AE and principal design contractors. The entire power block effort has, since project inception, been carried out under a full-scope 10 CFR 50, Appendix B, "Quality Assurance Program." Utility review and administration of the standard power block design effort is coordinated through the SNUPPS staff who are employees of Nuclear Projects, Inc. Responsibility for design of plant and site features outside the power block (e.g., ultimate heat sink, essential service water pumphouse, excavation, and backfill) is retained by the individual utilities.

Implementation of standard design and design commitments cited in the FSAR is accomplished through the development of standardized design criteria; system description; piping and instrumentation drawings (P&IDs); logic and schematic drawings and detailed design drawings simultaneously issued to each site for installation and erection. Design details and features are supported by standard engineering analyses, calculations, verification testing, and computer codes. Materials and equipment procured for each plant use the same standard specifications or material requisitions. Vendor-generated drawings, process procedures, and test specifications are also standardized and apply to both SNUPPS units. All design activity within the lead AE scope of work is accomplished by a Bechtel-SNUPPS project design team. Site liaison personnel are assigned by disposition of selected categories of Nonconformance Reports (NCRs) and Field Change Requests (FCRs); however, there is essentially no field engineering activity performed at either site. Design activity is performed on a generic basis at the home office. This applies to all elements of design including piping layout, instrument tubing design, pipe support and conduit (except for lighting and communications) detailing, cable tray and support design, and structural and rebar detailing.

The preceding description for the standard power block design applies to all design activities within the power block, regardless of safety classification.

Limited site-specific design features are contained within the power block. Mainly, these are in the form of instrumentation and controls for equipment located outside the power block. Elsewhere, nonstandard design features within the SNUPPS power block are limited to those resulting from resolution of field problems and deficiencies. Control and resolution of field-identified problems is accomplished through use of generic project procedures developed to ensure that standardization of design is effectively controlled and that any required deviations are rigidly managed.

17.5.2.2 Wolf Creek Specific Design

The standard power block comprises approximately 95% of the design work required for Wolf Creek. The Wolf Creek site-specific design work accounts for the remaining 5%.

Safety-related design work that is specific for Wolf Creek includes the essential service water (ESW) pumphouse, the ESW pipes/ductbank corridor, the ESW discharge structure, the excavation and backfill for safety-related structures, the ultimate heat sink and the 10 CFR 100 site investigation analyses. The delineation of major design responsibilities for these Wolf Creek site-specific structures and analyses follows:

(1) Dames & Moore performed the geotechnical site investigations and established the design soil parameters for the design analysis performed by both Bechtel and Sargent & Lundy.

(2) Dames & Moore assembled and reviewed historical and meteorological data specific for Wolf Creek.

(3) Bechtel designed the ESW pumphouse, the ESW pipe/ductbank corridor, and the crossover reinforcements where the ESW pipes/ductbank pass over non-safety-related underground facilities.

(4) Sargent & Lundy designed the excavation and backfill for the power block structures, the ESW pumphouse, and the ESW pipe/ductbank corridor.

(5) Sargent & Lundy designed the ultimate heat sink, including heat injection analysis and design of the basin, slopes, and dam.

The following lists the design organizations and their major responsibilities for both Callaway and Wolf Creek and shows that most of the two plants' design functions were performed by the same organizations:

Design activity	Callaway organization	Wolf Creek organization
Power block	Bechtel	Bechtel
ESW components	Bechtel	Bechtel
Excavation and backfill	Bechtel	Sargent & Lundy
Ultimate heat sink	Bechtel	Sargent & Lundy
Geotechnical consultant	Dames & Moore	Dames & Moore
Meteorological consultant	Dames & Moore	Dames & Moore

Beginning with the initial stages of the project, KG&E implemented administrative procedures for identifying and controlling both the design responsibilities and the design interface responsibilities among Sargent & Lundy, Bechtel, and Dames & Moore. Design responsibilities and design interfaces were coordinated through routine meetings, correspondence, and teleconferences among KG&E and the three design organizations, as appropriate. Each design organization's scope of work and specific interface responsibilities with the other design organizations were documented in scope, design criteria, and report documents generated by each design organization.

Development of the Wolf Creek-specific design was accomplished in accordance with written and approved procedures to ensure the accurate translation of design basis and regulatory commitments into drawings, specifications, and procedures. The applicant imposed design control requirements on each design organization performing safety-related work. Bechtel, Sargent & Lundy, and Dames & Moore are required to perform their respective scopes of design responsibilities in accordance with written procedures. Each organization's procedures, instructions, and standards describe the design process and prescribe methods for the planning, performance, verification, internal interface, and release of design work and changes to design work.

In addition to the procedural controls implemented by each design organization, the applicant has implemented written procedures for the technical review of selected documents generated by the site-specific design organizations. At the direction of the KG&E Manager Nuclear Plant Engineering, a technical review is performed on designated lead documents, including design criteria, functional descriptions, drawings, and specifications. The technical review considers operability and maintainability, compatibility between the SNUPPS design and the site design, inclusion of acceptance criteria for inspections and tests, and requirements imposed by plant operating equipment. Any comments generated as a result of the technical review are transmitted in written form to the responsible design organization for resolution and close out.

During the construction phase, changes in the design specific for Wolf Creek as a result of site interference problems, deficiencies, or material unavailability, are controlled in the same manner as described for the power block portion of Wolf Creek. Construction or startup proposed changes are documented in the same standardized FCR, NCR, and Startup Field Report (SFR) forms, and in accordance with the same procedures as for the standard power block portion of Wolf Creek. The forms are transmitted to either Sargent & Lundy or Bechtel for resolution and disposition, depending on the organization responsible for the design. Before approving a change to the design, Bechtel and Sargent & Lundy are procedurally required to check the change against the design bases, as documented in the preliminary safety analysis report (PSAR) and other design criteria documents. Both Bechtel and Sargent & Lundy were required to revise the FSAR, when affected, and to incorporate design changes into as-built documents, including design criteria specifications, construction specifications and drawings, and/or design compilation reports.

17.5.2.3 Callaway Integrated Design Inspection (IDI) Findings

Each Callaway IDI finding and unresolved item was evaluated for its applicability to Wolf Creek. Of the fifty numbered items in the Callaway Inspection Report, just four of these were concerned with specifics applicable only to the

Callaway project. They had to do with "as-built" conditions, nonconformances, and procedural controls for Callaway. Had the IDI team looked at Wolf Creek, it could have found similar nonconformances and procedures discrepancies, which potentially could have resulted in a finding or unresolved item. However, conceptually their resolution would have been the same because the same project controls and project management team are in place to deal with such situations. Therefore, corrective actions taken in response to the inspection report findings were generically applicable to both Callaway and Wolf Creek.

A meeting was held in Union Electric's corporate offices on January 27, 1984, to review the status of the resolution of all IDI items. Personnel from the NRC Office of Inspection and Enforcement, NRC Region III Office, Union Electric, Bechtel, SNUPPS, and KG&E attended. The status review indicated that the majority of the items had already been resolved. For those items yet outstanding a resolution plan was discussed and agreed upon. When all items are resolved, Region III personnel plan to document the closeout of the IDI in an inspection report. (Note that all Callaway IDI report open items were closed out by NRC Inspection Report 50-483/83-33.)

In summary, the IDI on the Callaway plant was really an inspection of the SNUPPS Project design process and the conclusions drawn from the inspection apply to Wolf Creek activities as well as to those at Callaway.

17.5.2.4 Design Quality Assurance Program

The QA controls described in PSAR Chapter 17.1 have been contractually imposed on the power block AE and NSSS supplier and, through them or through KG&E, passed on to all consultants, subvendors, and subcontractors responsible for furnishing safety-related goods and services.

Sargent & Lundy and Dames & Moore were committed to and functioned under 10 CFR 50, Appendix B, QA programs. The Sargent & Lundy scope of work was limited to design and did not include any procurement responsibilities except for the generation of specifications. Safety-related activities by Dames & Moore were limited in scope to field and laboratory testing and data assimilation and analysis. The QA Program Manual and changes thereto for both organizations were reviewed and approved by KG&E.

As with Bechtel and Westinghouse, both Sargent & Lundy and Dames & Moore used procedures and management controls to ensure that the design process progressed in accordance with 10 CFR 50, Appendix B, requirements and good engineering practice. These controls were subjected to internal auditing by independent quality assurance groups within both organizations, and any adverse finding was brought to management attention for resolution.

Another important element of the design quality assurance program involves a program of coordinated SNUPPS/utility reviews of key design features and AE-generated design documentation. At the initial stages of the power block design, a review was undertaken by the SNUPPS utilities and NPI/SNUPPS staff to identify key design documents requiring consolidated utility review. This review process was carried out by the SNUPPS Technical Committee that is made up of senior engineering representatives from each of the SNUPPS utilities. (Note: The SNUPPS utilities from 1973 to 1979, the period during which the basic plant design configuration was established, consisted of Union Electric, Kansas Gas & Electric, Kansas City Power & Light, Northern States Power, and Rochester Gas & Electric; the last two had extensive experience in the design and operation of commercial nuclear power plants.) This review process, described in SNUPPS staff and utility procedures, requires staff and utility review and approval of system descriptions, P&IDs, logic and schematic drawings, major equipment specifications, and other key design documentation developed by the AE. Individual utility design review comments and concerns are reconciled through the SNUPPS Technical Committee review process. Subsequent to this review, consolidated comments and direction are furnished to the AE for resolution before releasing the design document to the field. The generic review process is supplemented by Plant Review Group studies and assessments of selected design features of interest; e.g., human factors, inservice inspection access engineering and computer systems evaluation. (Note: The Plant Review Group consists of utility specialists selected for study of selected design features and topics and operates within the framework of the Technical Committee.) Where required, utility capability is supplemented by design reviews and studies provided by outside consultants and technical specialists under contract to SNUPPS and/or KG&E. As an example, technical specialists were used to provide independent review and assessment of the auxiliary feedwater pump design and to provide expertise in the review of generic fire protection systems.

In addition to the power block design reviews, KG&E was also involved in the technical aspects of the Sargent & Lundy, Bechtel, and Dames & Moore work through review of design documentation. Coordination and compatibility of design between the various design organizations was accomplished and controlled by KG&E.

The process of SNUPPS/utility review of key design features and documentation is described in SNUPPS staff and utility procedures and is subject to individual audit and surveillance by SNUPPS staff and by each utility QA organization. Audit findings have been systematically tracked through closeout resolution.

Independent utility verification of the management systems, design process, and interface controls used by the AE and NSSS supplier in the course of designing the Wolf Creek Generating Station has, since inception, been provided by means of a preplanned program of QA audit and surveillance. Audits of Bechtel and Westinghouse design activity have been accomplished through the SNUPPS QA Committee (representing senior utility/SNUPPS staff QA personnel) and by the SNUPPS organization. These audits provide assessments of the design process in vital areas such as

- SAR change control
- AE/NSSS design interface control
- control of standard (Bechtel) design-oriented computer programs; AE program for reconciliation of "as-built" data with final piping seismic analyses;
- design review and design change control programs
- design feedback program (from operating nuclear plants)
- NRC, FCR, and SFR processing and control

These audits have been supplemented by Quality Assurance Committee-initiated audits of major Bechtel and Westinghouse equipment subvendors, such as Combustion Engineering (reactor pressure vessel) and the Westinghouse Tampa (steam generator/pressurizer) and Pensacola (reactor internals) Divisions. These audits, which supplement the routine Bechtel and Westinghouse subvendor audit effort, were initiated because of the importance and safety significance of the specific equipment and to provide independent SNUPPS/utility examination and assessment of principal designer/subvendor design interfaces.

The SNUPPS project also has conducted technical design audits since 1977. These audits are conducted by independent, off-project designer personnel and provide focused examination and technical evaluation of key design features of the standard power block design. This effort, established by the SNUPPS utilities, is managed by the Bechtel Quality Assurance Project Manager and uses non-project, technical specialists from the various Chief Engineer staffs at the Gaithersburg, Md., office. Subject matter covered by these technical audits include

- hanger and pipe support design
- piping stress analysis and calculations
- design of pressure-relieving devices
- seismic analysis of the reactor, auxiliary, and control buildings
- HELB (high-energy-line-break) design analyses

The audits provide examination of subunit design discipline interface and provide an independent verification of design calculations and analysis used in the design process. Findings, discrepancies, or items of concern are reviewed and tracked through closeout.

The applicant has provided utility QA monitoring of the Sargent & Lundy design process through audits that were conducted during the course of the activity. Specific areas covered during these audits included

- establishment of design criteria
- design and design review process
- Sargent & Lundy internal auditing and corrective action system
- management overview
- computer program certification

The applicant also performed numerous field and home office audits and surveillances of Dames & Moore during the course of that company's role as seismological/geotechnical/meteorological consultant. Examples of the scope of these activities are

- meteorological tower instrument maintenance and calibration
- site core borings and geophysical testing
- investigation and recording of geological features
- atmospheric dispersion calculations

The design assurance program described above has been in place since the start of design of the Wolf Creek Generating Station and is considered in full compliance with NRC regulations and standards.

17.5.3 Conclusion

In the standard power block portion of the SNUPPS plants, the staff finds that no differences exist in the design process. From a review of the applicant's letters of March 9 and August 15, 1984, which includes the preceding information of Section 17.5.2, the staff finds that the needed interface controls were established that permit effective implementation of safety-related design activities for the Wolf Creek site-specific design. Also, the staff finds that the applicant's information regarding the design quality assurance is in agreement with the quality assurance program description previously and found acceptable by the staff. On the basis of the aforementioned information, the staff concludes that the applicant has demonstrated that the design process used in constructing Wolf Creek Unit 1 has complied with NRC regulations and licensing commitments, and that an independent design verification program is not necessary.

22 TMI-2 REQUIREMENTS

22.2 Discussion and Conclusions

I.A.1.1 Shift Technical Advisor

In Revision 9 to the FSAR, the applicant described its program for meeting the objective of the shift technical advisor (STA) program of Item I.A.1.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements." The proposed program would provide for upgrading the qualifications of senior reactor operators (SROs) through the college education program described in the FSAR to ensure that each shift supervisor had 60 semester-hours of college education. The content of this program met the guidelines of the Institute for Nuclear Power Operations (INPO) for STA training as listed in the INPO document appended to NUREG-0737. Whenever a shift did not have an SRO who met these guidelines, the applicant would provide a Duty/Call Technical Advisor to satisfy this requirement.

Thus, the principal emphasis of the applicant's proposal was to combine the duties of the STA/SRO through suitable upgrading of the SRO via the college education program. Since the staff's position on combining these duties had not been fully developed at the time the SER was issued, the proposal for meeting NUREG-0737, Item I.A.1.1, was left as an open item.

Since the the SER was issued, the applicant has completed the college education program for all shift supervisors and SROs. As noted in the SER, this consisted of at least 60 semester-hours of technical courses (64 semester-hours in actual practice). The applicant has been actively engaged in negotiations with the Kansas State University (KSU) system to develop a college degree program tailored to nuclear plant operations. The proposed degree would be a Bachelor of Science in Engineering Technology - Nuclear Option. In developing this program, the applicant has been guided by the existing curriculum for a Bachelor of Science Engineering Technology - Environmental Engineering and has proposed to replace 18 hours of this program with the training given at the site in the STA and SRO training program. The applicant has prepared the curriculum for this training and is seeking independent accreditation through the American Council of Education (ACE). This was accomplished in the fall of 1984 and proposed to the Kansas Board of Regents in early 1985 as a formal part of the new degree program. Thus, if adopted by the KSU system, all SROs will have completed the entire technical portion of the Bachelor of Science in Engineering Technology and need only to complete certain requirements in humanities, social science, and core liberal arts courses to have the degree conferred on them.

On August 1, 1984, the applicant met with the staff to discuss this program. The staff pointed out that current guidelines for engineering expertise on shift in order to combine the STA/SRO duties would require as one option that the individual had completed the entire technical portion of an accredited technical degree program (such as Engineering Technology). The normal requirements, in practice, for the technical portion of such degrees are 80 semester hours. The staff agreed that if the Kansas Board of Regents adopted the Engineering Technology - Nuclear Option program, this option for achieving the necessary engineering expertise on shift would be met.

The staff took note of the fact that nearly half of the 18 hours of new course work that the applicant was proposing to the Board of Regents had already received ACE accreditation, and that the program was receiving support from the KSU College of Engineering. The engineering faculty of KSU was reviewing the curriculum to determine how many credit hours could be granted for the material taught by the applicant.

The applicant indicated its intention to provide academic training for all SROs in the complete technical portion of the Engineering Technology degree program even if the Board of Regents should not adopt the full 18 hours proposed by the applicant and should substitute other courses. The applicant made a verbal commitment to complete the program as adopted by the Board of Regents on a schedule that would, of course, be largely dictated by the academic calendar of the university. Should the program proposed by the applicant be adopted in full, each SRO had in fact already completed the technical portion of the Engineering Technology degree.

The staff asked for this commitment in writing and received confirmation in the applicant's letter of August 7, 1984. This commitment will also be contained in a future revision to the FSAR.

The staff has now completed its review of the applicant's proposed program for providing the necessary on-shift engineering expertise and finds the use of a dual function STA/SRO acceptable. This closes staff action on NUREG-0737, Item I.A.1.1.

I.C.1 Short-Term Accident Analysis and Procedures Revision

The staff provided guidance for upgrading Emergency Operating Procedures (EOPs) in the SER. The schedule and review requirements for TMI Action Plan Item I.C.1 have been modified by Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability (Generic Letter 82-33)," dated December 17, 1982.

Supplement 1 to NUREG-0737 requires that technical guidelines be submitted to the NRC for review. For Wolf Creek Unit 1, this requirement is satisfied by the applicant's commitment in the Wolf Creek Generating Station Procedures Generation Package (PGP) (letters dated November 28, 1983, and August 10, 1984), to implement upgraded EOPs based on the Westinghouse Owners Group Emergency Response Guidelines (ERGs), Revision 1, without any deviations. The NRC staff approved ERGs, Revision 1, for implementation (letter dated December 24, 1984).

Supplement 1 to NUREG-0737 also requires that each applicant shall submit to NRC a PGP at least 3 months before the date the applicant plans to begin formal operator training on the upgraded procedures. The PGP shall include

(1) plant-specific technical guidelines
(2) a writer's guide
(3) a description of the program for validation of EOPs
(4) a description of the training program for the upgraded EOPs

Review criteria for the PGPs are not currently included in the SRP. Review criteria for PGPs are being developed based on (1) on the experience gained in performing the reviews for Item I.C.8 and (2) NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures" (which is the reference document for the EOP upgrade portion of Supplement 1 to NUREG-0737). NUREG-0899 identifies the elements necessary for licensees and applicants to prepare and implement EOPs that will provide the operator with directions to mitigate the consequences of a broad range of accidents and multiple equipment failures. In addition to identifying these elements, NUREG-0899 also outlines the process by which licensees and applicants should develop, implement, and maintain EOPs. Finally, to ensure that the elements are addressed in the new or upgraded procedures and that acceptable processes of development, implementation, and maintenance are followed, the staff will review the PGPs to ensure that EOPs written or upgraded according to a given plant's program will be acceptable.

The staff has reviewed the Wolf Creek Unit 1 PGP, submitted by letter from the applicant dated November 28, 1983, and additional information requested by the staff that was submitted in a revised PGP by letter from the applicant dated April 4, 1984. The staff review was conducted to determine the adequacy of the applicant's program for preparing and upgrading EOPs. NUREG-0899 and Supplement 1 to NUREG-0737 were used as the bases for the review. The review consisted of an evaluation of the applicant's (1) plant-specific technical guidelines, including the planned method for developing plant-specific EOPs from approved generic technical guidelines that are based on the reanalysis of transients and accidents, as described in NUREG-0660, Section I.C.1, and clarified in Item I.C.1 of NUREG-0737; (2) plant-specific writer's guide, detailing the specific methods to be used in preparing EOPs based on the technical guidelines to ensure that the EOPs are usable, accurate, complete, readable, convenient to use, and acceptable to control room personnel; (3) program for verifying and validating EOPs to ensure that they accurately reflect the technical guidelines and the writer's guide, and that the EOPs will guide the operator in mitigating transients and accidents; and (4) program for training operators on EOPs to ensure that the operators will be adequately trained before implementing the upgraded EOPs.

The applicant's PGP provides reasonable assurance, with the exception noted in the following paragraph, that the resulting EOPs will be based on approved technical guidelines that address a wide range of multiple and consequential failures, that the EOPs will be acceptable and usable, by operators, that the accuracy and usability of the EOPs will be validated, and that the operators will be trained on the EOPs before the procedures are implemented.

The exception is that the applicant needs to describe the process that was or will be used to derive the instrumentation and control characteristics from the information contained in Revision 1 of the generic guidelines and related background information. The function and task analysis, as required by Supplement 1 to NUREG-0737, applies to both the upgrade of EOPs and the Detailed Control Room Design Review (DCRDR) (TMI Action Plan Item I.D.1). Instrument characteristics include parameter, parameter type, dynamic range, setpoints, resolution/accuracy, speed of response, units, and the need for trending. Control characteristics include type (discrete or continuous), discrete functions (e.g., On, Off, Auto), rate, gain, response requirements, transfer function, criticality, and frequency of use. For the characteristics that cannot be derived from the ERGs and background documentation, the applicant needs to describe the process that was or will be used to generate information and control needs (from transient and accident analysis and associated information) that will then be used to derive instrumentation and control characteristics. For deviations from the generic instrumentation and control characteristics, or development of its own needed characteristics, the applicant needs to identify the deviations as part of the plant-specific deviations from the generic guidelines in the plant-specific technical guidelines portion of the PGP. This analysis and associated documentation may take place after the issuance of a full-power license. If the function and task analysis is not completed before fuel loading, the operating license will be conditioned to require the applicant to comply with the requirements of Supplement 1 to NUREG-0737 as applied to the DCRDR and the upgrade of EOPs on a schedule approved by the NRC.

The staff has evaluated the impact of not completing the function and task analysis before plant operation. The Westinghouse Owners Group ERGs identify the existing control room instrumentation and controls to be used during implementation of the EOPs for an assumed reference plant design. The instrumentation and controls do not appear to be derived from operator information and control needs. Revision 1 to the Westinghouse Owners Group ERGs does, however, identify the basic information and control needs that the existing instrumentation and controls can meet. With this information, each licensee and applicant can evaluate the adequacy of installed instrumentation and controls to meet the operators' information and control needs, and determine needed instrument and control characteristics to the level of information provided in the generic guidelines. Because the reference plant for the Westinghouse Owners Group ERGs was a SNUPPS design, and because Wolf Creek is a SNUPPS-designed plant, the staff has reasonable assurance that the Wolf Creek EOP upgrade effort will result in adequate EOPs. The staff believes, however, that it is necessary to complete the task analysis efforts, as required by Supplement 1 to NUREG-0737, to further reduce the risk to public health and safety.

I.C.7 NSSS Vendor Review of Procedures

In accordance with Item I.C.7, NSSS vendor review of low-power testing, power ascension, and emergency operating procedures is necessary to further verify adequacy of the procedures. The requirement for vendor review of EOPs has been satisfied by the involvement of Westinghouse representatives in the development of emergency response guidelines, as reported under Item I.C.1 above. In addition, in a letter dated August 28, 1981, the applicant committed to a vendor review of procedures. NRC regional personnel will verify the review in accordance with IE Temporary Instruction 2514/01. On this basis, the staff considers Item I.C.7 resolved.

I.C.8 Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants

The SER stated that the staff does not plan to conduct a pilot monitoring review of selected emergency operating procedures according to Item I.C.8 because the applicant has committed to implement emergency operating procedures based on the revised Westinghouse Owners Group guidelines, in accordance with the schedule of NUREG-0737. The staff review of the applicant's program for developing emergency operating procedures is described in Item I.C.1 above. The staff considers TMI Task Action Plan Item I.C.8 resolved.

I.D.1 Control Room Design Review

TMI Action Plan (NUREG-0660) Item I.D.1 states that operating licensees and applicants for operating licenses will be required to perform a Detailed Control Room Design Review (DCRDR) to identify and correct design discrepancies. The objective is to improve the ability of control room operators in nuclear power plants to prevent or cope with accidents, if they occur, by improving the information provided to them. Supplement 1 to NUREG-0737 confirmed and clarified the DCRDR requirement in NUREG-0660. As a result of Supplement 1 to NUREG-0737, each applicant or licensee is required to conduct the DCRDR on a schedule negotiated with NRC.

NUREG-0700 describes four phases of the DCRDR to be performed by the applicant and licensee. The phases are

(1) planning
(2) review
(3) assessment and implementation
(4) reporting

NUREG-0801, Draft, "Evaluation Criteria for Detailed Control Room Design Review," provides the necessary criteria for evaluating each phase.

As a requirement of Supplement 1 to NUREG-0737, applicants and licensees are required to submit a program plan that describes how the following elements of the DCRDR will be accomplished:

(1) establishment of a qualified multidisciplinary review team

(2) function and task analyses to identify control room operator tasks and information and control requirements during emergency operations

(3) a comparison of display and control requirements with a control room inventory

(4) a control room survey to identify deviations from accepted human factors principles

(5) assessment of human engineering discrepancies (HEDs) to determine which HEDs are significant and should be corrected

(6) selection of design improvements

(7) verification that selected design improvements will provide the necessary correction

(8) verification that improvements will not introduce new HEDs

(9) coordination of control room improvements with changes from other programs such as safety parameter display system (SPDS), operator training, Regulatory Guide 1.97 instrumentation, and upgrade of emergency operating procedures

The NRC requires each applicant and licensee to submit a summary report at the end of the DCRDR. This report should describe the proposed control room changes and implementation schedules, and should provide justification for leaving safety-significant HEDs uncorrected or partially corrected.

The NRC will evaluate the organization, process, and results of each DCRDR. The evaluation of the applicant's and licensee's DCRDR efforts will consist of the following, as described in NUREG-0801:

(1) an evaluation of the program plan report submitted by the licensee/applicant

(2) a visit to some of the plant sites to audit the progress of the DCRDR programs

(3) an evaluation of the licensee/applicant DCRDR summary report

(4) a possible preimplementation audit

(5) the preparation of an SER that will present the results of the NRC evaluation

Significant HEDs should be corrected. Improvements that can be accomplished with an enhancement program should be done promptly.

SNUPPS submitted the DCRDR Summary Report for Wolf Creek Unit 1 on February 2, 1984. The staff conducted an onsite audit March 1, 1984, and transmitted an audit report to the applicant June 5, 1984. By letter dated May 11, 1984, the applicant provided revised responses to correct two HEDs identified in the Preliminary Design Assessment completed in 1982, and schedules for implementing actions to correct plant-specific HEDs as well as HEDs identified during a supplemental survey and auxiliary shutdown panel survey. The applicant developed responses to the staff audit findings and submitted a revision to the DCRDR Summary Report on June 29, 1984. Revision 1 to the DCRDR Summary Report included the results of an environmental survey of the control room conducted in April 1984. The staff has reviewed the above documentation of the Wolf Creek program and provides the following summary of the degree to which the requirements of Supplement 1 to NUREG-0737 were satisfied.

Planning Phase

After reviewing the SNUPPS DCRDR Program Plan submitted in June 1983, the staff concluded that it was incomplete and did not address some of the elements in sufficient detail to establish how the element would be accomplished. After a meeting with the staff on October 25, 1983, SNUPPS submitted a revised plan on November 28, 1983. In addition, the DCRDR Summary Report contained samples of the forms used in documenting the methodologies and activities of the DCRDR.

The concerns expressed after the review of the original DCRDR Program Plan were as follows:

(1) qualifications of the human factors contractor and other engineering and training personnel

(2) involvement of the human factors consultant in the DCRDR

(3) level of involvement of each of the disciplines participating in the DCRDR for each DCRDR task

(4) organization of management for the DCRDR

With the exception of the level of involvement of an experienced human factors engineer in the System Function Review and Task Analysis (SFR&TA), subsequent discussions with SNUPPS and utility personnel and supplemental documentation satisfied these concerns. The SNUPPS DCRDR management structure and the qualifications and involvement of personnel were adequate to conduct a satisfactory DCRDR.

Review Phase

The activities included in SNUPP's review phase are:

(1) operating experience review
(2) system function review and task analysis
(3) control room inventory
(4) control room survey
(5) verification of task performance capabilities
(6) validation of control room functions

Activities 2 through 5 address specific DCRDR requirements contained in NUREG-0737, Supplement 1.

(1) Operating Experience Review

SNUPPS recognizes the value of operating experience input in the DCRDR and although this is not a requirement of NUREG-0737, Supplement 1, The SNUPPS group appears to have performed a review of operating experience which will provide valuable insights and feedback for other DCRDR activities.

(2) System Function Review and Task Analysis (SFR&TA)

Besides the limited use of human factors engineering expertise in the task analysis effort, the staff has several concerns about the approach taken by SNUPPS to define the required design characteristics of controls and displays. The points below summarize these concerns:

(a) No analysis was conducted to define the required characteristics of "digital" (discrete) controls or displays.

(b) Because plant-specific documentation was used to identify some of the design requirements against which plant-specific instrumentation was compared, the verification of instrument suitability may not have been valid.

(c) On the basis of the SFR&TA writeup, examples of continuous monitoring and modulating control tasks, and the sample Task Analysis and Verification Worksheet, it is unclear what analysis, if any, was conducted to determine the information and control characteristics required by operators to accomplish their tasks.

(d) There appears to be inconsistency in the requirements specified for certain parameters in Appendices B and F of the Summary Report (F and J of Revision 1 to the Summary Report).

(3) Control Room Inventory

The inventory of controls and displays in the control room that is used in the DCRDR consists of plant design drawings and specifications. In itself, the inventory of controls and displays appears to be comprehensive and should have provided adequate support to the DCRDR as an information source.

(4) Control Room Survey

The control room survey work was initiated as the Preliminary Design Assessment (PDA) in 1980 using NUREG/CR-1580 as the source of evaluation criteria. After the issuance of NUREG-0700, SNUPPS performed a supplementary survey and a survey of the auxiliary shutdown panel. An environmental survey was performed in April 1984. The results of these surveys are summarized below.

(a) Preliminary Design Assessment (PDA)

The open items from the NRC audit of the PDA which was performed in July 1981 have been determined to be adequately resolved and control room improvements implemented.

(b) Supplementary Survey (SS)

Appendix D of the DCRDR Summary Report (Appendix B of Revision 1 to the Summary Report) listed the human engineering discrepancies (HEDs) and SNUPPS responses resulting from the SS. The audit team in the control room examined the HEDs from each of the nine sections of the SS. The resolutions to all findings in the SS were determined to be acceptable.

(c) Auxiliary Shutdown Panel (ASP) Review

Appendix E of the DCRDR Summary Report (Appendix C of Revision 1 to the Summary Report) lists the HED and SNUPPS responses resulting from the ASP review.

The audit team of the auxiliary shutdown panel examined the HEDs from the nine sections of the ASP review. All findings in the ASP review were resolved by two submittals from the applicant dated March 21, 1984, and June 29, 1984, and were determined acceptable as was the schedule for implementation of improvements.

(d) Environmental Survey

Results of the environmental survey indicate that air velocity in the control room is higher than that recommended by human factors guidelines, and the ambient noise level is at the maximum for unimpeded communications. The staff does not expect that these items individually will have a detrimental effect on the safe operation of the Wolf Creek plant. However, the combination of maximums results in less than a desirable environment which could add unnecessarily to the overall stress level of the operator during emergency operations. In addition, if these undesired conditions exist when the systems and equipment are new, they cannot be expected to improve, but more likely will degrade with age. In general, the control room survey work performed during the PDA, SS, and ASP reviews and the environmental survey activities were comprehensive and have met the requirement of Supplement 1 to NUREG-0737 for "a control room survey to identify deviations from accepted human factors principles." In the context of this task, the staff believes the applicant is adequately resolving the HEDs identified and has improved the operability of the Wolf Creek control room.

(5) Verification of Task Performance Capabilities

Under the duplicate plant concept, the Callaway simulator is certainly an acceptable tool for verifying task performance capabilities. However, the staff is concerned that such verification (by performing tasks on the simulator) may lack objectivity through the natural tendency to uncritically accept, as suitable, that which already exists in the control room. Unless a set of predefined design requirements exist (from the task analysis), describing the characteristics of the controls and displays needed by the task, the only verification to be accomplished is that the controls and displays exist in the control room. Little can be said objectively about their suitability for performing the task. The acceptability of this task will be resolved as a part of the System Function Review and Task Analysis.

(6) Validation of Control Room Functions

SNUPPS performed two separate validations of control room functions. The first effort consisted of analyzing the video-taped walkthroughs of various procedures performed at the SNUPPS simulator at Zion. The findings from this analysis were incorporated as part of the PDA findings.

The second effort consisted of analyzing the video-taped walkthroughs of the entire set of 41 Westinghouse Owners Group Emergency Response Guidelines (WOG ERGs) at the Callaway simulator. This validation effort appears to have been focused primarily on validating the WOG ERGs. In addition, SNUPPS took the opportunity of analyzing the video tapes to evaluate control room instrument and control consistency with the procedures, operator workload, and workstation flow or traffic. The six HEDs produced from this second validation effort reflect an adequate evaluation.

Assessment and Implementation Phase

(1) Assessment of HEDs

Although a prioritization process was carried out for the large majority of HEDs identified in the PDA and DCRDR, prioritization did not serve very often as a criterion for HED resolution or selection of design improvement. The SNUPPS approach was to correct as many HEDs as possible regardless of the assigned priority.

The staff finds that the requirement of Supplement 1 to NUREG-0737 regarding assessment of HEDs has been met.

(2) Selection of Design Improvements

The staff has reviewed all design improvements, both implemented and proposed, including the two revised responses in the applicant's letter dated May 11, 1984, and believes that the applicant has met this NUREG-0737, Supplement 1, requirement.

(3) Schedules for Implementing HED Corrections

SNUPPS and utility personnel are responsive in accomplishing the improvements needed in the control room in an expeditious manner. Most improvements have already been completed. Only a few HEDs requiring long lead-time parts or more detailed design effort will be accomplished before startup from the first refueling outage.

(4) Verification That Improvements Will Provide the Necessary Corrections Without Introducing New HEDs

The procedure for this review includes (a) an evaluation of the redesign against the HED and recommended resolution, if provided, (b) a depiction and evaluation of significant changes on a full-scale mockup or control board drawing, and (c) performance of walkthroughs of selected procedures on either the full-scale mockup, the simulator, or the control room, after changes have been implemented. The revised design is scrutinized from a human engineering viewpoint by the DCRDR team and any feedback from the procedure walkthroughs is conveyed to the DCRDR team from utility operations and engineering personnel. The staff finds that this verification process satisfies the requirement of Supplement 1 to NUREG-0737.

(5) Coordination of the DCRDR With Other Improvement Programs

SNUPPS appears to be integrating the DCRDR with operator training, RG 1.97 instrumentation, development of emergency operating procedures (EOPs), and SPDS development in a manner that satisfies the requirement of Supplement 1 to NUREG-0737.

Conclusion

The staff concludes that the applicant, through the Standard Nuclear Unit Power Plant System, has conducted a DCRDR for Wolf Creek Generating Station Unit 1 that substantially meets the requirements of Supplement 1 to NUREG-0737 except in the area of System Function Review and Task Analysis and the subsequent activities resulting from that analysis. The staff requires additional information to determine the acceptability of the final control room design and the implementation of control room improvements. The applicant must conduct the task analysis to develop and document the following information:

(1) A description of how the design requirements were determined for the plant-specific documentation that was used to identify the design characteristics against which plant-specific instrumentation was compared.

(2) For each instrument and control used to implement the EOPs, an auditable record of how the needed instrument and control characteristics were determined. These characteristics should be derived through the task analysis

process from the information and control needs identified in the background documentation of the ERG and from plant-specific information. Once these information and control characteristics have been developed, a review of the control room must be accomplished to verify the existence and suitability of the displays and controls to satisfy the information and control requirements. Should any discrepancies result, these must be analyzed to determine their safety significance, requirements for corrective action, and an implementation schedule. In a meeting with the applicant on July 13, 1984, an agreement was reached to accomplish the above effort by April 30, 1985. Completion and documentation of the task analysis should be made a condition of the operating license.

I.D.2 Plant Safety Parameter Display System

By letter dated January 13, 1984, a safety analysis on the safety parameter display system (SPDS) for SNUPPS plants was submitted for staff review. The SNUPPS plants covered by the analysis are Callaway Unit 1 and Wolf Creek Unit 1. The staff conducted a design verification audit of the Wolf Creek SPDS during August 20-22, 1984. The supplement contains the staff's findings and conclusion from the audit and the review of the safety analysis.

The results of this review apply to the SPDS design of Wolf Creek Unit 1. On the basis of its review of the safety analysis (SNUPPS letter, Jan. 13, 1984) and the results from its audit of the Wolf Creek SPDS, the staff concludes that it is acceptable for these SNUPPS plants to continue implementing their SPDS program. However, the staff has conditioned its acceptance of continued SPDS implementation with the recommendation that data on containment isolation status be added to the SPDS. In addition, the staff requests SNUPPS to assess, evaluate, and report on the safety significance of the human engineering discrepancies and concerns defined during the staff audit. Further, the staff has additional recommendations on the plant-specific implementation of the generic SPDS design used by the plants. These recommendations, along with staff results, are presented next.

(1) Display Description

The SNUPPS SPDS conceptual design and prototype was the Safety Assessment System (SAS). The SAS was based on a design developed jointly by a group of Westinghouse nuclear steam supply system (NSSS) utilities of which the SNUPPS utilities were members. SAS provides a centralized, flexible, computer-based data and display system to assist control room personnel in evaluating the safety status of the plant. The highest level graphical display format contains a minimum set of key plant variables representative of the plants' safety status, and these constitute the SPDS. In addition, there are critical safety function display formats, accident identification display formats, and process variable trend display formats within the library of the display system.

The NRC staff has been previously briefed on SAS and also has witnessed a demonstration of a SAS prototype. The SAS prototype had many positive features, such as a hierarchial display concept, the use of the shape and color coding of information, and uncluttered display formats. Also, a program for operator evaluation of the prototype display system was described to the staff. However, the staff has not yet conducted a formal review of the SAS display system.

In performing an audit of the Wolf Creek SPDS, the staff recognized that it would be necessary to evaluate the generic design (SAS) as well as the plant-specific design and implementation of the SPDS. Staff review results and comments are presented in terms of the generic design and in terms of the SNUPPS-specific design of the SPDS.

In terms of the scope and depth of review, the staff concentrated its effort upon the MODE display formats. These display formats consisted of three formats designated as: (1) NORMAL, (2) HEAT UP, COOLDOWN, and (3) COLD SHUTDOWN. The staff did not evaluate the accident identification displays (AIDs), the safety system readiness monitor (SSRM), and the safety system performance monitor (SSPM), which are also part of SAS.

(2) Variable Selection

Section 4.1(f) of Supplement 1 to NUREG-0737 states that:

> The minimum information to be provided shall be sufficient to provide information to plant operators about:
>
> (i) reactivity control
> (ii) reactor core cooling and heat removal from the primary system
> (iii) reactor coolant system integrity
> (iv) radioactivity control
> (v) containment conditions

For review purposes, these five items have been designated as critical safety functions.

In the evaluation of the SPDS variables and in its recommendations, the staff considered the Westinghouse Owners Group, "Westinghouse Emergency Response Guidelines (ERGs) Program," which was reviewed and approved by the staff (Generic Letter 83-22, June 8, 1983), as a principal technical source of variables important to the safe operation of the plant. The SPDS variables selected by the applicant and their coordination with the critical safety functions are summarized in the SNUPPS submittal, dated January 13, 1984 (Table 1, grouping made by the applicant). Variable selection, SPDS design, and SPDS operation were demonstrated in an audit review by the staff. Certain key variables of interest (T_{HOT}, sump level, steam generator radiation, containment pressure, and containment hydrogen concentration) while not directly represented on the central top-level display format, are available to the operator on the SPDS console in a callup of other display formats.

Although the variables selected for display do comprise a generally comprehensive list, the status of containment isolation is not proposed for the Wolf Creek SPDS. Containment isolation is an important variable for use in making a rapid assessment of containment conditions. In particular, a determination that known process pathways through containment have been secured provides critical additional assurance that containment integrity is intact. The applicant has not identified "containment isolation" as an SPDS variable; however, isolation status indication is available in the control room and is readily observable. This assertion may be made for most of the proposed SPDS variables, but does not constitute adequate justification for omission. Therefore, adequate justification for the absence of data on containment isolation status from the SPDS has not been provided.

For a rapid assessment of radioactivity control, the applicant has not demonstrated how radiation in the secondary system (steam generators and steamlines) is monitored by the SPDS when the steam generators and/or their steamlines are isolated. The applicant should consider this capability in the verification and validation program.

The above discussed variables do, for given scenarios, provide unique inputs to determination of status for their respective critical safety functions, and they have not been discussed by the applicant as being satisfied by other variables in the proposed SNUPPS SPDS list. The staff recommends that the applicant address these variables and their functions by: (1) adding these variables to the SNUPPS SPDS, (2) providing alternate added variables along with justifications that these alternates accomplish the same safety functions for all scenarios, or (3) providing justification that the variables currently on the SNUPPS SPDS provide the minimum information to evaluate the safety functions for all scenarios.

On the basis of this review of the applicant's supporting analyses and staff observations that the selected variables appear to be consistent with the Westinghouse Owners Group ERGs, the staff finds the proposed list of key variables acceptable, with the exceptions noted above.

During the audit review, the applicant described the method used to derive the SPDS variable set and the overall program to validate the variable set. This program includes consideration of related Indian Point data, and industry studies (Science Applications, Inc., Electric Power Research Institute). As part of the validation program, the applicant will exercise the SPDS for a number of cases (approximately eight, including large LOCA, loss of main feedwater, and core power excursion) using the SNUPPS Wolf Creek plant-specific simulator and utilizing Westinghouse ERGs which will be implemented at the plant. Results of these exercises will be considered for possible changes to the variables selected for the SPDS. Although the final list of events to be exercised is not complete, the staff finds that this program provides a proper framework for validation of the variable set, but recommends that the set of events to be tested includes steam generator tube rupture with loss of offsite power (to verify radiation, cooldown, and inventory monitoring), large steamline break (to verify vessel integrity and reactivity control monitoring), and one or more severe accident cases (to verify compatibility with emergency operating procedures for beyond-design-basis scenarios).

(3) <u>Display Data Validation</u>

The staff reviewed the safety analysis (SNUPPS letter, Jan. 13, 1984) to determine that means are provided in the display's design to ensure that data displayed are valid. The analysis states that all data displayed by the SAS is validated by comparing redundant sensors, checking the value against reasonable limits, calculating rates of changes, and/or checking temperature versus pressure curves. Invalid or suspect data are identified by consistent use of color and pattern codes of text and symbols.

During the audit of the Wolf Creek SPDS, the staff learned that data validation was limited to range checks and to the comparison of redundant sensor signals. Also, the number of sensor signals to measure a variable used is considered in the validation process.

The data for valid/invalid variables are coded for display and operator use. The staff also witnessed a demonstration which illustrated the ease by which valid/invalid data could be identified on the screen of the display system.

In the review of other SAS-based SPDSs, the staff learned that the SPDS variables presented to the operator are validated by an algorithm developed by the generic SAS project. This algorithm implements a rejection criterion developed by Chauvenet and the algorithm was coded by Quadrex for use in the SAS demonstration software. The essence of the algorithm is to eliminate individual values of a variable (measured by multiple sensors), which exceed a "probable" deviation from a calculated mean value of the variable, in a recursive manner until at least two values remain. If the remaining two values are spread excessively, the final average is considered questionable and so indicated to the operator by coding the displayed data.

The staff has reviewed the data rejection algorithm and finds it acceptable as a means for rejecting outlying data when multiple sensor signals exist for a process variable. Also, on the basis of data presented in the safety analysis and the results from the staff audit, the staff confirms that means are provided in the SNUPPS design to ensure that the data displayed are valid.

(4) Human Factors Program

The staff evaluated the safety analysis for a comittment to a human factors program to ensure that the displayed information can be readily perceived and comprehended so as not to mislead the operator. The safety analysis states that all display formats for SAS have been carefully designed by persons with plant operating experience and then evaluated against human factors design criteria. Design and evaluation efforts focused on the fundamental objective of aiding the control room operators in rapidly and reliably assessing the safety status of the plant. No control room displays were superseded by SAS and the control room design reviews verified suitability of the control room designs without SAS for all normal and accident conditions. Furthermore, the analysis states that human factors engineering and industrial-design techniques have been effectively combined to establish man-machine interface design requirements, maximize system effectiveness, reduce training and skill demands, and minimize operator error.

The safety analysis also states that minimum use of color combined with simplified format for all of the cathode-ray tube (CRT) displays are key design features to aid both normal and off-normal pattern recognition. The operator, who is the end-user, has been directly involved from the conception of the design to ensure that man-machine interface goals of SAS have been satisfied. Human factors engineering standards and testing verification have been used which are consistent with accepted practices.

During the audit of the Wolf Creek SPDS, the staff evaluated the verification and validation (V&V) program and the design process used in the development of the prototype display and used for the SNUPPS-specific implementation of the prototype. The staff found the structure and scope of the generic V&V program as well as the SNUPPS-specific V&V program to be comprehensive, and if properly implemented, design errors in the display system should be minimized. The results from the generic validation program were not available for staff review at the audit site.

In evaluating the design process and standards used, the staff audited several generic as well as SNUPPS-specific design specifications. The staff noted that human factors engineering was included in the design, e.g., shape and color coding of information to indicate off-normal values was a documented requirement, as were the color codes to be used in the display. Also, the staff found the SNUPPS-specific design requirements defined specific modifications to the prototype display, such as labels and abbreviations, to make the display system plant specific. Staff audit of the design specifications concluded that the requirements appeared to be adequate for the display system.

Upon completing the audit of these documents, the staff asked if a human factors review of the generic display formats had been conducted. The applicant's response indicated that informal reviews had been conducted but no documented evidence existed. The staff requested that a summary of these activities, which defines personnel, methods, and results, be submitted.

This summary of human factors engineering's involvement in the review of display formats was provided by SNUPPS (letter, Sept. 5, 1984). The staff reviewed the summary and also reviewed the qualifications of the human factors professionals (provided with the summary). The summary describes recommendations and experiments made by human factors professionals to establish the best uses of color coding and shape coding of information, and to establish trend graph line weights and time axis labels. Also, human factors professionals played a major role in the operator evaluation program conducted on the Indian Point simulator. They guided the evaluation design, served as observers as the transients were run, debriefed the operators, and compiled the feedback data.

During the audit, the staff evaluated several SPDS display formats on a prototype of the Wolf Creek SAS. The staff found the display formats clutter free and easy to read, and information was reasonably grouped into logical sets. Access to the individual display formats is achieved by means of dedicated, labeled keypads. The individual keys within each keypad were also labeled and easily read and used to call up a display format.

In evaluating the position of the labels within the mode display formats, the staff noted that the labels for a set of data appeared at the bottom of the set rather than at the top. In looking at the CSF TREE display format, the staff noted that labels were at the top. The staff's concern is that the position of labels be consistent in a control room to minimize operator errors in the search and acquisition of data during emergencies.

NUREG-0737, Supplement 1, requires the coordination of the initiatives to achieve an integrated emergency response capability within a nuclear power plant. One of the initiatives is the detailed control room design review (DCRDR). The objective of the control room design review is to improve the ability of nuclear power plant control room operators to prevent accidents or to cope with accidents if they occur by improving the information provided to them. This objective is not met when the method used to label data sets in the SPDS is inconsistent with the method used to label data and devices on the control board.

In terms of a control room review, all examples of inconsistent application of label standards would be defined as human engineering discrepancies (HEDs). The staff requests that SNUPPS consider the inconsistent labeling methods as HEDs. Further, the staff requests that SNUPPS assess the safety significance of this HED and report the results to the staff.

The staff also noted that the coordination of related information was integrated into the display system. In the message area of the NORMAL OPERATION MODE display format, information on the status of the critical safety functions is presented. This status information is presented in the same color as other information in the message area. Because of the safety significance of this status information, the staff recommends that a highlighting technique be considered to enhance operator detection and response to off-normal critical safety functions. The staff requests that SNUPPS evaluate this recommendation and report on the results of the evaluation to the staff.

In evaluating the display formats used to present trend data of variables, the staff noted that the time scale read from right to left, with current time on the right. However, the scales and current magnitude of the variables were presented on the left side of the display format. Thus, an operator's monitoring task, such as evaluating margin to an operating limit and evaluating the current trend of the variable toward the limit, becomes difficult to perform as the data are located at opposite ends of the display screen.

On the basis of the review of the safety analysis and the data acquired during its audit, the staff confirms that SNUPPS did commit to a human factors program in the design of the SPDS. However, the staff recommends that the plant-specific implementation of the SPDS in the control room be coordinated and made consistent with the human factors engineering standards and guidelines employed in the control room in order to minimize operator error in the use of data provide by the SPDS.

(5) Electrical and Electronic Isolation

In the SNUPPS design, data are transmitted to the SPDS via a data link from the emergency response facility information system (ERFIS). The ERFIS receives its information on another data link from the balance-of-plant (BOP) computer and the nuclear steam supply system (NSSS) computer. Both of these computers have Class 1E and non-Class 1E sensor inputs. The Class 1E inputs are isolated from the NSSS computer by qualified isolation devices that were reviewed and accepted by the staff in the following documents: (1) WCAP-8892-A, "Westinghouse 7300 Series Process Control System Noise Tests," June 1977; (2) WCAP-7506-L, "Test Report Nuclear Instrumentation System Isolation Amplifier, Rev. 1," February 1970; and (3) WCAP-10621, "Westinghouse Thermocouple/Core Cooling Monitor System Test," July 1984.

The Class 1E sensor inputs to the BOP computer are isolated by qualified isolation devices that were reviewed and accepted by the staff in the following documents: (1) Meeting with staff, December 1980, documented in SNUPPS letters dated January 20, 1981, and February 27, 1981; (2) meeting with staff, July 17, 1981, NRC letter to SNUPPS dated August 13, 1981; (3) meetings with staff on May 18 and June 16, 1981, documented in SNUPPS letter dated August 14, 1981; and (4) Safety Evaluation Report, Section 7.3.2.6, "Isolation Devices in the Balance of Plant Engineered Safeguards Features Actuation System," October 1981.

On the basis of its audit of the documentation on isolation devices (optical and transformer) used within the SNUPPS design, the staff concluded that these devices are qualified isolators and are acceptable for interfacing the SPDS (via the NSSS and BOP computer) with safety systems.

Conclusion

The staff's review of SNUPPS SPDS safety analysis concludes:

(1) The variables selected for the Wolf Creek SPDS are acceptable; however, the staff recommends the addition of the data on containment isolation status to the SPDS, or additional justification that it may be excluded. Also, variable validation should include additional events to demonstrate usability of the SPDS,

(2) That means are provided in the display's design to ensure that the data displayed are valid,

(3) Human factors engineering principles are being considered in the display's design to ensure that the displayed information can be readily perceived and comprehended so as not to mislead the operator. However, the staff recommends that the plant-specific implementation of the SPDS in the control room be coordinated and made consistent with the human factors engineering standards and guidelines employed in the control room in order to minimize operator error in the use of data provided by the SPDS. In this regard, the staff requests SNUPPS to assess, evaluate, and report on the safety significance of the human engineering discrepancies and concerns defined during NRC audit.

(4) The isolation devices used in the design are acceptable for interfacing the SPDS with safety systems.

The continued implementation of the SPDS by SNUPPS is conditional to a satisfactory confirmatory review by the staff on the above design data requested from the licensee.

II.B.3 Postaccident Sampling System

On the basis of its evaluation of the postaccident sampling system (PASS), the staff concluded in Supplement 4 of the Wolf Creek SER that 9 of the 11 criteria were acceptable. The following criteria remained unresolved:

Criterion 6 Provide a core damage estimate procedure to include radionuclide concentrations and other physical parameters as indicators of core damage.

Criterion 9 Provide information demonstrating applicability of procedures and instrumentation in the postaccident water chemistry and radiation environment, and retaining of operators on semiannual basis.

By letter dated August 31, 1984, the applicant provided a procedure for estimating the degree of reactor core damage based on the Westinghouse Owners Group generic methodology, "Post-Accident Core Damage Assessment Methodology," Revision 1, dated March 1984.

The procedure takes into consideration other physical parameters such as reactor core temperature data, reactor water level, sample location, and containment radiation levels and hydrogen concentrations. The staff has determined that these provisions meet Criterion 6 and are, therefore, acceptable.

The accuracy, range, and sensitivity of the PASS instruments and analytical procedures are consistent with the recommendations of RG 1.97, Revision 3, and the clarifications of NUREG-0737, Item II.B.3, transmitted to the applicant on June 30, 1982. Therefore, they are adequate for describing the radiological and chemical status of the reactor coolant. The analytical methods and instrumentation were selected for their ability to operate in the postaccident sampling environment. The standard test matrix and radiation effect evaluation indicated no interference in the PASS analyses. The equipment and procedures used for the PASS will be tested or calibrated to maintain a high level of reliability. Training of operators will be conducted in accordance with the plant qualification program in conjunction with participation in semiannual emergency planning drills. The staff has determined that these provisions meet Criterion 9 of Item II.B.3 in NUREG-0737, and are, therefore, acceptable.

Conclusion

On the basis of its evaluation, the staff concludes that the postaccident sampling system now meets all 11 criteria of Item II.B.3 in NUREG-0737 and is, therefore, acceptable.

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

As required by NUREG-0737, Item II.D.1, all PWR licensees and applicants are required to demonstrate that their pressurizer safety valves, 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 NRC. Additionally, each PWR plant applicant for an operating license (OL) was required to submit a report, by fuel loading, that would demonstrate the operability of these valves and the associated piping.

On July 1, 1982, SNUPPS responded to this requirement with a submittal that contains information from the EPRI valve test program results that apply to Wolf Creek. The applicant has also responded by a January 7, 1983, submittal that states that the safety and relief valve discharge piping and supports have been verified to ensure that they are functional and have no adverse affect on valve operability.

The staff has not completed a detailed review of the applicant's submittals; however, on the basis of a preliminary review, the staff finds 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 uses safety valves, PORVs, and PORV block valves of the same size and model that performed satisfactorily for test sequences that are considered representative of or that bound conditions to which the SNUPPS valves could be exposed.

In summary, on the basis of its preliminary review, the staff has concluded that the applicant's general approach to responding to this 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 the staff completes its detailed review. If the completion of the detailed review reveals that modifications or adjustments to safety valves, PORVs, PORV block valves, or associated piping are needed to ensure that the overpressure protection systems can perform their intended functions, the staff will require that the applicant make appropriate modifications.

II.E.4.2 Containment Isolation Dependability

Requirement

Demonstration of operability of the containment purge and vent valves, particularly the ability of these valves to close during a design-basis accident (DBA), is necessary to ensure containment isolation. This demonstration of operability is required by BTP CSB 6-4 and SRP 3.10 for containment purge and vent valves which are not sealed closed during operating conditions 1, 2, 3, and 4.

Description of Purge and Vent Valves

The 36-in. large-volume, shutdown, purge valves are closed during operating modes 1, 2, 3, and 4 as required by the plant Technical Specifications, and thus are not a subject of this review. The 18-in. mini-purge valves listed below, which are subject to review, are manufactured by the Fisher Controls Company, Type 9200, and are equipped with G. H. Bettis air open-spring return actuators, Model No. T416B-SR3-12.

Valve	Size, in.	Use	Location
GT-HZ-04	18	Supply	Outside containment
GT-HZ-05	18	Supply	Inside containment
GT-HZ-11	18	Exhaust	Inside containment
GT-HZ-12	18	Exhaust	Outside containment

Demonstration

Operability demonstration information was provided for the Wolf Creek Generating Station in a SNUPPS letter dated January 16, 1984.

The containment conditions utilized in predicting the ΔP across the valves were extracted from the analysis for the large loss-of-coolant accident (LOCA) which results in the peak containment pressure.

The containment pressure rises from approximately 14 psig to 22.9 psig during the closure stroke of the valve. However, the pressure drop at incremental valve positions was calculated at the higher pressure of 22.9 psig.

The containment pressure during the closure cycle was assumed to be a constant 22.9 psig which corresponds to the pressure at the time that the valve is fully closed (T = 6 seconds). The lag time between the receipt of the signal to close at 3 seconds from LOCA start and the initiation of valve motion has been taken into account. Tests have verified that closure is accomplished with the required 3 seconds from receipt of the closure signal.

The valves are equipped with spring-return actuators. The unpressurized side of the piston actuator is vented to local ambient conditions. During valve closure, the pressure side is also vented to the same local ambient conditions. Therefore, no pressure differential will exist across the piston as a result of a surrounding local pressure rise. The spring will drive the actuators to the fail-safe (closed) position and will maintain that position.

Table 22.1 compares torque available to torque required as presented in the submittal.

Case 1 (Table 22.1) provides the calculated pressure drops across the valve for each 10° of valve position. These pressure drops were calculated on the basis of the installed piping system resistances (pressure drops), assuming that the redundant purge valve in the same line has failed in the open (least resistance) position. The predicted pressure drops are all calculated at 22.9 psig, which is the maximum pressure that would exist before valve closure. Data also presented for Case 1 include the actuator torque available and the ratio of torque available to torque required at the corresponding positions. The ratio of excess torque varies from 7.0 to 14.5 for all opening positions.

Case 2 (see Table 22.1) provides data for the worst possible case wherein the valve closure is assumed to be delayed until the peak calculated containment pressure of 47.2 psig is attained. This is not a design case, but is provided to demonstrate the large margins available to ensure valve closure. As can be seen from the data provided, excess torque margins of 4.0 to 7.9 exist even for this worst case. As with the design case, all pressure drops are calculated with 47.2 psig at the inlet to the purge line and the redundant valve in the line failed in the open position.

The shaft is considered to be the most critical valve component under most conditions, since the pins and keys are selected to be stronger than the shaft. Therefore, separate calculations for pins and keys are not necessary. Stress concentration factors are considered when evaluating the shafts at the pins and keyways. The maximum disc load occurs when the disc is in the closed condition and acceptable disc strength values have been established based on testing and experience.

The stresses reported in Table 22.2 are based on dynamic loadings that result from the LOCA pressurization transient. Shaft loadings that result from the seismic event are not specifically calculated or combined with LOCA loadings because the events are independent and not postulated to occur simultaneously. The purge valves are seismic Category I and have been tested for operability during and after a seismic event.

Evaluation

In demonstrating operability, the accident condition considered was a large LOCA that results in a peak-containment pressure of 47.2 psig. A containment pressure response curve is provided that also indicates the time the valves receive a signal to close and the time the valves would be fully closed, a total of 6 seconds. (The 6 seconds accounts for a 3-second lag time and a 3-second closure time. The applicant reports that tests have ensured that closure is accomplished within the required 3 seconds from receipt of the closure signal.) This corresponds to a pressure of 22.9 psig. Thus, the applicant conservatively has considered at each incremental valve position, a maximum ΔP of 22.9 psig across the valve (Case 1, as discussed above).

Additionally, the applicant has provided analysis results for the peak containment pressure of 47 psig (Case 2). This was provided to demonstrate the large margins available to ensure valve closure.

Throughout the submittal the applicant has not combined LOCA and seismic loadings either in the stress analysis (discussed below) or when developing resulting torques (discussed below). The NRC has historically required that the structural/mechanical responses to various accident loads and loads caused by natural phenomena (such as earthquakes) be combined when analyzing structures, systems, and components important to safety. This requirement flows from GDC 2 of 10 CFR 50, Appendix A, which calls for an appropriate combination of the effects of the above events to be reflected in the design bases of safety equipment.

The submittal stated the following regarding dynamic torque:

> Dynamic torque factors used in butterfly valve sizing were developed from test data obtained from models with similar disc configurations and flow characteristics. The dimensionless aspect ratio (defined as the ratio of the disc diameter to the thickness) was judged to be a significant parameter for evaluation of dynamic torques at various opening angles. Therefore, a series of water flow tests was conducted with a group of 4-inch and 6-inch butterfly valve models constructed with various aspect ratios, ranging from 3:1 to 14:1 (such as 3:1, 8:1, 11:1, and 14:1), in various disc configurations (conventional, offset, cammed), and in both flow directions.
>
> The tests were conducted using the Fluid Controls Institute (FCI) specifications for test arrangement and conduct, per FCI paper 58-2.
>
> The basis followed by the vendor (Fisher) in using <u>incompressible</u> (water) flow model tests to establish dynamic torque coefficients applicable to large diameter valves in <u>compressible</u> flow service is presented in the ISA [Instrument Society of America] Transactions article "Effects of Fluid Compressibility on Torque in Butterfly Valves," by Floyd P. Harthun.
>
> Application of the conclusions from the ISA paper was presented in Attachment 2, selection Figures 1 and 2 of the submittals: "Torque-Pressure Drop Relationships Used in Dynamic Torque Coefficients Selection," which are representations of the Figure 5

curves from the ISA paper. These show how the torque values for compressible flow are conservatively determined and related to incompressible flow torques. It should be noted that the compressible flow curve reaches a critical flow condition at larger ΔP values, resulting in a maximum torque value (T_c) that cannot be exceeded, regardless of how large ΔP becomes.

The available actuator torques and the required torques at the valve for each 10° of opening are shown in Table 22.1. The model valve bench test program used to develop dynamic torque coefficient are configured with straight pipe inlets that produce uniform approach flow to the test valves. Testing did not include inlet piping configurations involving tees and elbows upstream of the valves. Valve installation details provided indicate upstream elbows and tees that should be accounted for in developing dynamic torques.

On the basis of information derived from other valve manufacturers' model tests, the staff accepts a factor of 3.0 times the T_D predicted stemming from straight pipe or uniform approach flow developed dynamic torque coefficients for an "elbow-shaft out-of-plane" configuration and a factor of 1.5 for the "elbow shaft in-plane" configuration. Therefore, applying a factor of 3, assuming worst case configuration, to the torque required in Table 22.1, the ratio of available to required torque is 2.3 to 4.78.

The results of the analysis performed by the applicant when coupled with the factor of 3, to account for the worst case loading of "elbow-shaft out-of-plane," demonstrated that the valves will close from the 90° position (fully open) from dynamic loads only. Acknowledging the fact that significant margin exists, the applicant should verify that under combined seismic and dynamic loads the valves will close (see paragraph 3 under "Evaluation," above).

The stress analysis was based on three conditions: closed, Case 1, and Case 2 (see Table 22.1 above for clarification of Case 1 and 2). The results of this analysis are summarized above. Consistent with other licensee/applicant submittals, the maximum stresses have occurred at the 70° position with the most critical valve component being the shaft/disc/pin interface. Significant margin exists, as reported by the applicant (a factor of 7 times minimum), between the stress allowable and the calculated stresses for the design Class 1. The stresses reported are based on dynamic loadings from the LOCA pressurization transient. Shaft loadings which result from a seismic event were not presented or combined with LOCA loadings because the events are considered by the applicant to be independent and not postulated to occur simultaneously (see paragraph 3 under "Evaluation," above). The purge valves are seismic Category I.

The submittal does not address the structural integrity of the actuator and and actuator/valve interface. It is concluded, however, that since the actuator is capable of delivering forces that exceed the dynamic torques, the structural integrity of the actuator under fluid dynamic conditions is not of concern. Therefore, the applicant should review the valve, actuator, and valve/actuator interface under combined dynamic and seismic loads and confirm their qualification (see paragraph 3 under "Evaluation," above).

Summary

The staff accepts the applicant's position that had the loads been combined, the calculated stresses would not have exceeded the allowable stresses. This is an instance when the applicant did not follow guidance but still satisfied the requirement.

The staff completed its review of the information concerning operability of the containment purge and vent valves at Wolf Creek. The staff finds the information submitted demonstrated the ability of the Fisher Controls Co. 18-in. valves to close against the buildup of containment pressure in the event of a DBA/LOCA.

II.F.2 Instrumentation for Detection of Inadequate Core Cooling

During the OL review of the SNUPPS design, the staff noted that a Class 1E microprocessor-based plasma display system will be used (as a backup to the primary display) to provide information of inadequate core cooling (ICC) to the operator. Because the Class 1E microprocessor-based ICC indication system is a new design concept, the staff decided to perform an audit review of the system.

NUREG-0737 establishes the criteria for design and qualification of PWR thermocouples and indication systems for ICC monitoring. It requires that plants use a Class 1E backup display system that is energized from Class 1E power and is independent of the primary display. Additional design guidance is provided by Appendix B to NUREG-0737, and the following IEEE standards: 323-1974, 308-1978, 384-1977, 344-1975, 338-1977, 379-1977, 497-1977, and 742-1982.

The Wolf Creek backup display system consists of two identical microprocessor-based thermocouple core cooling monitor (TC/CCM) system assemblies manufactured by Westinghouse. The TC/CCM system is a microprocessor-driven plasma display unit that presents core cooling information derived from incore thermocouple readings, primary system pressure, and hot and cold leg reactor coolant temperatures. Each assembly is to be connected to a different Class 1E power source, thus establishing two redundant and independent monitoring trains or channels. The TC/CCM system assemblies are housed in different compartments of the same cabinet. The cabinet is seismically and environmentally qualified and complies with IEEE Standard 384-1977 requirements for redundant instrument systems. Each TC/CCM assembly consists of a microprocessor-based chassis, a remote plasma-display chassis, two optically isolated communications interface translators, a termination rack, an analog meter, and a printer.

During the audit review, the staff focused on (1) the interface between the safety-related and nonsafety-related circuits and (2) software verification and validation (V&V) procedures used in the development of the TC/CCM system. The Class 1E microprocessor-based ICC monitoring system communicates with the Technical Support Center (TSC) and, optionally, with the plant computer. Isolation devices are used for this Class 1E/non-Class 1E interface. In response to a staff request, the applicant provided qualification information on these isolation devices (Rake, 1983). A review of the information shows that the testing consisted of applying maximum credible faults (580-V ac and ±250-V dc) in both the transverse and common modes. The acceptance criteria are that "with the credible fault applied, the analog output, the digital printer output, and the remote display shall be in a normal operation mode and provide normal information within the execution cycle time of the microprocessor." The staff requires that the applicant submit the test results of this qualification program before the plant operates above 5% of rated thermal power. No further staff evaluation will be issued on this matter unless an unanticipated problem arises from the review of the test results.

At a 2-day meeting with Westinghouse to discuss and obtain information on the V&V procedures, the staff learned that the basic document used for the V&V program is the Westinghouse Integrated Protection System (IPS) Software Standard, SD-IPLS-590 (dated July 14, 1978). This document was also used as the basis for V&V procedures for the development of the RESAR 414 IPS. The staff has concluded that procedures (from the development of functional requirements to the preproduction system testing to the manufacturer, field installation, testing, and handling of changes) appear to meet reasonable V&V requirements for traceability of as-built documentation.

On the basis of its review of information and discussions with the applicant, the staff finds that the TC/CCM system manufactured by Westinghouse and proposed for installation at Wolf Creek complies adequately with the requirements set forth by NUREG-0737 for backup systems for the core exit thermocouple-based ICC primary monitoring system. However, several staff recommendations should be noted. The applicant should obtain from Westinghouse, as a minimum, the latest version of (1) the technical manual (WNES 9002-TC/CCM-002/003), (2) the reports associated with software design specifications (No. 955438) and functional requirements (No. 955439), and (3) documentation on test specifications (No. T324A23). This information is necessary for proper, efficient implementation and operation of the TC/CCM system. Use of these documents will have a positive impact on the reliability of the system. Areas that require particular attention by the applicant are

(1) administrative procedures for the modification of setpoints, system parameters, and disabling of failed sensors' signals

(2) procedures for periodic testing

(3) procedures for resolution of ambiguities between the redundant systems

Filed deficiencies found during inplant testing indicate a need to revise associated test specifications. However, the current V&V procedures do not systematically initiate a test specification revision when a field deficiency is found. Therefore, the staff recommends that a mechanism be developed to ensure that relevant field experience is incorporated into the test specifications.

Finally, the applicant should be made aware of variations among individual (same make and model) electronic components in their ability to consistently meet the manufacturers' specifications. "Quality control" programs for manufacturers may allow various electronic component characteristics to degrade in later production phases. In such cases, it has been noted that where a component initially passed a specific test, later production models may fail while subjected to the same tests. The staff recommends that, should this situation continue to occur, testing programs for TC/CCM systems should be developed that ensure that the individual electronic equipment item meets manufacturer design specifications.

The SNUPPS design incorporates a Class 1E microprocessor-based plasma display system to provide information of ICC to the control room operator. A detailed description of this display system is provided above. The applicant was required to submit the results of the qualification testing associated with the isolation devices used for the interface between the safety-related and nonsafety-related circuits. In response to this requirement, the licensee submitted information (Westinghouse Topical Report WCAP-10621, "Westinghouse Thermocouple/Core Cooling Monitor System Isolation Tests," dated July 1984) by letter dated July 26, 1984. The staff evaluation of this information follows.

The Class 1E microprocessor-based ICC monitoring system communicates (via isolation devices) with the TSC and, optionally, with the plant computer. Circuitry associated with the TSC and the plant computer is nonsafety related. The purpose of the qualification testing was to demonstrate that the isolation devices used will provide adequate protection for the Class 1E portion of the design.

The test configuration allowed the Class 1E portion of the system to be monitored and evaluated while subjecting the nonsafety-related portion of the system to credible faults. Simulated thermocouple readings were fed to the TC/CCM microprocessor. To simulate the normal configuration, the processed signals were then input to the Class 1E CCM, remote display, remote printer, and isolation device. These signals were monitored before and during the fault application (i.e., the remote visual display and printer were checked for any changes from pretest conditions). Also, an oscilloscope was connected to the input of the optical isolator so that any feedback (from nonsafety to safety) through the isolator could be detected while applying the faults. The test consisted of applying maximum credible faults (580 V ac, 120 V ac, and ±250 V dc), in the transverse mode, to the output side of the isolator.

The test results showed that the Class 1E input to the isolator was unaffected by the fault applications (i.e., the fault did not propagate through the optical isolation device). The Class 1E remote display and printer showed no changes from pretest conditions while the faults were being applied. Also, no spurious signals were noted on the oscilloscope which was connected to the optical isolator input.

Conclusion

The above test results confirm that the Class 1E microprocessor-based plasma display system will provide normal information to the operator while being subjected to a maximum credible fault and, thus, the acceptance criteria have been adequately met. The staff, therefore, concludes that the isolation capability of the optical isolator has been satisfactorily demonstrated through testing. Thus, the isolator is adequate for use in the TC/CCM system.

On the basis of the above information, including the recommendations, the staff considers the Wolf Creek microprocessor-based plasma display system acceptable for use as a backup to the thermocouple-based ICC primary monitoring system.

III.D.1.1 Integrity of Systems Outside Containment Likely To Contain Radioactive Materials

During a severe transient or accident, some systems outside the containment--such as the safety injection system, residual heat removal system, containment spray system, or chemical and volume control system--may contain very high levels of radioactive water. Any leakage from these systems will release highly radioactive material into the air, which could then be processed and released to the atmosphere via the filtered ventilation exhaust system. In addition to being environmentally offensive by contributing to the accidental doses to members of the public, the airborne activity within the plant spaces would be burdensome to operating personnel. Therefore, every reasonable effort should be made to minimize leakage from said systems. This effort (or program) is the focus of NUREG-0737, Item III.D.1.1.

The Wolf Creek Plan, submitted by the applicant and dated March 30, 1984, includes the following:

Wolf Creek Unit 1 Plan for Minimizing Leakage

1. The Operations Department will inspect all portions of the Containment Spray, Safety Injection, Chemical and Volume Control, Residual Heat Removal, and Nuclear Sampling Systems that reside outside the containment. This inspection will be performed prior to initial fuel load and once between each refueling cycle. Maintenance/repair will be initiated to correct leaks as they are found.

2. Wolf Creek has established a preventative maintenance program whereby bearings, wear rings, seals, and packing will be replaced at routine intervals. In addition, Wolf Creek is committed to perform inservice inspection on systems in accordance with ASME Section XI. This section requires leakage testing on systems at operating pressure.

Staff Evaluation

The staff considers the implementation of the planned surveillance and system leak testing programs, discussed above, proper methods for controlling unacceptable leakage from systems containing highly radioactive water. The applicant should confirm that leakage reduction activities are applied to the reactor coolant pump seal water supply and return system as part of the chemical and volume control system inspection program. Accordingly, Confirmatory Item (B)28, Item III.D.1.1, is acceptably addressed by the applicant and it is considered completed.

Item III.D.1.1 of NUREG-0737 also lists the gaseous waste processing system for leakage reduction surveillance and maintenance activities. The applicant has not included this system for the periodic inspection efforts that are to be applied to systems outside containment containing reactor coolant and must function during an accident. In a discussion on October 9, 1984, the applicant stated that because the gaseous waste processing system is normally continuously operated, any leakage would be detected by the routine monitoring instrumentation such as area radiation monitors or pressure or system flow abnormalities.

This continuous check on system operation and integrity exceeds the periodic (once each year or refueling) inspection level received by those systems included in the leakage reduction program.

The staff agrees with this appraisal and also notes that, because the Wolf Creek gaseous waste processing system components are designed for handling radioactive gases (consistent with RG 1.143), leakage pathways to outside the system are minimized. In addition, the system is isolated during a loss-of-coolant accident. On the basis of these considerations, the system is not likely to contribute to airborne activity. Accordingly, the Wolf Creek gaseous waste processing system does not need to receive the periodic leakage reduction surveillance and testing activities covered by the Item III.D.1.1 leakage reduction program.

Table 22.1 Comparison of torque available and torque required

Angle of opening	Torque avail in.-lb	Max* predicted ΔP	Torque req at ΔP=22.9, in.-lb	Ratio avail/req Torque	Max predicted ΔP	Torque req at ΔP=47.2, in.-lb	Ratio avail/req torque
Closed	29,900	22.9	6,220**	4.8	47.2	6,220**	4.8
10	25,600	22.9	1,785	14.5	47.2	3,239	7.9
20	23,000	22.8	2,223	10.4	47.2	3,961	5.8
30	21,700	22.7	2,223	9.8	46.8	3,961	5.5
40	21,400	22.2	2,223	9.6	45.8	3,961	5.4
50	22,100	21.0	2,728	8.0	44.0	4,791	4.6
60	23,900	18.2	3,412	7.0	39.2	5,918	4.0
70	27,200	12.8	3,662	7.4	30.8	6,330	4.3
80	33,000	7.3	3,581	9.2	16.1	6,197	5.3
90	43,000	4.3	3,581	12.0	9.9	6,197	6.9

*During the 3-second closure period, the containment pressure rises from 14 psig. All predicted ΔPs are based on flow conditions at 22.9 psig at the inlet to the purge piping.

**The 6,200 in.-lb torque required is based on 60 psig, which is the containment/valve design pressure. The maximum calculated LOCA pressure is 47.2 psig.

Table 22.2 Stress in critical parts

Stress consideration	Allowable[a] stress[b], ksi	Calculated stresses, ksi Closed	Case 1[d] at 70°F	Case 2[d] at 70°F
Shaft at disc hub (1.55) (bending and torsion)	52.5	4.7	2.3	4.3
Shaft at disc hub (0.75S) (torsion and tranverse shear)	26.25	6.1	3.1	5.7
Shaft at pin connection (0.75S)	26.25	5.8	3.7	6.3
Shaft at key connection (0.75S)	26.25	5.3	3.1	5.4
Bushing	8.5[e]	1.9	0.7	1.5

[a]These allowables were derated to 98% of that shown to account for the 320°F design temperature.

[b]Based on ASME Code Section III values (S) of 35 ksi for 17-4PH, condition H1100 (Table I-7.1, Appendix I). The allowable stres of 35 ksi is a conservative figure since an (S) value of 36.2 is allowed by the code for H1075 shaft material.

[c]Closed position stresses based on 60 psid across valve.

[d]Refer to discussion on Table 22.1 (above) for the definitions of Cases 1 and 2. Stresses are reported for 70° open since they are the maximum for opening angles 10°-70°.

[e]Graphite-filled bronze.

APPENDIX A

CONTINUATION OF CHRONOLOGY OF NRC STAFF RADIOLOGICAL SAFETY REVIEW OF WOLF CREEK

The following is an update of the chronology commencing with November 21, 1983, and ending with January 25, 1985:

November 21, 1983	Letter from applicant concerning the Wolf Creek Emergency Plan.
November 23, 1983	Letter from applicant concerning Midland Ross superstruts.
November 28, 1983	Letter from applicant concerning procedures generation package.
November 28, 1983	Letter from applicant concerning main dam seepage.
November 30, 1983	Letter from SNUPPS concerning NUREG-0737, Item I.D.1, "Control Room Design Reviews."
December 6, 1983	Letter from applicant concerning radiological effluent Technical Specifications.
December 13, 1983	Letter from applicant concerning certain pollution control bonds.
December 16, 1983	Letter to applicant requesting information on facility staffing.
December 19, 1983	Letter to applicant enclosing two copies of Supplement No. 4 to the Wolf Creek SER (NUREG-0881).
December 20, 1983	Letter to applicant concerning comments on steam generator tube plugging margin analysis.
December 22, 1983	Letter from SNUPPS concerning correction to preoperational test abstract.
December 23, 1983	Letter from applicant concerning facility staffing survey.
December 30, 1983	Letter to applicant concerning safety evaluation report issues.
December 30, 1983	Letter to applicant transmitting 20 printed copies of Supplement No. 4 to the Wolf Creek SER (NUREG-0881).
January 4, 1984	Letter to applicant concerning design verification activities.

January 6, 1984	Letter from SNUPPS concerning Administrative Procedures for Solid Water Operation.
January 9, 1984	Letter from SNUPPS concerning safety-related preoperational test procedures.
January 13, 1984	Letter from applicant concerning quality assurance program change.
January 13, 1984	Letter from SNUPPS concerning safety parameter display system.
January 16, 1984	Letter from SNUPPS concerning containment purge valve operability.
January 25, 1984	Representatives from NRC, UE, KG&E, and SNUPPS meet at the Callaway site in Fulton, Missouri, to evaluate the ultrasonic inspection of cast stainless steel welds and to discuss the preservice inspection program. (Summary issued April 3, 1984.)
January 27, 1984	Letter from SNUPPS concerning control of heavy loads.
January 27, 1984	Letter to applicant concerning Wolf Creek Physical Security Plan--Designation of Vital Equipment and Employee Screening Program.
January 31, 1984	Letter to applicant concerning replacement of water bags for radiation shielding.
January 31, 1984	Letter from SNUPPS concerning staff review of grid undervoltage.
January 31, 1984	Letter from SNUPPS concerning staff review of RG 1.63, Position 1.
February 1, 1984	Letter from SNUPPS concerning environmental qualification of safety-related electrical equipment.
February 1, 1984	Letter from SNUPPS concerning fire protection.
February 2, 1984	Letter from SNUPPS concerning safe shutdown following loss of instrument bus.
February 2, 1984	Letter from SNUPPS concerning diesel generator rocker arm lube oil temperature.
February 2, 1984	Letter from SNUPPS concerning Revision 13 to SNUPPS FSAR.
February 2, 1984	Letter from SNUPPS concerning seismic qualification of equipment.
February 2, 1984	Letter from SNUPPS concerning steam generator tube plugging requirements.

The Westinghouse material submitted is proprietary and withholding from public disclosure is requested.

February 3, 1984	Letter to applicant concerning deletion of home telephone numbers, unlisted utility numbers, etc., from emergency plans.
February 6, 1984	Representatives from NRC, KG&E, and SNUPPS meet in Bethesda, Maryland, to discuss differences and similarities in the design implementation process between Wolf Creek and Callaway.
February 7, 1984	Letter from SNUPPS concerning environmental qualification of safety-related electrical equipment.
February 7-9, 1984	Representatives from NRC, KG&E, and SNUPPS meet at the Wolf Creek site in Burlington, Kansas, to audit the fire protection design for the Wolf Creek plant.
February 8, 1984	Letter from SNUPPS concerning NRC request for information regarding Callaway and Wolf Creek preservice inspection programs.
February 9, 1984	Letter from applicant concerning initial test program changes.
February 14, 1984	Letter from SNUPPS concerning status of SER open items.
February 16, 1984	Letter from applicant concerning emergency plan letters of agreement and procedures.
February 17, 1984	Representatives from NRC, KG&E, UE, and SNUPPS meet in Bethesda, Maryland, to discuss the Technical Specification requirements on slave relay testing. (Summary issued March 8, 1984.)
February 17, 1984	Letter from SNUPPS concerning reactor cavity shielding.
February 22, 1984	Letter from SNUPPS concerning diesel generator rocker arm lube oil temperature.
February 23, 1984	Letter from SNUPPS concerning staff review of instrumentation and control systems.
February 23, 1984	Letter from SNUPPS concerning environmental qualification justifications for interim operation.
February 24, 1984	Letter from applicant concerning offsite dose calculation manual (ODCM).
February 24, 1984	Letter from SNUPPS concerning fire protection review.

February 27, 1984	Letter from SNUPPS concerning slave relay testing Technical Specification requirements.
February 29, 1984	Letter to applicant concerning request for additional information on staff site audit regarding instrumentation and control systems.
February 29, 1984	Letter to applicant requesting additional information related to the Wolf Creek Procedures Generation Package.
March 1, 1984	Letter to Westinghouse withholding from public disclosure CAW-84-003--SNUPPS Plants--Steam Generator Tube Plugging Margin Evaluation.
March 5, 1984	Letter from applicant concerning operating shift experience.
March 7, 1984	Letter to applicant concerning NRC Caseload Forecast Panel visit to Wolf Creek.
March 9, 1984	Letter from applicant concerning Wolf Creek design verification activities.
March 14, 1984	Letter from SNUPPS concerning Revision 14 to SNUPPS FSAR.
March 14, 1984	Letter from SNUPPS concerning fire protection.
March 15, 1984	Letter from SNUPPS concerning containment isolation dependability.
March 15, 1984	Representatives from NRC, KG&E, UE, and SNUPPS meet in Bethesda, Maryland, to discuss an appeal of the containment purge operation time limit given in the Callaway Technical Specifications. (Summary issued March 21, 1984.)
March 15, 1984	Letter from applicant transmitting the best-estimate dates for completion of construction--September 23, 1984; fuel loading--September 23, 1984; and commercial operation--Spring 1985.
March 16, 1984	Letter from applicant concerning Technical Specifications.
March 16, 1984	Letter from applicant concerning main dam seepage (Wolf Creek Safety Evaluation Report, Confirmatory Item A(2).
March 20, 1984	Letter from SNUPPS concerning SNUPPS Technical Specifications.
March 21, 1984	Letter from SNUPPS concerning NRC audit of SNUPPS Control Room Design Reviews, week of February 27, 1984.

March 23, 1984	Letter from applicant concerning NRC Caseload Forecast Team visit to Wolf Creek.
March 28, 1984	Letter from SNUPPS concerning control of heavy loads.
March 30, 1984	Letter from applicant concerning Confirmatory Item B(28) (III.D.1.1).
March 30, 1984	Letter from applicant concerning Revision 12 to the Wolf Creek FSAR Addendum.
April 2, 1984	Letter to applicant requesting additional information on instrument error for containment pressure setpoints.
April 2, 1984	Letter to applicant concerning schedule for implementing recommendations in FEMA's interim findings.
April 4, 1984	Letter from applicant transmitting Revision 12 to the Wolf Creek FSAR Addendum.
April 4, 1984	Letter from applicant regarding Procedures Generation Package.
April 6, 1984	Letter from SNUPPS concerning Chapter 15, "Analyses Affected by Assumption of Zero Boron Downstream of RWST."
April 9, 1984	Letter from SNUPPS concerning instrument error for containment pressure setpoints.
April 11, 1984	Letter from applicant transmitting the 1983 annual financial reports.
April 13, 1984	Letter from SNUPPS concerning final response to Generic Letter 83-10c.
April 16, 1984	Letter from SNUPPS concerning staff review on instrumentation and control systems.
April 17, 1984	Letter to applicant requesting additional information on SNUPPS Technical Specifications.
April 17, 1984	Letter from SNUPPS concerning a revision in the diesel generator start time.
April 17, 1984	Letter from SNUPPS concerning boron precipitation in the boron injection tank and associated piping.
April 23, 1984	Letter from SNUPPS concerning staff questions on instrumentation and control systems in the SNUPPS Technical Specifications.

April 23, 1984	Letter from SNUPPS concerning SNUPPS Technical Specifications on use of RHR suction relief valves for cold overpressurization protection.
April 23, 1984	Letter from SNUPPS concerning revision to diesel generator start time.
April 23, 1984	Letter from applicant concerning FEMA Interim Findings.
April 24, 1984	Letter from SNUPPS concerning NUREG-0737, Item II.B.3, Postaccident Sampling Capability.
April 25, 1984	Letter from SNUPPS concerning Wolf Creek Preoperational Test Program.
April 26, 1984	Letter from SNUPPS concerning SNUPPS Technical Specifications on use of RHR suction relief valves for cold overpressurization protection.
April 30, 1984	Letter from applicant concerning Emergency Plan letters of agreement.
April 30, 1984	Letter from applicant concerning main dam seepage (Wolf Creek Safety Evaluation Report, Confirmatory Item A(2).
May 1, 1984	Representatives from NRC and Wichita Chamber of Commerce meet in Bethesda, Maryland, to discuss economic aspects of the construction and operation of Wolf Creek Generating Station.
May 2, 1984	Letter from SNUPPS concerning revision in diesel generator start time.
May 7, 1984	Letter from applicant concerning request for additional information on Technical Specifications.
May 8, 1984	Letter from applicant concerning station security.
May 9, 1984	Letter from applicant concerning NRC staff visit to Wolf Creek to conduct a caseload forecast.
May 9, 1984	Letter from Westinghouse concerning withholding from public disclosure information in CAW-84-25, "SNUPPS/Westinghouse Interconnecting Wiring Diagrams and Process Control Block Diagrams Associated with Pressurizer Pressure Input, Wolf Creek and Callaway."
May 11, 1984	Letter from applicant concerning Wolf Creek human factors modifications.
May 14, 1984	Letter to applicant requesting additional information on Wolf Creek and Callaway Technical Specifications.

May 15, 1984	Letter to applicant requesting additional information on preoperational testing.
May 16, 1984	Letter from SNUPPS concerning SNUPPS Technical Specifications reactor systems issues.
May 16, 1984	Letter to applicant requesting additional information on instrumentation and control Technical Specifications.
May 18, 1984	Letter from SNUPPS concerning Technical Specification questions on instrumentation and control systems.
May 18, 1984	Letter from applicant concerning Caseload Forecast Team visit to Wolf Creek.
May 24, 1984	Representatives from NRC and KG&E meet in Burlington, Kansas, to review the construction status at Wolf Creek.
May 24, 1984	Letter from SNUPPS concerning pump and valve operability FSAR revision.
May 25, 1984	Letter from SNUPPS concerning SNUPPS Technical Specifications on reactor systems.
May 25, 1984	Letter to applicant concerning operating shift staffing for Wolf Creek.
May 31, 1984	Letter from SNUPPS concerning SNUPPS Technical Specifications on reactor systems.
June 4, 1984	Letter to applicant concerning Wolf Creek Generating Station Physical Security Plan.
June 5, 1984	Letter to applicant concerning results of preimplementation audit of Callaway and Wolf Creek control rooms.
June 6, 1984	Letter to applicant concerning fuel load forecast information.
June 7, 1984	Letter to applicant concerning design verification activities at Wolf Creek.
June 12, 1984	Letter to applicant requesting additional information on steam generator tube rupture event.
June 14, 1984	Letter from applicant concerning followup on transmittal of Revision 3 of the Wolf Creek Physical Security Plan.
June 18, 1984	Letter from applicant transmitting a revision to the OL.
June 19, 1984	Letter from applicant concerning Wolf Creek schedule.

June 21, 1984	Letter from SNUPPS concerning accident sampling capability.
June 21, 1984	Letter from applicant concerning Wolf Creek schedule.
June 21, 1984	Letter from applicant concerning polar crane testing.
June 25, 1984	Letter to applicant concerning request for additional information conformance to RG No. 1.97.
June 26, 1984	Letter from SNUPPS transmitting Revision 15 to the SNUPPS FSAR.
June 27, 1984	Letter to applicant requesting deletion of the Technical Specification on boron dilution mitigation system.
June 28, 1984	Letter from application transmitting a revision to the OL.
June 28, 1984	Letter to applicant transmitting proposed changes to Wolf Creek Technical Specifications.
June 28, 1984	Letter from applicant regarding State and County emergency plan submittal to FEMA.
June 29, 1984	Letter from SNUPPS concerning inadequate core cooling instrumentation testing.
June 29, 1984	Letter from SNUPPS concerning Revision 1 to Detailed Control Room Design Review Summary Report for SNUPPS.
June 29, 1984	Letter from applicant concerning Revision 13 to the Wolf Creek FSAR Addendum.
July 2, 1984	Letter to applicant requesting additional information on Wolf Creek Physical Security Plan.
July 3, 1984	Letter from applicant concerning station security in response to NRC request for modifications.
July 9, 1984	Letter from applicant concerning boron dilution mitigation system Technical Specifications.
July 9, 1984	Letter from applicant concerning Wolf Creek schedule.
July 13, 1984	Representatives from NRC, SNUPPS, UE, and KG&E meet in Bethesda, Maryland, to discuss the Detailed Control Room Design Review for the SNUPPS plants. (Summary issued July 18, 1984.)
July 20, 1984	Letter from applicant concerning Technical Specifications.
July 25, 1984	Letter from applicant concerning initial test program changes.

July 26, 1984	Letter to applicant requesting additional information on seismic and dynamic qualification.
July 26, 1984	Letter to applicant transmitting a summary of the May 24, 1984, Caseload Forecast Panel visit.
July 26, 1984	Letter from SNUPPS concerning inadequate core cooling instrumentation testing.
July 27, 1984	Letter from SNUPPS concerning seismic and dynamic qualification.
July 30, 1984	Letter from applicant concerning Technical Specifications.
July 31, 1984	Letter from applicant concerning inservice testing program for pumps and valves.
July 31, 1984	Letter from applicant concerning State and County Emergency Response Plans.
August 1, 1984	Representatives from NRC and KG&E meet in Bethesda, Maryland, to discuss KG&E's proposed dual function shift technical advisor.
August 7, 1984	Letter from applicant concerning shift technical advisors.
August 8, 1984	Letter from applicant transmitting the Wolf Creek Milestone Schedule for August 1, 1984.
August 10, 1984	Representatives from NRC, UE and KG&E meet in Bethesda, Maryland, to discuss the isolation features during a control room fire at the SNUPPS plants.
August 10, 1984	Letter from SNUPPS concerning fire protection review.
August 10, 1984	Letter from applicant concerning Procedures Generation Package.
August 14, 1984	Representatives from NRC, KG&E, UE, and SNUPPS meet in Bethesda, Maryland, to discuss staff's position on the SNUPPS safe shutdown analysis. (Summary issued August 17, 1984.)
August 15, 1984	Representatives from NRC, KG&E, UE, and SNUPPS meet in Bethesda, Maryland, to discuss fire protection. (Summary issued August 22, 1984.)
August 15, 1984	Letter from applicant concerning station security in response to NRC questions.
August 15, 1984	Letter from applicant concerning Environmental Protection Plan.

August 15, 1984	Letter from applicant concerning Wolf Creek design verification activities.
August 16, 1984	Letter from SNUPPS concerning conformance to RG 1.97.
August 22, 1984	Letter from applicant transmitting the Wolf Creek Milestone Schedule as of August 15, 1984.
August 22, 1984	Representatives from NRC, UE, KG&E, and SNUPPS meet in Bethesda, Maryland, to discuss the SNUPPS fire protection review. (Summary issued August 31, 1984.)
August 22, 1984	Letter from applicant concerning Revision 4 to Wolf Creek Safeguard Contingency Plan.
August 23, 1984	Letter from SNUPPS concerning fire protection.
August 28, 1984	Letter from SNUPPS concerning Wolf Creek Preservice Inspection Plan.
August 31, 1984	Letter from applicant concerning core damage assessment methodology.
August 31, 1984	Letter from applicant concerning FSAR changes relating to replacement and requalification training program.
September 5, 1984	Letter from SNUPPS concerning human factors involvement in SNUPPS SPDS design.
September 6, 1984	Letter from SNUPPS concerning preservice inspection relief request for the Wolf Creek plant.
September 7, 1984	Letter from SNUPPS concerning steam generator tube rupture event.
September 10, 1984	Letter from applicant concerning Wolf Creek schedule as of September 1, 1984.
September 19, 1984	Letter from SNUPPS concerning preservice inspection relief request for the Wolf Creek plant.
September 21, 1984	Order issued by ASLAB advising that the date in ALAB-784 states September 12, 1984, when it should say October 12, 1984.
September 27, 1984	Letter from applicant concerning changes to FSAR description of Quality Assurance Program.
September 28, 1984	Letter from applicant concerning Revision 5 to Wolf Creek Physical Security Plan.
September 28, 1984	Letter from applicant concerning SNUPPS licensing submittals.

September 28, 1984	Letter from applicant transmitting Revision 14 to the FSAR.
October 2, 1984	Letter from SNUPPS concerning equipment qualification justification for interim operation (JIO).
October 5, 1984	Letter from SNUPPS transmitting Revision 16 to SNUPPS FSAR.
October 5, 1984	Letter to applicant concerning Wolf Creek Physical Security Plan - Request for Additional Information.
October 8, 1984	Letter from applicant concerning Wolf Creek ASLB Initial Decision.
October 8, 1984	Letter from applicant transmitting the Wolf Creek Milestone Schedule as of October 1, 1984.
October 12, 1984	Letter from SNUPPS concerning justification for interim operation - seismic qualification.
October 17, 1984	Letter from applicant concerning Wolf Creek Generating Station Emergency Plan.
October 17, 1984	Letter from applicant concerning process control program (PCP).
October 18, 1984	Letter to applicant concerning design verification activities.
October 22, 1984	Letter from SNUPPS concerning neutron shielding water can activation.
October 23, 1984	Order issued by ASLAB requesting results from the staff of the November 7, 1984, emergency exercise for Wolf Creek.
October 25, 1984	Letter to applicant informing that the time to review ALAB-784 has expired; therefore, the decision became final on October 23, 1984.
November 6, 1984	Letter from applicant concerning partial exemption from Appendix J.
November 7, 1984	Letter to applicant concerning the Wolf Creek Technical Specifications.
November 21, 1984	Letter to applicant concerning Wolf Creek Generating Station Shift Advisors.
November 27, 1984	Meeting with representatives from NRC, UE, KG&E, and SNUPPS in Bethesda, Maryland, to discuss the SNUPPS utilities revised steam generator tube rupture analysis.

November 28, 1984	Meeting with representatives from NRC, KG&E, and SNUPPS in Bethesda, Maryland, to discuss status of completion and future schedule for licensing Wolf Creek.
November 30, 1984	Letter from applicant transmitting Revision 6 to Security Plans.
December 1, 1984	Letter from applicant requesting issuance of an operating license.
December 3, 1984	Letter from SNUPPS concerning steam generator tube rupture event.
December 3, 1984	Letter from applicant concerning Technical Specifications.
December 3, 1984	Letter from applicant concerning boron-dilution Technical Specifications.
December 3, 1984	Letter from SNUPPS concerning FSAR requirements for structural steel welding.
December 3, 1984	Letter from applicant transmitting the Wolf Creek schedule as of November 24, 1984.
December 3, 1984	Letter from applicant concerning Wolf Creek ASLB Initial Decision.
December 3, 1984	Letter from applicant concerning Environmental Protection Plan (Non-Radiological).
December 4, 1984	Letter to applicant concerning Wolf Creek Generating Station Draft License.
December 5, 1984	Letter from SNUPPS concerning emergency core cooling system (ECCS) check valve testing.
December 5, 1984	Letter from applicant concerning testing of pressurizer power-operated relief valves.
December 7, 1984	Letter from applicant concerning Wolf Creek Generating Station Shift Advisors.
December 7, 1984	Letter from applicant concerning reactor coolant system resistance temperature detector (RTD) testing.
December 10, 1984	Letter from applicant concerning Technical Specifications.
December 10, 1984	Letter from applicant concerning the Wolf Creek Emergency Plan.
December 12, 1984	Letter from applicant concerning testing of radiation monitors.

December 12, 1984	Letter from applicant concerning standby diesel generator lube oil keep warm pump.
December 14, 1984	Letter from SNUPPS concerning seismic qualification.
December 17, 1984	Letter from applicant transmitting the Wolf Creek schedule as of December 12, 1984.
December 18, 1984	Letter from SNUPPS concerning FSAR changes.
December 21, 1984	Letter from applicant concerning FSAR described programs.
December 21, 1984	Letter from applicant concerning FSAR changes relating to the Initial Test Program.
December 21, 1984	Letter from applicant concerning Wolf Creek Draft Operating License.
December 21, 1984	Letter from applicant concerning Wolf Creek Initial Startup Test Program reviews. This letter transmits Westinghouse proprietary information on post-critical testing; withholding requested.
December 21, 1984	Letter from SNUPPS concerning seismic qualification.
December 24, 1984	Letter to Westinghouse Owners Group concerning emergency response guidelines.
December 28, 1984	Letter from SNUPPS concerning Revision 17 to SNUPPS FSAR.
January 2, 1985	Letter from applicant concerning containment systems section in Technical Specifications.
January 4, 1985	Letter from applicant transmitting the Wolf Creek schedule as of December 28, 1985.
January 9, 1985	Letter from applicant concerning offsite dose calculation manual (ODCM).
January 10, 1985	Representatives from NRC and KG&E meet in Burlington, Kansas, at the site to permit NRC management to assess the operational readiness of Wolf Creek facility.
January 11, 1985	Letter from applicant concerning Revision 7 to the Security Plan.
January 11, 1985	Letter from applicant concerning changes to FSAR description of the quality program.
January 13, 1985	Letter from applicant concerning fire protection.
January 14, 1985	Letter from applicant concerning FSAR changes relating to the raceway and cable separation.
January 16, 1985	Letter from SNUPPS concerning FSAR changes.

January 18, 1985	Letter from SNUPPS concerning Wolf Creek Technical Specifications.
January 18, 1985	Letter from SNUPPS concerning FSAR changes relating to the diesel generator lube oil level control tank.
January 23, 1985	Letter from applicant concerning FSAR changes relating to RVLIS high volume sensor qualification.
January 25, 1985	Letter from applicant concerning Wolf Creek Technical Specifications.

APPENDIX B

BIBLIOGRAPHY

Atomic Safety and Licensing Board, Initial Decision (Operating License) in the Matter of Kansas Gas & Electric Co., et al. (Wolf Creek Generating Station, Unit No. 1), July 2, 1984.

---, Order (re: Applicant's Motion for Clarification of Initial Decision) in the Matter of Kansas Gas & Electric Co., et al. (Wolf Creek Generating Station, Unit No. 1), July 26, 1984.

Consolidated Controls Corporation, "Failure Modes and Effects Analysis for Load Shed and Emergency Sequencer for SNUPPS," Bethel, Connecticut, October 10, 1978.

---, "Reliability Analysis for Load Shedding and Emergency Sequencing," Bethel, Connecticut, September 14, 1981.

Federal Emergency Management Agency, "Interim Findings on the Adequacy of Radiological Emergency Response Planning by State and Local Governments at the Wolf Creek Generating Station, Burlington, Kansas," February 2, 1984.

---, Interim Findings on the Adequacy of Radiological Emergency Response Planning by State and Local Governments at the Wolf Creek Generating Station, Burlington, Kansas," September 28, 1984.

Rake, E. P. (Westinghouse), letter to G. Edison (NRC), September 1, 1983.

SNUPPS (Standardized Nuclear Unit Power Plant System), "Control Room Fire Hazard Analysis," November 15, 1982.

---, "Report of Independent Review of Environmental Qualification Programs to NUREG-0588," March 10, 1983; Rev. 1, May 27, 1983.

U.S. Nuclear Regulatory Commission, Generic Letter 82-33, see NUREG-0737, Suppl. 1.

---, Generic Letter 84-16, June 27, 1984.

---, NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," Vols. 1 and 2, April 1978; Vol. 3, December 1978; Vol. 4, March 1980.

---, NUREG-0577, Rev. 1, "Potential for Law Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports, Unresolved Safety Issue A-12," October 1983.

---, NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," November 1979; Rev. 1, July 1981.

---, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36," July 1980.

---, NUREG-0654/FEMA-REP-1, Rev. 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," November 1980.

---, NUREG-0660, "NRC Action Plan Developed As a Result of the TMI-2 Accident," Vols. 1 and 2, May 1980; Rev. 1, August 1980.

---, NUREG-0700, "Guidelines for Control Room Design Reviews," August 1981.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

---, NUREG-0737, Suppl. 1, "Clarification of TMI Action Plan Requirements: Requirements for Emergency Response Capability," December 17, 1982.

---, NUREG-0744, Rev. 1, "Resolution of the Reactor Vessel Materials Toughness Safety Issue," Vols. 1 and 2, October 1982.

---, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," LWR Edition, July 1981.

---, NUREG-0801, For comment, "Evaluation Criteria for Detailed Control Room Design Review," October 1981.

---, NUREG-0809, "Safety Evaluation Report Review of Resistance Temperature Detector Time Response Characteristics," August 1981.

---, NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures," August 1982.

---, NUREG-0927, Rev. 1, "Evaluation of Water Hammer Occurence in Nuclear Power Plants," August 1984.

---, NUREG/CR-1580, "Human Engineering Guide to Control Room Evaluation," Draft, July 1980.

---, "10 CFR Part 50 Analysis of Potential; Pressurized Thermal Shock Events," 49 Fed. Reg. 4498 (February 7, 1984).

---, SECY-82-465, "Pressurized Thermal Shock (PTS)," November 23, 1982.

---, Office of Inspection and Enforcement, IE Bulletin 79-15, "Deep Draft Pump Deficiencies," July 11, 1979.

---, IE Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," November 30, 1979.

---, IE Circular 79-02, "Failure of 120-Volt Vital AC Power Supplies," January 16, 1979.

---, IE Information Notice 83-23, "Inoperable Containment Atmosphere Sensing Systems," April 25, 1983.

---, Inspection Report 50-483/83-83, April 6, 1984.

---, Inspection Report 50-482/84-25, December 21, 1984.

---, Inspection Report 50-482/84-29, October 9, 1984.

---, Inspection Report 50-482/85-02, to be issued.

---, Temporary Instruction 2514/01, Rev. 2, "OL Applicant Inspection Requirements," May 8, 1981.

Westinghouse, WCAP-8691(P), Rev. 1, "Fuel Rod Bow Evaluation," October 1981.

---, WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model, 1981 Version," February 1982.

---, WCAP-10297P, "Dropped Rod Methodology for Negative Flux Rate Trip Plants," January 1982.

Westinghouse Owners Group, "Emergency Response Guidelines."

---, "Emergency Response Guidelines," Rev. 1.

---, "Post-Accident Core Damage Assessment Methodology," Rev. 1, March 1984.

APPENDIX C

NUCLEAR REGULATORY COMMISSION UNRESOLVED SAFETY ISSUES

C.5 Discussion of Tasks As They Relate to Wolf Creek Unit No. 1

A-1 Water Hammer

This issue has been resolved by issuance of NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants."

A-11 Reactor Vessel Materials Toughness

This issue has been resolved by issuance of NUREG-0744, "Resolution of the Task A-11, Reactor Vessel Materials Toughness Safety Issue," Vols I and II, Revision 1.

A-12 Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports

This issue has been resolved by issuance of NUREG-0577, Revision 1, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports." However, it should be noted that the requirements for resolving this issue are applied to new construction permit and preliminary design assessment plants only. Therefore, this issue is not applicable to Wolf Creek.

A-46 Seismic Qualification of Equipment in Operating Plants

The scope of Task A-46 is limited to dealing with seismic qualification of equipment in operating plants. In addition, Wolf Creek was designed on the basis of current seismic design criteria, and commitments for seismic equipment qualification are in accordance with the latest codes and standards. Therefore, the issue related to USI A-46 is not applicable to Wolf Creek.

A-48 Hydrogen Control Measures and Effects on Hydrogen Burns on Safety Equipment

This issue is limited to plants with pressure suppression containments (ice condensers for pressurized-water-reactor (PWR) plants, and Mark I, II, and III containments for boiling-water-reactor (BWR) plants). The containment for Wolf Creek is a large dry containment. Therefore, this issue is not applicable to Wolf Creek.

A-49 Pressurized Thermal Shock

The issue of pressurized thermal shock (PTS) arises because in PWRs transients and accidents can occur that result in severe overcooling (thermal shock) of the reactor pressure vessel, concurrent with or followed by repressurization.

In these PTS events, rapid cooling of the reactor vessel internal surface results in thermal stress with a maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress depends on the temperature profile across the reactor vessel wall as a function of time. The effects of this thermal stress are compounded by pressure stresses.

Severe reactor system overcooling events simultaneous with or followed by pressurization of the reactor vessel (PTS events) can result from a variety of causes. These include system transients, some of which are initiated by instrumentation and control systems malfunctions (including stuck-open valves in either the primary or secondary system), and postulated accidents such as small-break loss-of-coolant accidents (LOCAs), main steamline breaks (MSLBs), and feedwater line breaks.

The PTS issue is a concern for PWRs only after the reactor vessel has lost its fracture toughness properties and is embrittled by neutron irradiation. The standards and regulatory requirements to which the Wolf Creek reactor vessel was designed and fabricated are described in Section 5.3 of the SER.

As long as the fracture resistance of the reactor vessel material is relatively high, overcooling events are not expected to cause vessel failure. However, the fracture resistance of reactor vessel materials decreases with exposure to fast neutrons during the life of a nuclear power plant. The rate of decrease is dependent on the metallurgical composition of the vessel walls and welds. If the fracture resistance of the vessel has been reduced sufficiently by neutron irradiation, severe overcooling events could cause propagation of small flaws that might exist near the inner surface. The assumed initial flaw might be enlarged into a crack through the vessel wall of sufficient extent to threaten vessel integrity and, therefore, core cooling capability.

For the reactor pressure vessel to fail and constitute a risk to public health and safety, a number of contributing factors must be present. These factors are (1) a reactor vessel flaw of sufficient size to initiate and propagate; (2) a level of irradiation (fluence) and material properties and composition sufficient to cause significant embrittlement (the exact fluence depends on materials present, high copper content causes embrittlement to occur more rapidly); (3) a severe overcooling transient with pressurization; and (4) a crack resulting from the propagation of initial cracks of such size and location that the vessel fails.

As a result of the evaluation of the PTS issue, the staff recommended to the Commission in SECY-82-465 (November 23, 1982) actions to prevent PTS events in operating reactors. The Commission accepted the staff recommendations and directed the staff to develop a Notice of Proposed Rulemaking that would establish an RT_{NDT} screening criterion (below which PTS risk is considered acceptable), require licensees to submit present and projected values of RT_{NDT}, require early analysis and implementation of such flux-reduction programs as are reasonably practicable to avoid reaching the screening criterion, and require plant-specific PTS safety analyses before plants are within 3 calendar years of reaching the screening criterion, including analyses of proposed alternative to minimize the PTS problem.

Such a proposed rule has been published for public comment [NRC, "10 CFR Part 50 Analysis of Potential," 49 Federal Register 4498, February 7, 1984] by the staff.

The staff believes that the Wolf Creek plant could easily meet the requirements of the proposed rule.

On the basis of the above consideration, the staff concludes that the Wolf Creek plant can be operated before complete resolution of this issue and completion of the proposed rulemaking without undue risk to the health and safety of the public.

APPENDIX D

NRC STAFF CONTRIBUTORS

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

Name	Title	Review Branch
G. Bagchi	Section Leader	Equipment Qualification
R. Becker	Senior Nuclear System Engineer	Procedures and Systems Review
L. Beltracchi	Sr. Human Factors Engineer	Human Factors Engineering
S. Block	Senior Health Physicist	Radiological Assessment
J. Clifford	Operational Safety Engineer	Procedures and Systems Review
L. Crocker	Section Leader	Licensee Guidance
S. Diab	Nuclear Engineer	Reactor Systems
R. Eberly	Fire Protection Engineer	Chemical Engineering
D. Eckenrode	Human Factors Engineer	Human Factors Engineering
B. Elliot	Materials Engineer	Materials Engineering
R. Fell	Nuclear Engineer	Meteorology and Effluent Treatment
Y. Hsii	Nuclear Engineer	Core Performance
J. Huang	Sr. Containment Systems Engineer	Containment Systems
M. Hum	Sr. Materials Engineer	Materials Engineering
F. Kantor	Emergency Planning Specialist	Emergency Preparedness
S. Kim	Sr. Structural Engineer	Structural and Geotechnical
J. Knox	Sr. Reactor Systems Engineer	Electrical Power Systems
N. Laubin	Section Leader	Reactor Systems
A. Lee	Electrical Engineer	Equipment Qualification
W. LeFave	Sr. Auxiliary Systems Engineer	Auxiliary Systems
V. Leung	Reactor Engineer	Reactor Systems
B. Mann	Nuclear Engineer	Reactor Systems
L. Marsh	Section Leader	Reactor Systems
A. Masciantonio	Equipment Qualification Engineer	Equipment Qualification
W. Middleton	Engineering Psychologist	Procedures and Systems Review
C. Moon	Project Manager	Standardization and Special Projects
D. Rohrer	Emergency Planning Specialist	Emergency Preparedness
M. Schoppman	Management Systems Engineer	License Qualifications
R. Skelton	Plant Protection Analysis	Physical Security
D. Smith	Materials Engineer	Materials Engineering
R. Stevens	Reactor Engineer	Instrumentation and Control
J. Spraul	Quality Assurance Engineer	Quality Assurance
B. Turovlin	Materials Engineer	Materials Engineering

Name	Title	Review Branch
M. Wigdor	Reactor Engineer	Instrumentation and Control
R. Wright	Sr. Equipment Qualifications Engineer	Equipment Qualifications
P. Wu	Chemical Engineer	Chemical Engineering
M. Wangler	Health Physicist	Radiological Assessment

APPENDIX J

PRESERVICE INSPECTION RELIEF REQUEST EVALUATION

I. INTRODUCTION

This section was prepared with the technical assistance of Department of Energy contractors from the Idaho National Engineering Laboratory.

For nuclear power facilities whose construction permit was issued on or after July 1, 1974, 10 CFR 50.55a(g)(3) specifies that components shall meet the preservice examination requirements set forth in editions of Section XI of the ASME Boiler and Pressure Vessel Code and addenda applied to the construction of the particular component. The provisions of 10 CFR 50.55a(g)(3) also state that components (including supports) may meet the requirements set forth in subsequent editions and addenda of this Code that are incorporated by reference in 10 CFR 50.55a(b), subject to the limitations and modifications listed therein.

In letters dated September 6 and 19, 1984, the applicant submitted requests for relief from ASME Code, Section XI, requirements that the applicant has determined to be impractical and provided supporting information pursuant to 10 CFR 50.55a(a)(2)(i). Therefore, the staff evaluation consisted of reviewing the applicant's submittals to the requirements of the applicable Code edition and addenda and determining if relief from the Code requirements was justified.

II. TECHNICAL REVIEW CONSIDERATIONS

A. The construction permit for the Wolf Creek Generating Station was issued on May 17, 1977. In accordance with 10 CFR 50.55a(g)(3), components (including supports) that are classified as ASME Code Class 1 and 2, have been designed and provided with access to enable the performance of required preservice examinations set forth in the 1977 Edition of ASME Code, Section XI, including the addenda through Summer 1978.

B. Verification of as-built structural integrity of the primary pressure boundary is not dependent on the Section XI preservice examination. The applicable construction codes to which the primary pressure boundary was fabricated contain examination and testing requirements that by themselves provide the necessary assurance that the pressure boundary components are capable of performing safely under all operating conditions reviewed in the FSAR and described in the plant design specification. As a part of these examinations, all of the primary pressure boundary full penetration welds were volumetrically examined (radiographed) and the system will be subjected to hydrostatic pressure tests.

C. The intent of a preservice examination is to establish a reference or baseline before the initial operation of the facility. The results of subsequent inservice examination can then be compared with the original condition to determine if changes have occurred. If review of the inservice inspection results shows no change from the original condition, no action is required. In the case where baseline data are not available, all flaws must be treated as new flaws and evaluated accordingly. Section XI of the ASME Code contains acceptance standards that may be used as the basis for evaluating the acceptability of such flaws.

D. Other benefits of the preservice examination include providing redundant or alternative volumetric examination of the primary pressure boundary using a test method different from that employed during the component fabrication. Successful performance of preservice examination also demonstrates that the welds so examined are capable of subsequent inservice examination using a similar test method.

   In the case of Wolf Creek, a large portion of the preservice examination required by the ASME Code was performed. Failure to perform a 100% preservice examination of the welds identified below will not significantly affect the assurance of the initial structural integrity.

E. In some instances where the required preservice examinations were not performed to the full extent specified by the applicable ASME Code, the staff may require that these examinations or supplemental examinations be conducted as a part of the inservice inspection program. Requiring supplemental examinations to be performed at this time would result in hardships or unusual difficulties without a compensating increase in the level of quality or safety. The performance of supplemental examinations, such as surface examinations, in areas where volumetric inspection is difficult will be more meaningful after a period of operation. Acceptable preoperational integrity has already been established by similar ASME Code, Section III, fabrication examinations.

   In cases where parts of the required examination areas cannot be effectively examined because of a combination of component design or current examination technique limitations, the development of new or improved examination techniques will continue to be evaluated. As improvements in these areas are achieved, the staff will require that these new techniques be made a part of the inservice examination requirements for the components or welds that received a limited preservice examination.

Several of the preservice inspection relief requests involve limitations to the examination of the required volume of a specific weld. The inservice inspection (ISI) program is based on the examination of a representative sample of welds to detect generic degradation. In the event that the welds identified in the preservice inspection (PSI) relief requests are required to be examined again, the possibility of augmented ISI will be evaluated during review of the applicant's initial 10-year ISI program. An augmented program may include increasing the extent and/or frequency of inspection of accessible welds.

III. EVALUATION OF RELIEF REQUESTS

The applicant requested relief from specific PSI requirements in submittals dated September 6 and September 19, 1984. On the basis of the information submitted by the applicant and the review of the design, geometry, and materials of construction of the components, certain preservice requirements of the ASME Code, Section XI, have been determined to be impractical. Imposing these requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(2), conclusions that these preservice requirements are impractical are justified as follows. Unless otherwise stated, references to the Code refer to the ASME Code, Section XI, 1977 Edition, including addenda through Summer 1978.

A. Class 2, Examination Category C-F, Pressure Retaining Welds in Piping (8 Welds Total)

Code Requirement: Examination Category C-F requires a surface and volumetric examination of all pipe welds over 1/2-in. nominal wall thickness. A surface examination only is required for pipe welds with 1/2-in. or less nominal wall thickness.

Code Relief Request: Relief is requested from performing 100% of the Code-required volumetric examination on each of the subject welds.

Reason for Request: The design of Class 2 piping systems has welded joints, such as pipe-to-fitting and pipe-to-component joints, which physically obstruct all or part of the required Section XI examinations from the fitting or component side of the weld specified. The applicant has identified the piping system welds with geometric obstructions, identified the obstruction, and estimated the percentage of loss of volume coverage in Table J.1.

Table J.1 Identified obstructions in piping system welds

Component No.	Description	Basis for relief
Main feedwater system		
1-AE-05-FW302	14" valve to 14" pipe	Valve geometry and sockolet obstruction affects scan path; 5% loss of volume coverage.
1-AE-05-F020	14" valve to 14" pipe	
1-AE-04-F020	14" valve to 14" pipe	
1-AE-04-F005	14" valve to 14" pipe	
1-AE-04-F033	4" elbow to 4" valve	Valve geometry obstructs scan path loss of transducer contact on the elbow inner radius; 10% loss of volume coverage.
1-AE-05-F031	4" elbow to 4" valve	
1-AE-04-F031	4" elbow to 4" valve	
High-pressure coolant-injection system		
1-EM-02-S008-M	6" tee to 6" pipe	Tee geometry obstructs scan path. Limited scan because of lug obstruction on pipe side; 5% loss of volume coverage.

Staff Evaluation: A large portion (90 to 95%) of the preservice examination required by the ASME Code was performed. The staff has determined that the volumetric examination of the subject welds to the extent required by the Code is impractical because of the design of the piping systems. The applicant has conducted the preservice surface examinations on these welds. The staff therefore concludes that the large extent of Section XI ultrasonic examinations that were performed, the volumetric examinations performed during fabrication, and the hydrostatic test demonstrate an acceptable level of preservice structural integrity.

B. Class 2, Examination Category C-A, Pressure Retaining Welds in Steam Generators (5 Welds Total)

Code Requirement: Examination Category C-A requires volumetric examination of welds at gross structural discontinuities, head-to-shell welds, and tubesheet-to-shell welds. This examination includes essentially 100% of the weld length.

Code Relief Request: Relief is requested from performing 100% of the Code-required volume on each of the subject welds.

Reason for Request: The design of the steam generators has obstructions such as lugs, gauges, and nozzles that obstruct part of the Class 2 welds requiring Section XI examinations. The applicant has identified the welds with geometric obstructions, identified the obstruction, and estimated the percentage of loss of volume coverage in Table J.2.

Table J.2 Obstructions identified in steam generators

Component No.	Description	Basis for relief
Steam generator system		
1-EBB01A-SEAM-2-W	Tubesheet to stub barrel	Flange obstruction limiting scan length on tubesheet side. Three latches, instrumentation, nozzle, and I.D. plate obstructing scan path on stub barrel side; 5% loss of volume coverage at 60° and 45° scan angle.
1-EBB01A-SEAM-6-W	Transition cone to shell Section C	Four instrumentation nozzles, two lugs, gauges, and a feedwater nozzle obstructing scan path; 5% loss of volume coverage at 60° scan angle and 10% at 45° scan angle.
1-EBB01A-SEAM-8-W	Shell Section D to top head	Loss of transducer contact due to transition section, lug, and gauge obstructions; 10% loss of volume coverage.
1-EBB01A-SEAM-5-W	Shell Section B to transition cone	Loss of transducer contact because of transition section and two gauges; 10% loss of volume coverage at 60° scan angle and 5% loss of volume at 0° scan angle.

Table J.2 (Continued)

Component No.	Description	Basis for relief
1-EBB01A-SEAM-3-W	Stub barrel to shell Section A	Loss of transducer contact because of transition section and two gauges; 10% loss of volume coverage.

Staff Evaluation: The staff has determined that the volumetric examination of the subject welds to the extent required by the Code is impractical because of the design of the steam generators. A large portion (90 to 95%) of the preservice examination required by the ASME Code was performed. Failure to perform a 100% preservice examination of these welds will not significantly affect the assurance of the initial structural integrity. The staff therefore concludes that the limited Section XI ultrasonic examination, the volumetric examinations performed during fabrication, and the hydrostatic test demonstrate an acceptable level of preservice structural integrity.

C. Class 2, Examination Category C-F, Reactor Coolant Pump Seal Water Injection Line Welds (16 Welds With Pipe Diameter of 1.5 Inches or Less)

Code Requirements: Although the ASME Code, Section XI, does not require a volumetric examination of the reactor coolant pump seal water injection welds, the applicant committed to perform augmented volumetric examinations.

Relief Request: Relief is requested from performing the augmented volumetric examination on the subject welds.

Reason for Request: These 16 small-diameter (1.5 inches or less) pipe-to-component welds (4 welds each loop) could not receive a meaningful augmented volumetric examination due to a combination of the small pipe diameter and the minimum wall thickness. The applicant stated that these welds (Table J.3) will receive an alternative liquid penetrant surface examination.

Table J.3 Injection line welds

Component No.	Component weld description
Pump A seal water injection line welds	
1-BG-09-W686	2" x 1-1/2" reducer to 1-1/2" pipe
1-BG-09-FW881	1-1/2" pipe to valve
1-BG-09-FW882	Valve to 1-1/2" pipe
1-BG-09-W779	1-1/2" pipe to 2" x 1-1/2" reducer
Pump B seal water injection line welds	
1-BG-09-W814	2" x 1-1/2" reducer to 1-1/2" pipe
1-BG-09-FW875	1-1/2" pipe to valve
1-BG-09-FW876	Valve to 1-1/2" pipe
1-BG-09-W696	1-1/2" pipe to 2" x 1-1/2" reducer

Table J.3 (Continued)

Component No.	Component weld description
Pump C seal water injection line welds	
1-BG-09-W806	2" x 1-1/2" reducer to 1-1/2" pipe
1-BG-09-FW877	1-1/2" pipe to valve
1-BG-09-FW878	Valve to 1-1/2" pipe
1-BG-09-W807	1-1/2" pipe to 2" x 1-1/2" reducer
Pump D seal water injection line welds	
1-BG-09-W790	2" x 1-1/2" reducer to 1-1/2" pipe
1-BG-09-FW879	1-1/2" pipe to valve
1-BG-09-FW880	Valve to 1-1/2" pipe
1-BG-09-W859	1-1/2" pipe to 2" x 1-1/2" reducer

Staff Evaluation: This relief request is acceptable for PSI based on the following considerations:

(1) During fabrication, ASME Code, Section III, requires the subject welds to receive a radiographic examination of the entire weld volume.

(2) For PSI an alternative liquid penetrant surface examination was performed.

(3) The required ASME Code, Section III, examination along with the supplemental liquid penetrant examination for PSI demonstrate an acceptable level of preservice structural integrity. However, for ISI the applicant should evaluate and develop improved procedures which will allow these augmented examinations to be part of the ISI program.

D. Class 3, Examination Category D-A, Essential Service Water System Pump Supports (6 Supports) and Examination Category D-C, Fuel Pool Cooling and Cleanup Pipe Supports (4 Supports)

Essential Service Water Pump Supports

K-EF11-R005, K-EF11-R006, K-EF11-B003, K-EF11-B004, K-EF11-R001, K-EF11-R002

Fuel Pool Cooling and Cleanup Pipe Supports

1-EC-04-R026, 1-EC-04-R027, 1-EC-04-R029, 1-EC-04-R030

Code Requirement: Examination Categories D-A and D-C require a visual (VT-3) examination of the subject pump and pipe supports during preservice inspection.

Relief Request: Relief is requested from performing the Code-required preservice VT-3 inspection.

Reason for Request: The pump supports are inaccessible due to their submersion within the essential service water pump pit. The pipe supports will be submerged in the spent fuel pool during the life of the plant.

Staff Evaluation: In the submittal dated September 6, 1984, the applicant identified the specific component supports that are inaccessible for examination. Based on the review of this information, the staff has determined that the nondestructive examinations performed during fabrication exceed the Section XI required visual inspections and therefore are an acceptable alternative to the Code-required preservice VT-3 inspection. However, for ISI the applicant should incorporate remote visual inspection devices that will allow the Code-required VT-3 examinations to be part of the ISI program.

E. Class 1, Examination Category B-F, Pressurizer Dissimilar Metal Welds (6 Welds)

Code Requirement: Table IWB-2500-1, Examination Category B-F, requires a 100% volumetric and surface examination of the subject welds.

Code Relief Request: Relief is requested from performing 100% of the Code-required volumetric examination.

Reason for Request: Table J.4 describes the examination limitations that result from a combination of component geometry (restricting search unit movement) and metallurgical obstruction (caused by Inconel buttering inhibiting shear wave transmission).

Table J.4 Examination limitations

Weld No.	Description	Percentage Not Examined
1-TBB03-4-W	Relief nozzle to safe-end weld	20%-60° axial scan 45%-45° axial scan
1-TBB03-3-A-W	Safety nozzle to safe-end weld	50%-60° axial scan 35%-45° axial scan
1-TBB03-1-W	Surge nozzle to safe-end weld	15%-60° axial scan 40%-45° axial scan
1-TBB03-3-B-W	Safety nozzle to safe-end weld	55%-60° axial scan 40%-45° axial scan
1-TBB03-2-W	Spray nozzle to safe-end weld	10%-60° axial scan 40%-45° axial scan
1-TBB03-3-C-W	Safety nozzle to safe-end weld	20%-60° axial scan 40%-45° axial scan

Staff Evaluation: The subject welds are partially inaccessible and the material inhibits shear wave transmission as stated by the applicant. The staff concludes that the limited Section XI volumetric examination, the volumetric and surface examination performed during fabrication, and the hydrostatic test demonstrate an acceptable level of preservice structural integrity.

F. Class 1, Examination Category B-J, Branch Pipe Connection Welds (18 Welds Total)

Code Requirements: Table IWB-2500-1, Examination Category B-J, item B9.31 requires a surface and volumetric examination for branch connection piping welds, 2 in. nominal pipe size and greater.

Code Relief Request: Relief is requested from performing the required volumetric examination on the subject welds.

Reason for Request: Because of the materials of construction and the design and fabrication geometry of these corner types of branch connections (Table J.5), the applicant has concluded that meaningful examination by ultrasonic methods is not feasible and that no other practical volumetric method is available. As an alternative, VT-2 examinations for leakage will be conducted in accordance with IWA-5240 during the system leakage and hydrostatic pressure tests.

Table J.5 Branch pipe connection welds

Loop	Westinghouse Weld No.	Wolf Creek Weld No.
1	15	1BB-01-S102-3
	17	1BB-01-S105-5
	19	1BB-01-S101-5
	21	1BB-01-S101-8
	22	1BB-01-S101-9
2	15	1BB-01-S202-3
	17	1BB-01-S205-5
	19	1BB-01-S201-5
	21	1BB-01-S201-8
3	15	1BB-01-S302-3
	17	1BB-01-S305-5
	21	1BB-01-S305-6
	20	1BB-01-S301-5
4	15	1BB-01-S402-3
	16	1BB-01-S402-4
	18	1BB-01-S405-5
	20	1BB-01-S401-5
	22	1BB-01-S401-6

Staff Evaluation: This relief request is acceptable for preservice inspection (PSI) based on the following considerations:

(1) During fabrication, the subject welds have received liquid penetrant examinations and radiographic examination of the entire weld volume in accordance with ASME Code, Section III, requirements.

(2) For PSI, an alternative VT-2 examination for leakage will be conducted during the system hydrostatic test and these welds will also receive the required surface examination.

(3) The combination of required surface examination, visual examination for leakage, and the Code-required fabrication examinations demonstrate an acceptable level of preservice structural integrity.

G. Class 1, Examination Categories B-A and B-D, Reactor Pressure Vessel Welds (11 Welds Total)

<u>Flange-to-Vessel Weld</u>

1-RV-101-121

<u>Lower-Head-to-Dollar-Plate Weld</u>

1-RV-102-151

<u>Lower-Head-to-Shell Weld</u>

1-RV-101-141

<u>Meridional Welds in Lower Head</u>

1-RV-101-154A, 1-RV-101-154B, 1-RV-101-154C, 1-RV-101-154D

<u>Outlet-Nozzle-to-Vessel Welds</u>

1-RV-107-121-A, 1-RV-107-121-B, 1-RV-107-121-C, 1-RV-107-121-D

<u>Code Requirement</u>: Table IWB-2500-1, Examination Categories B-A and B-D, require a 100% volumetric examination of the subject welds.

<u>Code Relief Request</u>: Relief is requested from performing 100% of the Code-required volumetric examination.

<u>Reason for Request</u>:

1-RV-101-121 On the flange-to-vessel weld, the parallel scan portion of examination can only be done from the lower side because a flange taper was present above the weld. A complete perpendicular scan was done from the flange mating surface. Therefore, approximately 25% of the required weld volume was not examined.

1-RV-102-151 Approximately 10% of the required weld volume on the lower-head to-dollar-plate weld was not examined because of obstructions presented by the instrumentation nozzles.

1-RV-101-141 The perpendicular examination (shooting down) for the 60° scan angle ultrasonic shear wave examination not performed because of outside diameter surface taper geometry limiting ultrasonic head contact on the lower-head-to-shell weld. A 45° longitudinal wave examination was performed in lieu of the 60° shear wave examination. Approximately 10% of the required weld volume was not examined.

Weld No.	% Not Examined
1-RV-101-154A	22*
1-RV-101-154B	27*
1-RV-101-154C	12*
1-RV-101-154D	14*
1-RV-107-121-A	10**
1-RV-107-121-B	10**
1-RV-107-121-C	10**
1-RV-107-121-D	10**

*O.D. surface condition limits ultrasonic head contact.

**Approximately 10% of the total weld volume for each outlet-nozzle-to-vessel weld is obstructed by contact between the examination head and the nozzle knuckle extending from the nozzle opening through the plane of the reactor pressure vessel inner diameter.

Staff Evaluation: The subject welds are partially inaccessible for examination because of the existing design. The staff concludes that the limited Section XI volumetric examination, the volumetric and surface examination performed during fabrication, and the hydrostatic test demonstrate an acceptable level of preservice structural integrity.

IV. CONCLUSIONS

On the basis of the foregoing, pursuant to 10 CFR 50.55a(a)(2), certain Code-required Section XI preservice examinations are impractical, and compliance with the requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety.

The staff technical evaluation has not identified any practical method by which the existing Wolf Creek Generating Station can meet all the specific preservice inspection requirements of Section XI of the ASME Code. Requiring exact compliance with all the required Section XI inspections would delay the full-power operation of the plant in order to redesign a significant number of plant systems, obtain sufficient replacement components, install the new components, and repeat the preservice examination of these components. Examples of components that would require redesign to meet the specific preservice examination provisions are the reactor vessel and a significant number of the piping and component support systems. Even after the redesign effort, complete compliance with the preservice examination requirements probably could not be achieved. However, the as-built structural integrity of the existing primary pressure boundary has already been established by the construction code fabrication examinations.

On the basis of staff review and evaluation, it is concluded that the public interest is not served by imposing certain provisions of Section XI of the ASME Code that have been determined to be impractical. Pursuant to 10 CFR 50.55a(a)(2), relief is allowed from these requirements which are impractical to implement and would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

APPENDIX K

TECHNICAL EVALUATION REPORT
ON CONTROL OF HEAVY LOADS,
PHASE I

ABSTRACT

The Nuclear Regulatory Commission (NRC) has requested that all nuclear plants, either operating or under construction, submit a response of consistency with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." EG&G Idaho, Inc., has contracted with the NRC to evaluate the responses of those plants presently under construction. This report contains EG&G's evaluation and recommendations for Wolf Creek Plant.

EXECUTIVE SUMMARY

Based on the information provided, EG&G Idaho concludes that the Wolf Creek Plant is consistent with the intent of NUREG-0612.

CONTENTS

TABLES

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS

WOLF CREEK PLANT

(PHASE I)

1. INTRODUCTION

1.1 Purpose of Review

This technical evaluation report documents the EG&G Idaho, Inc., review of general load-handling policy and procedures at Wolf Creek Plant. This evaluation was performed with the objective of assessing conformance to the general load-handling guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" [1], Section 5.1.1.

1.2 Generic Background

Generic Technical Activity Task A-36 was established by the U.S. Nuclear Regulatory Commission (NRC) staff to systematically examine staff licensing criteria and the adequacy of measures in effect at operating nuclear power plants to assure the safe handling of heavy loads and to recommend necessary changes to these measures. This activity was initiated by a letter issued by the NRC staff on May 17, 1978 [2], to all power reactor applicants, requesting information concerning the control of heavy loads near spent fuel.

The results of Task A-36 were reported in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The staff's conclusion from this evaluation was that existing measures to control the handling of heavy loads at operating plants, although providing protection from certain potential problems, do not adequately cover the major causes of load-handling accidents and should be upgraded.

In order to upgrade measures for the control of heavy loads, the staff developed a series of guidelines designed to achieve a two-phase objective using an accepted approach or protection philosophy. The first portion of the objective, achieved through a set of general guidelines identified in NUREG-0612, Article 5.1.1, is to ensure that all load-handling systems at nuclear power plants are designed and operated such that their probability of failure is uniformly small and appropriate for the critical tasks in which they are employed. The second portion of the staff's objective, achieved through guidelines identified in NUREG-0612, Articles 5.1.2 through 5.1.5, is to ensure that, for load-handling systems in areas where their failure might result in significant consequences, either (a) features are provided, in addition to those required for all load-handling systems, to ensure that the potential for a load drop is extremely small (e.g., a single-failure-proof crane) or (b) conservative evaluations of load-handling accidents indicate that the potential consequences of any load drop are acceptably small. Acceptability of accident consequences is quantified in NUREG-0612 into four accident analysis evaluation criteria.

The approach used to develop the staff guidelines for minimizing the potential for a load drop was based on defense in depth and is summarized as follows:

- Provide sufficient operator training, handling system design, load-handling instructions, and equipment inspection to assure reliable operation of the handling system

- Define safe load travel paths through procedures and operator training so that, to the extent practical, heavy loads are not carried over or near irradiated fuel or safe shutdown equipment

- Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

Staff guidelines resulting from the foregoing are tabulated in Section 5 of NUREG-0612.

1.3 Plant-Specific Background

On December 22, 1980, the NRC issued a letter [3] to Kansas Gas and Electric, the applicant for Wolf Creek Plant Unit 1 requesting that the applicant review provisions for handling and control of heavy loads at Wolf Creek Plant Unit 1, evaluate these provisions with respect to the guidelines of NUREG-0612, and provide certain additional information to be used for an independent determination of conformance to these guidelines. On June 22, 1981, the Kansas Gas and Electric provided the initial response [4] to this request. Based on this information, a preliminary draft of this report was prepared and discussed with the applicant. As a result of this discussion, the applicant conducted additional internal review and provided additional information in Reference [9].

2. EVALUATION AND RECOMMENDATIONS

2.1 Overview

The following sections summarize Kansas Gas and Electric's review of heavy load handling at Wolf Creek Plant accompanied by EG&G's evaluation, conclusions, and recommendations to the applicant for making the facilities more consistent with the intent of NUREG-0612. Standardized Nuclear Unit Power Plant System's review of the facilities does not differentiate between Calloway Plant Unit 1 and Wolf Creek Plant so it is assumed that both units are of identical design. The applicant has indicated the weight of a heavy load for this facility (as defined in NUREG-0612, Article 1.2) as 2000 lbs.

2.2 Heavy Load Overhead Handling Systems

This section reviews the applicant's list of overhead handling systems which are subject to the criteria of NUREG-0612 and a review of the justification for excluding overhead handling systems from the above-mentioned list.

2.2.1 Scope

"Report the results of your review of plant arrangements to identify all overhead handling systems from which a load drop may result in damage to any system required for plant shutdown or decay heat removal (taking no credit for any interlocks, technical specifications, operating procedures, or detailed structural analysis) and justify the exclusion of any overhead handling system from your list by verifying that there is sufficient physical separation from any load-impact point and any safety-related component to permit a determination by inspection that no heavy load drop can result in damage to any system or component required for plant shutdown or decay heat removal."

A. Summary of Applicant's Statements

The applicant's review of overhead handling systems identified the cranes and hoists shown in Table 2.1 as those which handle heavy loads in the vicinity of irradiated fuel or safe shutdown equipment.

The applicant has also identified numerous other cranes that have been excluded from satisfying the criteria of the general guidelines of NUREG-0612. These are shown in Table 2.2. These cranes were excluded based on sufficient physical separation from any load impact point that could damage any system or component required for plant shutdown or decay heat removal.

B. EG&G Evaluation

The applicant's response indicates that the overhead handling devices at the Wolf Creek Plant are listed in Table 2.1 and Table 2.2. Figures 1 through 19 of Reference 4 identify the locations of all applicable overhead handling systems in the plant and their proximity to safety-related components. EG&G concludes that the applicant's list of cranes and hoists in the aforementioned tables is complete and satisfies the requirements of NUREG-0612.

The applicant reviewed the Wolf Creek Plant arrangement and indicated that based on physical separation, no load drops in the radwaste or turbine buildings could result in damage to any system or component required for safe shutdown or decay heat removal. Therefore, the overhead handling devices in these buildings were excluded from further concern. EG&G agrees with the applicant's evaluation of these devices, and the justification for exclusion of these systems from NUREG-0612.

TABLE 2.1 NON-EXEMPT HEAVY LOAD HANDLING SYSTEMS
WOLF CREEK PLANT (FUEL HANDLING CRANE DATA[1])

Parameters	Polar Crane	Cask Handling Crane	Spent Fuel Pool Bridge Crane	Refueling Machine
	Name of Crane			
Capacity of main hoist	260 tons	150 tons	2 tons	2.4 tons
Capacity of auxiliary monorail hoist (const)	25 tons	5 tons		
Capacity of auxiliary monorail hoist (normal)	25 tons	5 tons and 2 tons[2]		
Capacity of main trolley	260 tons	130 tons	2 tons	1.5 tons
Capacity of lift beam	500 tons			2.4 tons
Maximum main hoist speed (normal)	5 fpm	3.75 fpm	21 fpm	20 fpm
Minimum main hoist speed (normal)	3 fpm	2 fpm	7 fpm	
Maximum auxiliary monorail hoist speed (normal)	40 fpm			
Minimum auxiliary monorail hoist speed (normal)	3 fpm			
Maximum trolley speed (normal)	51.5 fpm	20 fpm	30 fpm	20 fpm
Minimum trolley speed (normal)	6 fpm	6 fpm	10 fpm	
Maximum bridge speed (normal)	51.5 fpm	20 fpm	30 fpm	40 fpm
Minimum bridge speed (normal)	6 fpm	6 fpm	10 fpm	
Maximum load during plant operation	167.5 tons	125 tons	1,870 lbs	
Normal expected load range	0-167.5 tons	0-125 tons	0-1,870 lbs	
Maximum construction load	475 tons			
Maximum main hoist speed (constr)	5 fpm			
Minimum main hoist speed (constr)	3 fpm			
Maximum trolley speed (constr)	51.5 fpm			
Minimum trolley speed (constr)	6 fpm			
Maximum bridge speed (constr)	51.5 fpm			
Minimum bridge speed (constr)	6 fpm			
Normal load range (constr)	0-475 tons			

TABLE 2.1 (continued)

Parameters	Name of Crane: Polar Crane	Cask Handling Crane	Spent Fuel Pool Bridge Crane	Refueling Machine
Maximum monorail hoist speed		20 fpm		22 fpm
Minimum monorail hoist speed		10 fpm		7 fpm
Maximum monorail trolley speed		32 fpm		
Minimum monorail trolley speed		16 fpm		
Lifting limitation	28.5 ft (above vessel flange)	Cask bottom 3 inches above floor El. 2047'-6"	24'-3" (hook limit is 2066'-8")	
Seismic Class	(3)	(3)	(3)	(3)
Design Standards General	CMAA No. 70 (1975)	CMAA No. 70 (1975)	CMAA No. 70 (1975)	CMAA No. 70 (1975)
Structural	Covered by CMAA	Covered by CMAA	Covered by CMAA	ASME Sect. III, App. XVII, Subarticle XVII-2200
Electrical	NFPA Vol. 5 Art. 610 1974-1975	NFPA Vol. 5 Art. 610 1974-1975	NFPA Vol. 5 Art. 610 1974-1975	NFPA Vol. 5 Art. 610 1974-1975
Materials	ASTM Std's.	ASTM Std's.	ASTM Std's.	ASTM Std's.
Others	OSHA 29 CFR 1910 & 1926	OSHA 29 CFR 1910 & 1926	OSHA 29 CFR 1910 & 1926	OSHA 29 CFR 1910 & 1926

NOTES:

(1) Rated speeds given are within ± 10 percent of the actual speeds.

(2) Refer to Figure 23, Reference 4. A 2-ton limit to the monorail hoist exists only over area B on Figure 23.

(3) Seismic Category I.

TABLE 2.2 EXEMPT HEAVY LOAD HANDLING SYSTEMS--WOLF CREEK PLANT

Equipment Number	Service Description	Hoist Capacity	Primary Loads Lifted
HKF03A-D	Containment Jib Crane	3 tons	Hydrogen mixing fans
HKF05	Secondary Shield Wall Area Jib Crane	3 tons	Cooling fan
HKF06	Hot Machine Shop Bridge Crane	3 tons	Chemical tanks
HKF08A & B	Diesel Generator Underhung Monorail and Bridge Crane	5 tons	Emergency fuel oil day tank and miscellaneous equipment
HKF09A & B	Fuel Pool Cooling Pump Monorail and Hoist	2 tons	Fuel pool cooling heat transfer
HKF10	Auxiliary Building Filter Room Monorail and Hoist	5 tons	Reactor coolant filters
HKF11A-C	Auxiliary Feedwater Pump Monorail and Hoist	4 tons	Auxiliary feedwater pumps
HKF12A-C	Component Cooling Water Pump Monorail and Hoist	5 tons	Component cooling water heat exchangers
HKF13	Component Cooling Water Surge Tank Area Monorail and Hoist	10 tons	Component cooling water surge tanks
HKF15A & B	Centrifugal Charging Pump Monorail and Hoist	5 tons	Centrifugal charging pumps
HKF16A & B	Safety Injection Pump Monorail and Hoist	5 tons	Safety injection pumps
HKF17A & B	RHR Pump Monorail and Hoist	5 tons	RHR pumps
HKF18A & B	Containment Spray Pump Monorail and Hoist	5 tons	Containment spray pumps
HKF19	Reciprocating Charging Pump Monorail and Hoist	5 tons	Positive displacement charging pumps
HKF23	Auxiliary Building HVAC Monorail and Hoist	10 tons	Miscellaneous fans and equipment

TABLE 2.2 (continued)

Equipment Number	Service Description	Hoist Capacity	Primary Loads Lifted
HKF24	Moderation Heat Exchanger Monorail and Hoist	2 tons	Miscellaneous heat exchangers
HKF29A-D	Main Steam Relief Isolation Valve Monorail and Hoist	10 tons	Main steam relief isolation valve and feedwater piping
HKF30	Resin Charging Tank Area Monorail and Hoist	1 ton	Boron resin thermal regenerative filter
HKF32	Communication Corridor Hot Water Packaging Area Monorail, Hoist and Switch	5 tons	Miscellaneous equipment
HKF33	Boric Acid Batch Tank Monorail and Hoist	1 ton	Boric acid tanks
HKF41	ESW Pump House Hoist	10 tons	ESW pumps

C. EG&G Conclusions and Recommendations

Based on the information provided, EG&G concludes that the applicant has included all applicable hoists and cranes in their list of handling systems which must comply with the requirements of the general guidelines of NUREG-0612.

2.3 General Guidelines

This section addresses the extent to which the applicable handling systems comply with the general guidelines of NUREG-0612, Article 5.1.1. EG&G's conclusions and recommendations are provided in summaries for each guideline.

The NRC has established seven general guidelines which must be met in order to provide the defense-in-depth approach for the handling of heavy loads. These guidelines consist of the following criteria from Section 5.1.1 of NUREG-0612:

- Guideline 1--Safe Load Paths
- Guideline 2--Load-Handling Procedures
- Guideline 3--Crane Operator Training
- Guideline 4--Special Lifting Devices
- Guideline 5--Lifting Devices (Not Specially Designed)
- Guideline 6--Cranes (Inspection, Testing, and Maintenance)
- Guideline 7--Crane Design.

These seven guidelines should be satisfied for all overhead handling systems and programs in order to handle heavy loads in the vicinity of the reactor vessel, near spent fuel in the spent-fuel pool, or in other areas where a load drop may damage safe shutdown systems. The succeeding paragraphs address the guidelines individually.

2.3.1 Safe Load Paths [Guideline 1, NUREG-0612, Article 5.1.1(1)]

> "Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent-fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practical, structural floor members, beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. These load paths should be defined in procedures, shown on equipment layout drawings, and clearly marked on the floor in the area where the load is to be handled. Deviations from defined load paths should require written alternative procedures approved by the plant safety review committee."

A. Summary of Applicant's Statements

The applicant presented a discussion of the cranes of Table 2.1 which demonstrated the adequacy of the design, and administrative measures to ensure that load-handling operations remain within safe load paths. Figures 1 through 11 [4] show equipment configurations and areas of movements of cranes in the Reactor and Auxiliary Buildings.

Figures 20 through 25 of Reference 4 show crane configurations of the Polar Crane, Fuel Building, and Spent-Fuel Bridge Crane.

The applicant discusses compliance with NUREG-0612 criteria for safe load paths in the Fuel Building in Sections 2.2 and 2.3 of Reference 4. Safe load path considerations in the Auxiliary Building are limited since load movements are limited by monorails associated with overhead lifting devices. Sections 2.3 and 2.4.1 discuss load movements and load paths in the Reactor Building.

Load paths for anticipated load movements have been completed [10] for the Polar Crane. The safe load paths which take advantage of structural members has been delineated on plant layout drawings. Any deviation from established load paths will require approval of the plant safety review committee, except in emergencies at which time the crane signal man will be authorized to take immediate action to ensure plant or personnel safety.

B. EG&G Evaluation

The applicant has indicated that safe load paths will be defined in areas where load drops could affect safe shutdowns.

Safe load paths for the polar crane have been delineated on plant layout drawings. Should deviation from an established load path become necessary, the plant safety review committee will approve the new path.

C. EG&G Conclusions and Recommendations

Based on the information presented, EG&G concludes that the applicant is consistent with the intent of this guideline.

2.3.2 Load-Handling Procedures [Guideline 2, NUREG-0612, Article 5.1.1(2)]

"Procedures should be developed to cover load-handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. At a minimum, procedures should cover handling of those loads listed in Table 3-1 of NUREG-0612. These procedures should include: identification of required equipment; inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; defining the safe path; and other special precautions."

A. Summary of Applicant's Statements

The applicant responded to this guideline by describing the cranes of Table 2.1 and defining various devices that are used to ensure safe handling of fuel assemblies.

It was also stated that certain heavy loads such as the spent fuel cask are being designed and the loads evaluated. Written procedures that reflect the results of this evaluation will be developed to govern the handling of all heavy loads that could damage fuel or safe shutdown equipment. These procedures will incorporate requirements provided in Section 5.1.1(2) of NUREG-0612.[10]

B. EG&G Evaluation

The applicant has indicated that written procedures will be developed for heavy loads that could damage fuel or safe shutdown equipment. The procedures will be developed with the requirements of this guideline.

C. EG&G Conclusions and Recommendations

Based on the information presented, EG&G concludes that the applicant is consistent with the intent of this guideline.

2.3.3 Crane Operator Training [Guideline 3, NUREG-0612, Article 5.1.1(3)]

"Crane operators should be trained, qualified, and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976, 'Overhead and Gantry Cranes' [6]."

A. Summary of Applicant's Statements

The applicant stated in the response that specific plant procedures will be developed that address operator training, qualification and conduct for those cranes identified in Table 2.1. These procedures will incorporate the guidance provided by ANSI B30.2-1976 Chapter 2-3.

B. EG&G Evaluation

With the applicant developing procedures that will comply with the criteria of Guidelines 3, EG&G considers the guideline to be satisfied.

C. EG&G Conclusions and Recommendations

The applicant is consistent with the intent of Guideline 3.

2.3.4 Special Lifting Devices [Guideline 4, NUREG-0612, Article 5.1.1(4)]

"Special lifting devices should satisfy the guidelines of ANSI N14.6-1978, 'Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials' [7]. This standard should apply to all special lifting devices which carry heavy loads in areas as defined above. For operating plants, certain inspections and load tests may be accepted in lieu of certain material requirements in the standard. In addition, the stress design factor stated in Section 3.2.1.1 of ANSI N14.6 should be based on the combined maximum static and dynamic loads that could be imparted on the handling device based on characteristics of the crane which will be used. This is in lieu of the guideline in Section 3.2.1.1 of ANSI N14.6 which bases the stress design factor on only the weight (static load) or the load and of the intervening components of the special handling device."

A. Summary of Applicant's Statements

The applicant states that the lifting devices for the reactor vessel head, the reactor vessel internals, and the coolant pump motor have been designed. In addition, the supplier has performed analyses with regard to consistency with ANSI N14.6-1978. These results are documented in WCAP 10164.

Certain heavy loads, such as the spent-fuel cask, have not been specifically designed for KG&E at this time. However, load values that envelop the expected weight of these items are included in the evaluation. The spent-fuel cask will be designed in accordance with ANSI N14.6-1978.

B. <u>EG&G Evaluation</u>

The applicant has indicated that the special lifting devices design will be consistent with the requirements of N14.6-1978.

C. <u>EG&G Conclusions and Recommendations</u>

Based on the information provided, EG&G considers the applicant to be consistent with the intent of this guideline.

2.3.5 <u>Lifting Devices (Not Specially Designed) [Guideline 5, NUREG-0612, Article 5.1.1(5)]</u>

"Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9-1971, 'Slings' [8]. However, in selecting the proper sling, the load used should be the sum of the static and maximum dynamic load. The rating identified on the sling should be in terms of the 'static load' which produces the maximum static and dynamic load. Where this restricts slings to use on only certain cranes, the slings should be clearly marked as to the cranes with which they may be used."

A. Summary of Applicant's Statements

The applicant has stated that lifting devices for the miscellaneous hoists and cranes will meet the guidance provided by ANSI B30.9-1971 as clarified in NUREG-0612 Section 5.1.1(5).

B. EG&G Evaluation

EG&G considers that the applicant has met the criteria of this guideline. Slings will be provided in accordance with guidelines of ANSI B30.9-1971 "Slings."

C. EG&G Conclusions and Recommendations

The applicant has indicated intention to provide slings in accordance with ANSI B30.9-1971 "Slings," and, therefore, the intent of the requirements of this guideline have been satisfied.

2.3.6 Cranes (Inspection, Testing, and Maintenance) [Guideline 6, NUREG-0612, Article 5.1.1(6)]

"The crane should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' with the exception that tests and inspections should be performed prior to use where it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where frequency of crane use is less than the specified inspection and test frequency (e.g., the polar crane inside a PWR containment may only be used every 12 to 18 months during refueling operations, and is generally not accessible during power operation. ANSI B30.2, however, calls for certain inspections to be performed daily or monthly. For such cranes having limited usage, the inspections, test, and maintenance should be performed prior to their use)."

A. Summary of Applicant's Statements

The applicant's response states that procedures will be developed for inspection, testing, and maintenance of those cranes defined in Table 2.1. These procedures will include the guidance provided by Chapter 2.2 of ANSI B30.2-1976 as clarified in NUREG-0612 paragraph 5.1.1(6) with regard to frequency of inspections, tests, and maintenance.

B. EG&G Evaluation

EG&G considers the applicant to have met the intent of Guideline 6.

C. EG&G Conclusions and Recommendations

The applicant is consistent with the intent of Guideline 6 of NUREG-0612.

2.3.7 Crane Design [Guideline 7, NUREG-0612, Article 5.1.1(7)]

"The crane should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' and of CMAA-70, 'Specifications for Electric Overhead Traveling Cranes' [9]. An alternative to a specification in ANSI B30.2 or CMAA-70 may be accepted in lieu of specific compliance if the intent of the specification is satisfied."

A. Summary of Applicant's Statements

The applicant has stated that the cranes of Table 2.1 are designed to the standards of CMAA-70 (1975). The Wolf Creek cranes were ordered in 1974, and their purchase specifications included reference to ANSI B30.2-1967 which was the applicable edition for design requirements.

B. EG&G Evaluation

The applicant's cranes (Table 2.1) were designed to meet the applicable criteria and guidelines for CMAA-70 (1975) and ANSI B30.2-1967. EG&G concludes that Guideline 7 of NUREG-0612 has been satisfied.

C. EG&G Conclusions and Recommendations

The applicant is consistent with the intent of Guideline 7 of NUREG-0612.

2.4 Interim Protection Measures

The NRC staff has established (NUREG-0612, Article 5.3) that six measures should be initiated to provide reasonable assurance that handling of heavy loads will be performed in a safe manner until final implementation of the general guidelines of NUREG-0612, Article 5.1, is complete. Four of these six interim measures consist of general Guideline 1, Safe Load Paths; Guideline 2, Load-Handling Procedures; Guideline 3, Crane Operator Training; and Guideline 6, Cranes (Inspection, Testing, and Maintenance). The two remaining interim measures cover the following criteria:

- Heavy load technical specifications
- Special review for heavy loads handled over the core.

Applicant implementation and evaluation of these interim protection measures is contained in the succeeding paragraphs of this section.

2.4.1 Interim Protection Measure 1--Technical Specifications

"Licenses for all operating reactors not having a single-failure-proof overhead crane in the fuel storage pool area should be revised to include a specification comparable to Standard

Technical Specification 3.9.7, 'Crane Travel - Spent Fuel Storage Pool Building,' for PWRs and Standard Technical Specification 3.9.6.2, 'Crane Travel,' for BWRs, to prohibit handling of heavy loads over fuel in the storage pool until implementation of measures which satisfy the guidelines of Section 5.1."

A. Summary of Applicant's Statements

Interim measures are not applicable. Plant is not in operation.

B. EG&G Evaluation

Interim measures are not applicable. Plant is not in operation.

C. EG&G Conclusions and Recommendations

Interim measures are not applicable. Plant is not operational.

2.4.2 Interim Protection Measures 2, 3, 4, and 5 - Administrative Controls

"Procedural or administrative measures [including safe load paths, load-handling procedures, crane operator training, and crane inspection]... can be accomplished in a short time period and need not be delayed for completion of evaluations and modifications to satisfy the guidelines of Section 5.1 of [NUREG-0612]."

A. Summary of Applicant's Statements

Summaries of applicant's statements are contained in discussions of the respective general guidelines in Sections 2.3.1, 2.3.2, 2.3.3, and 2.3.6, respectively.

B. EG&G Evaluations, Conclusions, and Recommendations

EG&G evaluations, conclusions, and recommendations are contained in discussions of the respective general guidelines in Sections 2.3.1, 2.3.2, 2.3.3, and 2.3.6.

2.4.3 Interim Protection Measure 6--Special Review for Heavy Loads Over the Core

"Special attention should be given to procedures, equipment, and personnel for the handling of heavy loads over the core, such as vessel internals or vessel inspection tools. This special review should include the following for these loads: (a) review of procedures for installation of rigging or lifting devices and movement of the load to assure that sufficient detail is provided and that instructions are clear and concise; (b) visual inspections of load-bearing components of cranes, slings, and special lifting devices to identify flaws or deficiencies that could lead to failure of the component; (c) appropriate repair and replacement of defective components; and (d) verify that the crane operators have been properly trained and are familiar with specific procedures used in handling these loads, e.g., hand signals, conduct of operations, and content of procedures."

A. Summary of Applicant's Statements

Interim measures are not applicable. Plant is not operational.

B. EG&G Evaluation

Interim measures are not applicable. Plant is not operational.

C. EG&G Conclusion

Interim measures are not applicable. Plant is not operational.

3. CONCLUDING SUMMARY

3.1 Applicable Load-Handling Systems

The list of cranes and hoists supplied by the applicant as being subject to the provisions of NUREG-0612 is apparently complete (see Section 2.2.1).

3.2 Guideline Recommendations

The applicant is consistent with the intent of the seven NRC guidelines for heavy load handling (Section 2.3) at Wolf Creek Plant. This conclusion is represented in tabular form as Table 3.1.

Guideline	Recommendation
1. Section 2.3.1	a. In compliance.
2. Section 2.3.2	a. In compliance.
3. Section 2.3.3	a. In compliance.
4. Section 2.3.4	a. In compliance.
5. Section 2.3.5	a. In compliance.
6. Section 2.3.6	a. In compliance.
7. Section 2.3.7	a. In compliance.

TABLE 3.1. WOLF CREEK PLANT COMPLIANCE MATRIX

Equipment Designation	Heavy Loads	Weight or Capacity (tons)	Guideline 1 Safe Load Paths	Guideline 2 Procedures	Guideline 3 Crane Operator Training	Guideline 4 Special Lifting Devices	Guideline 5 Slings	Guideline 6 Crane-Test and Inspection	Guideline 7 Design
Polar Crane	C	260/25	C	C	C	C	C	C	C
Cask Handling Crane	C	150/5	C	C	C	C	C	C	C
Spent Fuel Pool Bridge Crane	C	2	C	C	C	C	C	C	C
Refueling Machine	C	2.4	C	C	C	C	C	C	C

C = Applicant action complies with NUREG-0612 Guidelines.

NC = Applicant action does not comply with NUREG-0612 Guidelines.

3.3 Interim Protection

EG&G's evaluation of information provided by the applicant indicates that the following actions are necessary to ensure that the six NRC staff measures for interim protection at Wolf Creek Plant are met:

Interim Measure	Recommendation
None needed	Applicant has satisfied all guidelines.

3.4 Summary

The applicant is considered to be in compliance with the intent of all seven guidelines.

4. REFERENCES

1. NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, NRC.

2. V. Stello, Jr. (NRC), Letter to all applicants. Subject: Request for Additional Information on Control of Heavy Loads Near Spent Fuel, NRC, 17 May 1978.

3. USNRC, Letter to Kansas Gas and Electric. Subject: NRC Request for Additional Information on Control of Heavy Loads Near Spent Fuel, NRC, 22 December 1980.

4. N. A. Petrick, Standardized Nuclear Unit Power Plant System Letter to H. R. Denton (NRC) Subject: Response to Staff Position Interim Actions for Control of Heavy Loads dated June 22, 1981.

5. N. A. Petrick, Standardized Nuclear Unit Power Plant System Letter to H. R. Denton (NRC) Subject: Control of Heavy Loads, dated 8/4/82.

6. ANSI B30.2-1976, "Overhead and Gantry Cranes."

7. ANSI N14.6-1978, "Standard for Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or more for Nuclear Materials."

8. ANSI B30.9-1971, "Slings."

9. CMAA-70, "Specifications for Electric Overhead Traveling Cranes."

10. N. A. Petrick, Standardrized Nuclear Unit Power Plant System, Letter to H. R. Denton (NRC) Subject: Control of Heavy Loads dated January 27, 1984.

APPENDIX L

TECHNICAL EVALUATION REPORT ON CONTROL OF HEAVY LOADS, PHASE II

ABSTRACT

The Nuclear Regulatory Commission (NRC) has requested that all nuclear plants, either operating or under construction, submit a response of consistency with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." EG&G Idaho, Inc., has contracted with the NRC to evaluate the responses of those plants presently under construction. This report contains EG&G's evaluation and recommendations for Wolf Creek Plant for the requirements of Sections 5.1.2, 5.1.3, 5.1.5, and 5.1.6 of NUREG-0612 (Phase II). Section 5.1.1 (Phase I) was covered in a separate report [1].

EXECUTIVE SUMMARY

- Based on the information provided EG&G Idaho concludes that Wolf Creek Plant is in compliance with the intent of the requirements of NUREG 0612.

CONTENTS

TABLES

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS

WOLF CREEK PLANT

(PHASE II)

1. INTRODUCTION

1.1 Purpose of Review

This technical evaluation report documents the EG&G Idaho, Inc., review of general load-handling policy and procedures at Wolf Creek Plant. This evaluation was performed with the objective of assessing conformance to the general load-handling guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" [2], Sections 5.1.2, 5.1.3, 5.1.5, and 5.1.6. This constitutes Phase II of a two-phase evaluation. Phase I assesses conformance to Section 5.1.1 of NUREG-0612 and was documented in a separate report [1].

1.2 Generic Background

Generic Technical Activity Task A-36 was established by the U.S. Nuclear Regulatory Commission (NRC) staff to systematically examine staff licensing criteria and the adequacy of measures in effect at operating nuclear power plants to assure the safe handling of heavy loads and to recommend necessary changes to these measures. This activity was initiated by a letter issued by the NRC staff on May 17, 1978 [3], to all power reactor applicants, requesting information concerning the control of heavy loads near spent fuel.

The results of Task A-36 were reported in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The staff's conclusion from this evaluation was that existing measures to control the handling of heavy loads at operating plants, although providing protection from certain potential problems, do not adequately cover the major causes of load-handling accidents and should be upgraded.

In order to upgrade measures for the control of heavy loads, the staff developed a series of guidelines designed to achieve a two-phase objective using an accepted approach or protection philosophy. The first portion of the objective, achieved through a set of general guidelines identified in NUREG-0612, Article 5.1.1, is to ensure that all load-handling systems at nuclear power plants are designed and operated such that their probability of failure is uniformly small and appropriate for the critical tasks in which they are employed. The second portion of the staff's objective, achieved through guidelines identified in NUREG-0612, Articles 5.1.2 through 5.1.5, is to ensure that, for load-handling systems in areas where their failure might result in significant consequences, either (a) features are provided, in addition to those required for all load-handling systems, to ensure that the potential for a load drop is extremely small (e.g., a single-failure-proof crane) or (b) conservative evaluations of load-handling accidents indicate that the potential consequences of any load drop are acceptably small. Acceptability of accident consequences is quantified in NUREG-0612 into four accident analysis evaluation criteria as follows:

- "Releases of radioactive material that may result from damage to spent fuel based on calculations involving accidental dropping of a postulated heavy load produce doses that are well within 10 CFR Part 100 limits of 300 rem thyroid, 25 rem whole body (analyses should show that doses are equal to or less than 1/4 of Part 100 limits);

- "Damage to fuel and fuel storage racks based on calculations involving accidental dropping of a postulated heavy load does not result in a configuration of the fuel such that k_{eff} is larger than 0.95;

- "Damage to the reactor vessel or the spent-fuel pool based on calculations of damage following accidental dropping of a postulated heavy load is limited so as not to result in water leakage that could uncover the fuel, (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost is borated); and

- "Damage to equipment in redundant or dual safe shutdown paths, based on calculations assuming the accidental dropping of a postulated heavy load, will be limited so as not to result in loss of required safe shutdown functions."

The approach used to develop the staff guidelines for minimizing the potential for a load drop was based on defense in depth. This plan includes proper operator training, equipment design, and maintenance coupled with safe load paths and crane interlock devices restricting movement over critical areas.

Staff guidelines resulting from the foregoing are tabulated in Section 5 of NUREG-0612.

1.3 Plant-Specific Background

On December 22, 1980, the NRC issued a letter [4] to Kansas Gas & Electric Company, the applicant for Wolf Creek Plant requesting that the applicant review provisions for handling and control of heavy loads at Wolf Creek Plant, evaluate these provisions with respect to the guidelines of NUREG-0612, and provide certain additional information to be used for an independent determination of conformance to these guidelines. Standardized Nuclear Unit Power Plant System provided responses to this request on June 22, 1981, SLNRC 81-48 [5]; August 4, 1982, SLNRC 82-033 [6] and January 27, 1984, SLNRC 84-0008 [1].

2. EVALUATION AND RECOMMENDATIONS

2.1 Overview

The following sections summarize Kansas Gas & Electric Company's review of heavy load handling at Wolf Creek Plant accompanied by EG&G's evaluation, conclusions, and recommendations to the applicant for making the facilities more consistent with the intent of NUREG-0612.

2.2 Heavy Load Overhead Handling Systems

Table 2.1 presents the applicant's list of overhead handling systems which are subject to the criteria of NUREG-0612. The applicant has indicated that the weight of a heavy load for the facilities as 2,000 pounds per the NUREG-0612 definition.

2.3 Guidelines

The following are specific guidelines for overhead handling systems operated in plant areas containing equipment required for reactor shutdown, decay heat removal or spent fuel cooling. The Wolf Creek Plant is a pressurized water reactor plant.

2.3.1 Spent-Fuel Pool Area [NUREG-0612, Article 5.1.2]

(1) "The overhead crane and associated lifting devices used for handling heavy loads in the spent-fuel pool area should satisfy the single-failure-proof guidelines of Section 5.1.6 of this report.

OR

(2) "Each of the following is provided:

(a) Mechanical stops or electrical interlocks should be provided that prevent movement of the overhead crane load block over or within 15 feet horizontal (4.5 meters) of the spent-fuel pool. These mechanical stops or electrical interlocks should not be bypassed

TABLE 2.1 NONEXEMPT HEAVY LOAD HANDLING SYSTEMS CALLOWAY PLANT UNITS 1 AND 2
(FUEL HANDLING CRANE DATA)

	Name of Crane			
Parameters	Polar Crane	Cask Handling Crane	Spent Fuel Pool Bridge Crane	Refueling Machine
Capacity of main hoist	260.0 tons	150.0 tons	2.0 tons	2.4 tons
Capacity of auxiliary monorail hoist (constr)	25.0 tons	5.0 tons		
Capacity of auxiliary monorail	25.0 tons	5.0 tons and 2.0 tons (2)		
Capacity of main trolley	260.0 tons	130.0 tons	2.0 tons	1.5 tons
Capacity of lift beam	500.0 tons			2.4 tons
Maximum main hoist speed (normal)	5.0 fpm	3.75 fpm	21.0 fpm	20.0 fpm
Maximum auxiliary monorail hoist speed (normal)	40.0 fpm			
Minimum auxiliary monorail hoist speed (normal)	3.0 fpm			
Maximum trolley speed (normal)	51.5 fpm	20.0 fpm	30.0 fpm	20.0 fpm
Minimum trolley speed (normal)	6.0 fpm	6.0 fpm	10.0 fpm	
Maximum bridge speed (normal)	51.5 fpm	20.0 fpm	30.0 fpm	40.0 fpm
Minimum bridge speed (normal)	6.0 fpm	6.0 fpm	10.0 fpm	
Maximum load during plant operation	167.5 tons	125.0 tons	1,870 lbs	
Normal expected load range	0-167.5 tons	0-125.0 tons	0-1,870 lbs	
Maximum construction load	475.0 tons			
Maximum main hoist speed (constr)	5.0 fpm			
Minimum main hoist speed (constr)	3.0 fpm			
Maximum trolley speed (constr)	51.4 fpm			
Minimum Trolley speed (constr)	6.0 fpm			
Maximum bridge speed (constr)	51.5 fpm			
Minimum bridge speed (constr)	6.0 fpm			
Normal load range (constr)	0-475.0 tons			
Maximum monorail hoist speed		20.0 fpm		22.0 fpm
Minimum monorail hoist speed		10.0 fpm		7.0 fpm
Maximum monorail trolley speed		32.0 fpm		
Minimum monorail trolley speed		16.0 fpm		
Lifting limitation	28.5 ft (above vessel flange)	Cask bottom 3 in. above floor El. 2047 ft-6 in.	24 ft-3 in. (hook limit is 2066 ft-8 in.)	

TABLE 2.1 (continued)

Parameters	Polar Crane	Cask Handling Crane	Spent Fuel Pool Bridge Crane	Refueling Machine
	Name of Crane			
Seismic Class	(3)	(3)	(3)	(3)
Design Standards General	CMAA No. 70 (1975)	CMAA No. 70 (1975)	CMAA No. 70 (1975)	CMAA No. 70 (1975)
Structural	Covered by CMAA	Covered by CMAA	Covered by CMAA	ASME Sect. III,
Electrical	NFPA Vol. 5 Art. 610 1974-1975	NFPA Vol. 5 Art. 610 1974-1975	NFPA Vol. 5 Art. 610 1974-1975	NFPA Vol. 5 Art. 610 1974-1975
Materials	ASTM Std's.	ASTM Std's.	ASTM Std's.	ASTM Std's.
Others	OSHA 29 CFR 1910 & 1926	OSHA 29 CFR 1910 & 1926	OSHA 29 CFR 1910 & 1926	OSHA 29 CFR 1910 & 1926

NOTES:

(1) Rated speeds given are within ± 10 percent of the actual speeds.

(2) Refer to Figure 23, Reference 4. A 2-ton limit to the monorail hoist exists only over area B or Figure 23.

(3) Seismic Category 1.

when the pool contains "hot" spent fuel, and should not be bypassed without approval from the shift supervisor (or other designated plant management personnel). The mechanical stops and electrical interlocks should be verified to be in place and operational prior to placing "hot" spent fuel in the pool.

(b) The mechanical stops or electrical interlocks of 5.1.2(2)(a) above should also not be bypassed unless an analysis has demonstrated that damage due to postulated load drops would not result in criticality or cause leakage that could uncover the fuel.

(c) To preclude rolling if dropped, the cask should not be carried at a height higher than necessary and in no case more than six (6) inches (15 cm) above the operating floor level of the refueling building or other components and structures along the path of travel.

(d) Mechanical stops or electrical interlocks should be provided to preclude crane travel from areas where a postulated load drop could damage equipment from redundant or alternate safe shutdown paths.

(e) Analyses should conform to the guidelines of Appendix A.

<u>OR</u>

(3) "Each of the following are provided (Note: This alternative is similar to (1) above, except it allows movement of a heavy load, such as a cask, into the pool while it contains "hot" spent fuel if the pool is large enough to maintain wide separation between the load and the "hot" spent fuel.):

(a) "Hot" spent fuel should be concentrated in one location in the spent-fuel pool that is separated as much as possible from load paths.

(b) Mechanical stops or electrical interlocks should be provided to prevent movement of the overhead crane load block over or within 25 feet horizontal (7.5 m) of the "hot" spent fuel. To the extent practical, loads should be moved over load paths that avoid the spent-fuel pool and kept at least 25 feet (7.5 m) from the "hot" spent fuel unless necessary. When it is necessary to bring loads within 25 feet of the restricted region, these mechanical stops or electrical interlocks should not be bypassed unless the spent fuel has decayed sufficiently as shown in Table 2.1-1 and 2.1-2, or unless the total inventory of gap activity for fuel within the protected area would result in off-site doses less than 1/4 of 10 CFR Part 100 if released, and such bypassing should require the approval from the shift supervisor (or other designated plant management individual). The mechanical stops or electrical interlocks should be verified to be in place and operational prior to placing "hot" spent fuel in the pool.

(c) Mechanical stops or electrical interlocks should be provided to restrict crane travel from areas where a postulated load drop could damage equipment from redundant or alternate safe shutdown paths. Analyses have demonstrated that a postulated load drop in any location not restricted by electrical interlocks or mechanical stops would not cause damage that could result in criticality, cause leakage that could uncover the fuel, or cause loss of safe shutdown equipment.

(d) To preclude rolling, if dropped, the cask should not be carried at a height higher than necessary and in no case more than six (6) inches (15 cm) above the operating floor level of the refueling building or other components and structures along the path of travel.

(e) Analyses should conform to the guidelines of Appendix A.

OR

(4) "The effects of drops of heavy loads should be analyzed and shown to satisfy the evaluation criteria of Section 5.1 of this report. These analyses should conform to the guidelines of Appendix A."

A. Summary of Applicant's Statements

The applicant states that two cranes are capable of carrying loads over the spent fuel pool. These are

1. Fuel Building Cask Handling Crane

2. Spent Fuel Pool Bridge Crane

Neither crane is designed to be single failure proof.

The spent fuel pool bridge crane is designed to CMAA-70 Class B standards. The crane is also designed to maintain its integrity during a safe shutdown event.

The crane consists of a 5 ton capacity wheeled bridge structure with a steel deck walkway, a 2 ton motorized monorail trolley, and a 5 ton monorail push type trolley. The 2 ton electric hoist is used to transfer new and spent fuel assemblies. The 5 ton manual chain hoist and trolley are used to transfer the fuel pool transfer gates to and from their normal positions. The hoists share the same monorail. Geared-type upper and lower limit switches are used to control the electric systems of the crane. In addition to the geared type limit switches, a weight operated hoist upper limit switch is used in the spent fuel pool bridge crane electric hoist system.

The fuel building cask handling crane is a CMAA 70 Class A type indoor electrical overhead traveling bridge crane with a single trolley with the necessary controls and a festooned control station. The main hoist capacity is 150 tons. The cask handling crane is also equipped with a monorail and a 5 ton hoist.

Limit switches and mechanical stops are located to prevent any crane other than the spent fuel bridge crane from traveling over the spent fuel pool. Geared-type upper and lower limit switches are used in the control circuit of each hoist system of the cask handling crane. A weight operated hoist upper limit switch is used in each hoist system of the crane in addition to the geared-type limit switches. The two types of hoist upper limit switches are redundant and independent.

The cask handling crane is also prevented from traveling near or over the spent fuel pool by both mechanical and electrical interlocks. These interlocks will not be by-passed once the spent fuel storage racks have been installed, since there is no foreseeable need to handle other heavy loads over the spent fuel pool. The distance between the fuel pool edge and the center of the cask is 11 feet thus precluding a drop into the pool. The fuel cask handling crane lift will also be limited to 3 inches above the floor by redundant and electrical interlocks.

Administrative procedures will be developed to ensure that heavy loads carried over the spent fuel will be limited in lift height and potential energies less than that analyzed by the rack vendor, and thus racks containing fuel will not be subjected to damaging drops that could raise the K_{eff} above 0.95.

A 12 ft thick base mat below the spent fuel pool prevents pool prevents pool damage of a magnitude to develop a leak in the event that a load drop occurred. Make up water to the spent fuel pool can be supplied from any of three separate sources should leakage occur.

A fuel handling accident in the fuel building was postulated and analyzed in Snupps FSAR Section 15.7.4 and the dose resulting from such a drop is within one-fourth of 10 CFR Part 100.

Analyses have been made of a spent fuel shipping cask drop at three locations namely into the decontamination pit, into the cask loading pool, and onto the slab between the two pools.

No load drop analyses were made of loads dropped into the spent fuel pool since interlocks prevent loads from passing over the pool and three independent, concurrent failures would have to occur for this to happen.

The layout and arrangement of the fuel handling area is such that no critical equipment is located under the area traversed by the cask crane. This effectively precludes damage to any critical component in the fuel handling area. A drop of equipment handled by the fuel pool cooling pump monorail and hoist would not result in any damage to equipment needed for safe shutdown.

B. EG&G Evaluation

The cask handling crane and the spent fuel bridge crane are the only cranes capable of handling loads over the spent fuel pool.

Neither the cask handling crane nor the spent fuel bridge crane meet the single failure proof guidelines of Section 5.1.6 of NUREG 0612.

The applicant has demonstrated compliance with the intent of Acceptance Criteria I, II, III, and IV of NUREG 0612 by utilizing alternative No. 2 of NUREG 0612 Section 5.1.2 for the cask handling crane. Both mechanical and electrical interlocks have been provided on the crane to prevent crane travel near or over the spent fuel. By design the distance between the load block and the spent fuel is approximately 11 feet and does not meet the requirement of this guideline, of 15 feet. However calculations of a load drop at this location indicate the cask would not drop into the pool.

Redundant electrical interlocks will prevent the cask from being raised more than 3 inches above the cask floor, thus precluding a cask tipover should a drop occur near the pool.

Alternative No. 4 of NUREG 0612 Section 5.1.2 was used to demonstrate that loads carried over the spent fuel pool will meet Acceptance Criteria I, II, III, and IV of Section 5.1.

C. <u>EG&G Conclusions and Recommendations</u>

1. The applicant has shown that the two heavy load handling systems in the spent fuel pool area will satisfy the criteria of NUREG 0612 Article 5.1.2 by showing that the cask handling crane meets Alternative 2 of this guideline and that the spent fuel bridge crane meets Alternative 4 of this guideline.

2. The applicant should verify that cask handling crane, electrical interlocks, and mechanical stops are in place and operational prior to placing hot fuel into the crane.

   EG&G considers the applicant to be consistent with the intent of this guideline.

2.3.2 <u>Reactor Building [NUREG-0612, Article 5.1.3]</u>

(1) "The crane and associated lifting devices used for handling heavy loads in the containment building should satisfy the single-failure-proof guidelines of Section 5.1.6 of this report.

<u>OR</u>

(2) "Rapid containment isolation is provided with prompt automatic actuation on high radiation so that postulated releases are within limits of evaluation Criterion I of Section 5.1 taking into account delay times in detection and actuation; and analyses have been performed to show that evaluation criteria II, III, and IV of Section 5.1 are satisfied for postulated load drops in this area. These analyses should conform to the guidelines of Appendix A.

<u>OR</u>

(3) "The effects of drops of heavy loads should be analyzed and shown to satisfy the evaluation criteria of Section 5.1. Loads analyzed should include the following: reactor vessel head; upper vessel internals; vessel inspection platform; cask for damaged fuel; irradiated sample cask; reactor coolant pump; crane load block; and any other heavy loads brought over or near the reactor vessel or other equipment required for continued decay heat removal and maintaining shutdown. In this analysis, credit may be taken for

containment isolation if such is provided; however, analyses should establish adequate detection and isolation time. Additionally, the analysis should conform to the guidelines of Appendix A."

A. Summary of Applicant's Statements

The polar crane and refueling machine are capable of handling loads and traveling over the reactor vessel.

The polar crane is built in compliance with CMAA 70 Class C and the weld stresses are in compliance with AWS D1.1 requirements. The crane is designed to maintain its integrity with a load during an SSE.

Although the crane is considered highly reliable with many safety features, it does not meet the single failure-proof guidelines of NUREG 0612 Section 5.1.6.

The polar crane bridge and polar crane trolley are equipped with seismic restraints.

Except for infrequent uses of the polar cane for light loads during hot standby and hot shutdown, the polar crane will not be operated except during cold shutdown and refueling. After vessel head and upper internals are removed, the crane will be administratively controlled to preclude travel over the open vessel while fuel is in the reactor.

A reactor vessel head drop analysis was performed and results indicated that the vessel and core would remain intact with no fuel damage should a load drop occur.

A drop of the upper internals and lifting rig would produce smaller loads on the reactor vessel, reactor nozzles and fuel assemblies and likewise would produce no fuel damages. Therefore damaged core and criticality calculations were not needed.

Drop of the vessel head onto the vessel flange may damage seal rings causing water to leak from the refueling canal into lower levels of the containment.

Adequate water from the reactor water storage tank is available to continue core cooling until enough water is available in the containment recirculation sump to transfer the reactor heat removal pumps from the reactor water supply tank to the containment recirculation sump. The spent fuel pool is not affected by a head drop which could produce a seal leak.

The refueling machine is used to change out fuel from the reactor. This machine includes the following provisions to ensure safe handling of fuel assemblies

a. Safety Interlocks. Operations which could endanger the operators or damage the fuel are prohibited by mechanical or fail safe electrical interlocks or by redundant electrical interlocks.

b. Bridge and Trolley Holddown Devices. Horizontal and vertical restraints are both adequately designed to withstand the force and movements resulting from an SSE.

c. Main Hoist Braking System. The main hoists are equipped with two independent braking systems, an electrically actuated brake, and a mechanically actuated brake.

d. Fuel Assembly Support System. The main hoist system is supplied with redundant paths of load support so that failure of any component will not result in a fuel assembly being dropped.

The SNUPPS FSAR provides an analysis of a fuel handling accident which indicates doses are well within one fourth of 10 CFR 100 guidelines.

B. EG&G Evaluation

1. The applicant has performed analyses on the reactor vessel head drop at six critical points along its travel path to the vessel head storage stand in support of compliance with Acceptance Criteria I, II, and III of NUREG 0612, Section 5.1.

   The analysis indicated that there would be no damage of any consequence to the structural integrity of the reactor vessel nozzles. Core cooling capability and fuel cladding integrity will be maintained.

2. Analyses were conducted for five postulated drops of the component cooling water pumps and motors. The analyses indicated no floor slab failure in the event of a drop. It is concluded that no administrative procedures are required in the removal of these items.

3. Three postulated drop cases of the auxiliary feedwater pump and motor in the auxiliary building were analyzed. The analyses indicated no floor slab failure would occur should a drop occur.

4. Three postulated drop cases of the mixed hydrogen fan were analyzed. In each case the calculated bending and shear stresses of the affected beams were within the elastic range. Connection capacities of the beams were also found to be acceptable.

C. EG&G Conclusions and Recommendations

1. The applicant has demonstrated by load drop analyses that postulated load drops of the reactor vessel head, the component cooling pump and motor, the auxiliary pump and motor and the mixed hydrogen fans, all in the reactor and auxiliary building, would not result in damage to the reactor vessel, the floor slab or safe shutdown equipment. EG&G concludes that the applicant has met the intent of this guideline.

2.3.3 Other Areas [NUREG-0612, Article 5.1.5]

(1) "If safe shutdown equipment are beneath or directly adjacent to a potential travel load path of overhead handling systems, (i.e., a path not restricted by limits of crane travel or by mechanical stops or electrical interlocks) one of the following should be satisfied in addition to satisfying the general guidelines of Section 5.1.1:

(a) The crane and associated lifting devices should conform to the single-failure-proof guidelines of Section 5.1.6 of this report;

OR

(b) If the load drop could impair the operation of equipment or cabling associated with redundant or dual safe shutdown paths, mechanical stops or electrical interlocks should be provided to prevent movement of loads in proximity to these redundant or dual safe shutdown equipment. (In this case, credit should not be taken for intervening floors unless justified by analysis.)

OR

(c) The effects of load drops have been analyzed and the results indicate that damage to safe shutdown equipment would not preclude operation of sufficient equipment to achieve safe shutdown. Analyses should conform to the guidelines of Appendix A, as applicable.

(2) "Where the safe shutdown equipment has a ceiling separating it from an overhead handling system, an alternative to Section 5.1.5(1) above would be to show by analysis that the

largest postulated load-handled by the handling system would not penetrate the ceiling or cause spalling that could cause failure of the safe shutdown equipment."

A. Summary of Applicant's Statements

The applicant has designed and laid out the SNUPPS plants so that in the event of a load drop, the redundant safety related trains are sufficiently separated, and protected by locating the trains on opposite sides of the reactor and auxiliary building. Systems separated were the mechanical components, electrical components, and essential support systems such as component cooling water, essential services water, and ventilation.

Reactor Building Cranes

The following describes the analysis of each crane and load drop inside the reactor building.

1. Polar Crane During Cold Shutdown

The polar crane can be used during cold shutdown to lift any load not associated with reactor coolant pressure boundaries inside the containment.

A load drop outside of the secondary shield will not affect the safe shutdown or decay heat removal capability because of the physical separation of the RHR System and its power supplies.

A load drop inside the secondary shield walls could potentially affect a Class I branch line. However the RHR system would assure continuing decay heat removal.

2. Polar Crane During Hot Standby and Hot shutdown

The crane will be used during these operating conditions to assist with maintenance that can be performed in a hot plant condition. Loads at these conditions will be light and infrequent. All load paths for these conditions will be defined at the time of operation. A postulated load drop which would damage either the primary or secondary side pressure boundaries would not result in radioactive releases above the 10 CFR part 100 limits.

3. Polar Crane During Refueling

The polar crane is used to remove the reactor vessel head. A load drop would not be any more severe than the cases above and adequate cooling water would be available.

4. Containment, and Secondary Shield Wall Area Jib Cranes

The containment jib cranes are used to remove the hydrogen mixing fans from their bottom skirts. The fans are located above the primary pumps on concrete slabs, which are removable for access to the reactor coolant pumps. The concrete slabs are removed by the polar crane during cold shutdown. A load drop of these blocks would cause no more damage than those described for the polar crane during cold shutdown.

The secondary shield wall area jib crane is used during cold shutdown and refueling to handle components located in the pressurizer compartments. A load drop by this crane would cause no more severe damage than that caused by a polar crane load drop. Analysis indicates there would be no effect on decay heat removal should a drop break the pressurizer boundary.

5. <u>Containment Equipment Hatch Hoists During Cold Shutdown</u>

A drop of the hatch will not affect safe shutdown or decay heat removal.

6. <u>Reactor Building Elevator Auxiliary Monorail and Hoist</u>

This monorail and hoist is used to handle miscellaneous equipment. A load drop would not affect safe shutdown or decay heat removal.

<u>Auxiliary Building</u>

The following describes the analyses of the overhead load handling devices inside the auxiliary building.

Load drop accidents of the following hoists and monorails, which service elevation 1974', were not analyzed under the criteria of NUREG 0612 since there is sufficient physical separation of the redundant safe shutdown equipment train to preclude loss of both trains of safe shutdown equipment.

(a) Centrifugal charging pump monorail and hoist

(b) Safety injection pump monorail and hoist

(c) RHR pump monorail and hoist

(d) Containment spray pump monorail and hoist

(e) Positive displacement charging pump monorail and hoist

(f) Moderating heat exchanger monorail and hoist.

Load drop accidents of the following hoists and monorails were evaluated as follows:

1. <u>Auxiliary Building Filter Room Monorail and Hoists</u>

This equipment is used to handle the reactor coolant, seal water injection, seal water return, and boric acid filters and their respective removable plugs. In the event of a load drop and floor slab failure, the following equipment could be affected.

a. Modulating heat exchanger
b. Letdown chiller heat exchanger
c. CVCS Chiller Unit
d. Centrifugal charging pump

Of the above equipment, only the centrifugal charging pump is required for safe shutdown, but a redundant pump is sufficiently separated such that a loss of both pumps is extremely unlikely.

A load drop on a pipe chase in the area would not preclude safe shutdown of the plant.

2. <u>Moderating Heat Exchanger Monorail and Hoist</u>

This equipment is used to handle miscellaneous equipment in the northern half of the CVCS sampling room and the seal water heat exchanger.

In the event of a load drop and floor slab failure, the affected equipment below would be a centrifugal charging pump needed for safe shutdown and a pipe chase. A single load drop on the charging pump would not preclude safe shutdown due to equipment redundancy and a load drop on the pipe chase would not adversely affect safe shutdown.

In the event of a load drop of the seal water heat exchanger or floor, failure the affected equipment below would be a centrifugal charging pump, a charging pump room cooler, and a pipe chase. The pump and cooler are needed for safe shutdown but due to separation of the redundant equipment, safe shutdown of the reactor would not be impaired by the drop. A load drop on the pipe chase would not adversely affect safe shutdown.

3. Auxiliary Feedwater Monorail and Hoist

This equipment is used to handle the auxiliary feedwater motor and turbine driven pumps. In the event of a load drop or floor slab failure the affected equipment below would be the auxiliary steam condensate recovery pump, and storage tank, the auxiliary steam deaerator feed pump and the auxiliary feedwater sump pumps. None of this equipment is needed for safe shutdown. Additional equipment in the path of the load drop and slab failure are two service water lines approximately 2 feet apart which feed redundant trains of the auxiliary feedwater motor and turbine drum pump. An analysis of this drop indicated that a slab failure would not occur and therefore the two lines would not be affected.

4. Boric Acid Batch Tank Monorail and Hoist

This equipment is used to handle the boric acid batch tank and chemical storage. In the event of a load drop and floor slab failure, the boric acid tank, boric acid transfer pump and associated equipment would be affected. These components may be used for safe shutdown, but separated redundant equipment including cable tray exist. Therefore a load drop will not adversely affect a safe shutdown.

5. Resin Charging Tank Area Monorail and Hoist

This equipment is used to handle loading of the resin charging tank and boral thermal regeneration filters. Affected equipment below in the event of a load drop are the boron regeneration filters, and a pipe chase. A drop on either the filters or pipe chase will not adversely affect safe shutdown.

6. Component Cooling Water Pump Monorail and Hoist

This equipment is used to handle the component cooling water pumps. In the event of a load drop and slab failure, the affected equipment below would be

1. Nuclear sampling panels
2. Seal water injection filters
3. Seal water return filter (2)
4. Reactor coolant filter
5. Boric acid filters
6. Position displacement charging pump
7. Safety injection pump
8. Centrifugal charging pump
9. Moderating heat exchanger
10. Letdown reheat heat exchanger
11. Letdown chiller heat exchanger
12. CVCS chiller unit

Items 1, 3, 4, 5, 9 through 12 are not required for safe shutdown. Items 7 and 8 are needed for safe shutdown, but physical separation exists to preclude safe shutdown capability in the event of a load drop and slab failure. The two seal injection filters, item 2, are located rather close together and are needed for safe shutdown as well as item 5, a boric acid transfer line, and a refueling water line. An analysis of a load drop by the component cooling water pump monorail and hoist indicated that a slab failure would not occur and therefore the equipment and components needed for a safe shutdown would not be affected.

7. Auxiliary Building HVAC Monorail and Hoist

This equipment is used to handle miscellaneous fans and equipment through the hatch which extends to the auxiliary building basement. In the event of a load drop, the loss of the equipment carried would not preclude a safe shutdown.

8. Component Cooling Water Surge Tank Area Monorail and Hoist

This equipment is used to handle miscellaneous equipment through the hatch which continues down to the basement of the auxiliary building. In the event of a load drop, the equipment would fall to the basement floor and no other equipment would be affected except conduit which has redundancy. Therefore no damage would preclude a safe shutdown.

9. Main Steam Relief and Isolation Valve Monorail and Hoist

This equipment is used to handle main steam isolation valves as well as main feedwater isolation valves. In order for these valves to be lifted they must first be disassembled. Removal during plant cold shutdown may be through the equipment hatches in the turbine building wall. In the event of a load drop, the affected components would be the main area steam and main feedwater piping. However damage to these would not preclude a safe shutdown, since the reactor would already be in shutdown.

In the event of a slab failure due to a load drop, the following equipment would be affected.

1. Auxiliary feedwater pump room coolers (2)

2. Auxiliary feedwater motor driven pumps (2)

3. Auxiliary feedwater turbine driven pump

4. Auxiliary steam condensate recovery and storage tanks

5. Auxiliary steam deaerator fuel pump

6. Auxiliary feedwater sump pump

7. Pipe chase

Only items 2, 3 and two essential service water lines in item 7 are needed for safe shutdown. An analysis of a drop by the above crane/hoist indicated slab failure would not occur.

B. EG&G Evaluation

1. The applicant has indicated that safe load path considerations, training and administrative procedures will ensure plant safety when using the polar crane during hot standby and hot shutdown.

2. Heavy load drops in any area of the plant will not prevent safe shutdown, due to physical, and horizontal separation of the safe shutdown equipment.

3. Detailed structural analyses of the floors were made for drops of the main steam valve, the main feedwater isolation valve, the component cooling water pumps and the auxiliary feedwater pump in the auxiliary building; and for the hydrogen mixing fan, and the spent fuel shipping cask in the fuel building.

   The analyses indicated that none of the drops would cause a floor failure which would endanger safe shutdown equipment.

C. EG&G Conclusions and Recommendations

1. The applicant has analyzed the impact of load drops in the auxiliary building, the reactor building and the fuel building and determined that no floor failures would occur.

2. Load lifting height limits should be set so as to not exceed the lifting height used in the load drop impact analyses.

3. EG&G concludes that the applicant is consistent with the intent of this guideline.

2.3.4 Single-Failure-Proof Handling Systems [NUREG-0612, Article 5.1.6]

(1) "Lifting Devices:

(a) Special lifting devices that are used for heavy loads in the area where the crane is to be upgraded should meet ANSI N14.6 1978, "Standard For Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More For Nuclear Materials," as specified in Section 5.1.1(4) of this report except that the handling device should also comply with Section 6 of ANSI N14.5-1978. If only a single lifting device is provided instead of dual devices, the special lifting device should have twice the design safety factor as required to satisfy the guidelines of Section 5.1.1(4). However, loads that have been evaluated and shown to satisfy the evaluation criteria of Section 5.1 need not have lifting devices that also comply with Section 6 of ANSI N14.6.

(b) Lifting devices that are not specially designed and that are used for handling heavy loads in the area where the crane is to be upgraded should meet ANSI B30.9-1971, "Slings" as specified in Section 5.1.1(5) of this report, except that one of the following should also be satisfied unless the effects of a drop of the particular load have been analyzed and shown to satisfy the evaluation criteria of Section 5.1:

(i) Provide dual or redundant slings or lifting devices such that a single component failure or malfunction in the sling will not result in uncontrolled lowering of the load;

OR

(ii) In selecting the proper sling, the load used should be twice what is called for in meeting Section 5.1.1(5) of this report.

(2) "New cranes should be designed to meet NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." For operating plants or plants under construction, the crane should be upgraded in accordance with the implementation guidelines of Appendix C of this report.

(3) "Interfacing lift points such as lifting lugs or cask trunions should also meet one of the following for heavy

loads handled in the area where the crane is to be upgraded unless the effects of a drop of the particular load have been evaluated and shown to satisfy the evaluation criteria of Section 5.1:

(a) Provide redundancy or duality such that a single lift point failure will not result in uncontrolled lowering of the load; lift points should have a design safety factor with respect to ultimate strength of five (5) times the maximum combined concurrent static and dynamic load after taking the single lift point failure.

<u>OR</u>

(b) A non-redundant or non-dual lift point system should have a design safety factor of ten (10) times the maximum combined concurrent static and dynamic load."

A. <u>Summary of Applicant's Statements</u>

Not applicable. Applicant has satisfied evaluation criteria of Section 5.1.

B. <u>EG&G Evaluation</u>

Not applicable. Applicant has satisfied evaluation criteria of Section 5.1.

C. <u>EG&G Conclusions and Recommendations</u>

Not applicable. Applicant has satisfied evaluation criteria of Section 5.1.

3. CONCLUDING SUMMARY

3.1 Guideline Recommendations

The applicant has shown consistency with the intent of the guidelines of Section 2.3. Conclusions are indicated in Table 3.1.

Guideline	Recommendation
Section 2.3.1 (NUREG 0612, Article 5.1.2)	Consistent with intent of guideline
Section 2.3.2 (NUREG 0612, Article 5.1.3)	Consistent with intent of guideline
Section 2.3.3 (NUREG 0612, Article 5.1.5)	Consistent with intent of guideline
Section 2.3.4 (NUREG 0612, Article 5.1.6)	Not applicable

3.2 Additional Recommendations

3.3 Summary

EG&G concludes that the applicant is consistent with the intent of the following guidelines regarding overhead handling systems.

1. Spent Fuel Area [NUREG 0612, Article 5.1.2]

2. Reactor Building [NUREG 0612, Article 5.1.3]

2. Other Areas [NUREG 0612, Article 5.1.5].

TABLE 3.1 WOLF CREEK PLANT--NUREG-0612 OBJECTIVES COMPLIANCE MATRIX

Handling System	Single-Failure-Proof System	Off-site Radioactive Release	Damaged Fuel Criticality	Fuel Cover Water Inventory Loss	Safe Shutdown Equipment Loss
Polar Crane	C	C	C	C	C
Cask Handling Crane	C	C	C	C	C
Spent Fuel Pool Bridge Crane	C	C	C	C	C
Refueling Machine	C	C	C	C	C

C = Applicant action consistent with NUREG-0612 Risk Reduction Objective.
NC = Applicant action not consistent with NUREG-0612 Risk Reduction Objective.
NA = Risk Reduction Objective is not applicable to this handling system.

4. REFERENCES

1. Control of Heavy Loads of Nuclear Power Plants Wolf Creek Plant Phase I Final.

2. NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, NRC.

3. V. Stello, Jr. (NRC), Letter to all applicants. Subject: Request for Additional Information on Control of Heavy Loads Near Spent Fuel, NRC, 17 May 1978.

4. USNRC, Letter to Kansas Gas and Electric Company. Subject: NRC Request for Additional Information on Control of Heavy Loads Near Spent Fuel, NRC, 22 December 1980.

5. Nicholas A. Petrick, Standardized Nuclear Unit Power Plant System, Letter to H. R. Denton, Subject: Control of Heavy Loads, June 22, 1981, SLNRC 81-48.

6. Nicholas A. Petrick, Standardized Nuclear Unit Power Plant System, Letter to H. R. Denton, Subject: Control of Heavy Loads; SLNRC-033 File 0278, August 4, 1982.

7. ANSI B30.2-1976 "Overhead and Gantry Lanes."

8. ANSI N14.6-1978 Standards for Lifting Devices for Shipping Containers Weighing 10,000 lbs (4500 kg) or more for Nuclear Materials.

9. ANSI B30.9-1971 Slings.

10. CMAA-70 "Specifications for Electric Overhead Traveling Cranes."

11. Nicholas A. Petrick, Standardized Nuclear Unit Power Plant System, Letter to H. R. Denton, Subject: Control of Heavy Loads, SLNRC 84-0008, January 27, 1984.

APPENDIX M

STAFF EVALUATION OF THE MECHANICAL PROPERTIES OF THERMALLY AGED CAST STAINLESS STEEL PIPE MATERIALS REPORTED IN WESTINGHOUSE REPORT WCAP-10456

Introduction

The primary coolant piping in some Westinghouse nuclear steam supply systems (NSSSs) contains cast stainless steel base metal and weld metal. The base metal and weld metal are fabricated to produce a duplex structure of delta (δ) ferrite in an austenitic matrix. The duplex structure produces a material that has a higher yield strength, improved weldability, and greater resistance to intergranular stress corrosion cracking than a single-phase austenitic material. However, as early as 1965 (Beck et al.) it was recognized that long-time thermal aging at primary loop water temperatures (550°F-650°F) could significantly affect the Charpy impact toughness of the duplex structured alloys. Since the Charpy impact test is a measure of a material's resistance to fracture, a loss in Charpy impact toughness could result in reduced structural stability in the piping system.

The purpose of Westinghouse Report WCAP-10456 is to evaluate whether cast stainless steel base metal and weld metal containing postulated cracks will be sensitive to unstable fracture during the 40-year life of a nuclear power plant. In order to determine whether a piping system will behave in such a fashion, the pipe materials' mechanical properties, design criteria and method of predicting failure must be established. In this evaluation, the NRC staff assesses the mechanical properties of thermally aged cast stainless steel pipe materials, which are reported in WCAP-10456.

Discussion

(1) Weld Metal

Report WCAP-10456 refers to test results reported in a paper by Slama et al. (1983) to conclude that the weld metal in primary loop piping would not be overly sensitive to aging and that the aged cast pipe base metal material would be structurally limiting. In the Slama report, eight welds were evaluated. The tensile properties were only slightly affected by aging. The Charpy V-notch impact energy in the most highly sensitive weld decreased from 7daJ/cm^2 (40 ft-lb) to near 4daJ/cm^2 (24 ft-lb) after aging for 10,000 hours at 400°C (752°F). This change was not considered significant. The relatively small effect of aging on the weld, as compared to cast pipe material was reported to be caused by a difference in microstructure and lower levels of ferrite in the weld than in the cast pipe material.

(2) Cast Stainless Steel Pipe Base Metals

Report WCAP-10456 contains mechanical property test results from a number of heats of aged cast stainless steel material and a metallurgical study, which was performed by Westinghouse, to support a statically based model for predicting the effect of thermal aging on the Charpy impact test properties of cast stainless steel. As a result of these tests and the proposed model, Westinghouse concluded that the fracture toughness test results from one heat of material tested represents end-of-life conditions for the 10 plants surveyed. The 10 plants surveyed are identified as Plants A through J.

(a) Mechanical Property Test Results Reported in WCAP-10456

Mechanical property test results on aged and unaged cast stainless steel materials were reported in papers by Landerman and Bamford (1978); Bamford, Landerman, and Diaz (1983); and Slama et al. (1983); these papers were discussed in WCAP-10456. In addition, Westinghouse performed confirmatory Charpy V-notch and J-integral tests on aged cast stainless steel material, which was tested and evaluated by Slama's group.

The results of these tests indicate that

- The fatigue crack growth rate of aged or unaged material in air and pressurized water reactor environments were equivalent.

- Tensile properties were essentially unaffected except for a slight increase in tensile strength and a decrease in ductility.

- J-integral test results indicate that the J_{1c} and tearing modulus, T, are affected by aging.

(b) Mechanism Study in WCAP-10456

The tests and literature survey conducted by Westinghouse indicate that the proposed mechanism of aging occurs in the range of operating temperatures for pressurized water reactors and the data from accelerated aging studies can be used to predict the behavior at operating temperatures.

(c) Cast Stainless Steel Pipe Test

The materials data discussed in the previous section (Section 2) of this evaluation were obtained from small specimens. As a consequence, the J-R results are limited to relatively short crack extensions. To investigate the behavior of cast stainless steel in actual piping geometry, Westinghouse performed two experiments: one test was performed on thermally aged cast stainless steel and the other test was performed on cast stainless steel that was not thermally aged.

Each pipe tested contained a through-wall circumferential crack to the extent specified in WCAP-10456. The pipe sections were closed at the ends, pressurized to nominal PWR operating pressure, and then had bending loads applied.

The results of the tests were very similar, in that both pipes displayed extensive ductility, and stable crack extension. There was no observed unstable crack extension or fast fracture.

The results of the Westinghouse pipe experiments indicate that cast stainless steel, both aged and unaged, can withstand crack extensions well beyond the range of J-R results with small specimens. However, if crack extension is predicted in a actual application of thermally aged cast stainless steel in a piping system, the staff finds that it is prudent to limit the applied J to 3,000 in.-lb/in.^{2} or less unless further studies and/or experiments demonstrate that higher values are tolerable. Loss of initial toughness from thermal aging of cast stainless steels at normal nuclear facility operating temperatures occurs slowly over the the course of many years; therefore, continuing study of the aging phenomenon may lead to a relaxation of this position. Conversely, in the unlikely event that the total loss of toughness and the rate of toughness loss are greater than those projected in this evaluation, the staff will take appropriate action to limit the values to those that can be justified by experimental data. Because the aging is a slow process, the staff finds there would be sufficient time for the staff to recognize the problem and to rectify the situation. However, the staff finds this to be a highly unlikely situation, because the staff has accepted only the lower bounds of data that were gathered among 10 plants encompassing the range of materials in use.

(d) Effects of Thermal Aging on Westinghouse-Supplied Centrifugally Cast Reactor Coolant Piping Reported in WCAP-10456

The reactor coolant cast stainless steel piping materials in the plants identified in WCAP-10456 as A through J, were produced to Specification SA-351, Class CF8A, as outlined in ASME Code, Section II, Part A, and also to Westinghouse Equipment Specification G-678864, as revised. For these materials, Westinghouse has calculated the predicted end-of-life Charpy V-notch properties, based on their proposed model. The two standard deviation end-of-life lower limit value for all the plants surveyed was greater than the Charpy V-notch properties of the aged reference materials, which Westinghouse indicates represent end-of-life properties for all the plants. As a result, Westinghouse concluded that the amount of embrittlement in the aged reference material exceed the amount projected at end-of-life for all cast stainless steel pipe materials in Plants A through J.

Conclusions

On the basis of its review of the information and data contained in Westinghouse Report WCAP-10456, the staff concludes that:

(1) Weld metal that is used in cast stainless steel piping system is initially less fracture resistant than the cast stainless steel base metal. However, the weld metal is less susceptible to thermal aging than the cast stainless steel base metal. Hence, at end of life the cast stainless steel base metal is anticipated to be less fracture-resistant material.

(2) The Westinghouse proposed model may be used to predict the relative amount of embrittlement on a heat of cast stainless steel material. The two standard deviation lower confidence limit for this model will provide a useful engineering estimate of the predicted end-of-life Charpy impact properties for cast stainless steel base metal.

(3) Since there is considerable scatter in J-integral test data for the heats of material tested, lower bound values for J_{1c} and T should be used as engineering estimates for the fracture resistance of the aged reference material. The staff believes these values should also provide a lower bound for the fracture resistance of aged and unaged weld metal. If crack extension is predicted in an actual application of cast stainless steel in a piping system, the staff concludes that the applied J should be limited to 3,000 in.-lb/in.^{2} or less, unless further studies and tests demonstrate that higher values are tolerable. The Westinghouse pipe tests demonstrate that this may be possible.

(4) Since the predicted end-of-life Charpy impact values for the materials in Plants A through J are greater than the value measured for the aged reference material, the lower bound fracture properties for aged reference material may by used to determine the fracture resistance for the cast stainless steel material in Plants A through J.

References

Bamford, W. H., E. I. Landerman, and E. Diaz, "Thermal Analysis of Cast Stainless Steel and Its Impact on Piping Integrity," ASME Pressure Vessel and Piping Conference, Portland, Oregon, June 1983, American Society of Mechanical Engineers, N.Y., Circumferential Cracks in Pressure Vessels and Piping, Vol. II, 1984, pp. 137-172.

Beck, F. H., E. A. Schoefer, J. W. Flowers, and M. E. Fontana, "New Cast High Strength Alloy Grades by Structural Control," ASTM STP 369 (1965).

Landerman, E. I., and W. H. Bamford, "Fracture Toughness and Fatigue Characteristics of Centrifugally Cast Type 316 Stainless Steel After Simulated Thermal Service Conditions." Presented at the Winter Annual Meeting of the ASME, San Francisco, Ca., December 1978 (MPC-8 ASME).*

Slama, G., P. Petrequin, S. H. Masson, and T. R. Mager, "Effect of Aging on Mechanical Properties of Austenitic Stainless Steel Casting and Welds," presented at SMIRT 7 Post Conference Seminar 6 - Assuring Structural Integrity of Steel Reactor Pressure Boundary Components, August 29/30, 1983, Monterey, Ca.*

*Available from Project Manager, Paul W. O'Connor, U.S. Nuclear Regulatory Commission, Division of Licensing, Washington, D.C. 20555

NRC FORM 335 (6-83) U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any): NUREG-0881 Supplement No. 5

2. Leave blank

3. TITLE AND SUBTITLE: Safety Evaluation Report related to the operation of Wolf Creek Generating Station, Unit No. 1

4. RECIPIENT'S ACCESSION NUMBER

5. DATE REPORT COMPLETED: MONTH March YEAR 1985

6. AUTHOR(S)

7. DATE REPORT ISSUED: MONTH March YEAR 1985

8. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

9. PROJECT/TASK/WORK UNIT NUMBER

10. FIN NUMBER

11. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Same as 8. above.

12a. TYPE OF REPORT: Technical

12b. PERIOD COVERED (Inclusive dates): December 1983 - March 1985

13. SUPPLEMENTARY NOTES: STN 50-482

14. ABSTRACT (200 words or less)

Supplement No. 5 to the Safety Evaluation Report related to the operation of the Wolf Creek Generating Station, Unit No. 1 updates the information contained in the Safety Evaluation Report, dated April 1982 and Supplements 1, 2, 3 and 4, dated August 1982, June 1983, August 1983, and December 1983, respectively. Supplement No. 5 also addresses open issues and items concerning the issuance of a five percent low power license.

The Safety Evaluation And its supplements pertains 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 18, 1980. The Construction Permit No. CPPR-147 was issued on May 17, 1977. The facility is located in Coffey County, Kansas.

15a. KEY WORDS AND DOCUMENT ANALYSIS

15b. DESCRIPTORS

16. AVAILABILITY STATEMENT: Unlimited

17. SECURITY CLASSIFICATION (This report): UNCLASSIFIED

18. NUMBER OF PAGES

19. SECURITY CLASSIFICATION (This page): UNCLASSIFIED

20. PRICE: $

431-02-0418
Box 13

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FIRST CLASS MAIL
POSTAGE & FEES PAID
USNRC
WASH. D.C.
PERMIT No. G-67