Questions 005.1 through 031.20, are the November 17, 1977 NRC questions which were addressed to all construction permit applications that referenced RESAR-3. The following provides the response, as applicable to WCGS.

Q005.1 Provide the list of transients that were analyzed in determining the maximum steam system pressure transient for sizing the steam generator safety valves.

RESPONSE

Refer to Section 5.2.2.

Q005.2 In reference to Section 5.3.4, provide Reactor Coolant System Temperature - Percent Power map for plant with loop stop valves if different from Figure 5.3-1.

RESPONSE

Since WCGS does not incorporate loop stop valves, this question is not applicable.

Q005.2.2 Provide a discussion of the consequences of inadvertent overpressurization resulting from a malfunction or operator error when the reactor coolant system is water-solid during startup or shutdown. The discussion should include consideration of the pressure-temperature operating limitations on the reactor vessel to protect against brittle fracture. In addition, discuss any design provisions that will be incorporated into the facility design to prevent overpressurization incidents that would exceed allowable pressures in this particular plant condition.

RESPONSE

Refer to Section 5.2.2.

Q005.2.7 Discuss the ability to assure that the operational capability of the valves that are required to function in the short and long term LOCA modes of ECCS operation are not impaired by potential crystallization of boric acid solutions on the valve stem due to leakage. Appropriate methods may include the ability to detect individual valve stem leakoff or periodic operational testing of the valves.

RESPONSE

Refer to Section 6.3.2.2.

Q005.3 Justify the fouling factor resistance specified in Section 5.5.2.3.1. Correct the difference between Section 5.5.2.3.1 and Table 5.5-3 with regard to the fouling factor.

RESPONSE

The fouling factor is discussed in Section 5.4.2.5.1 and is consistent with the value reported in Table 5.4-3.

Q005.4 Provide pressurizer relief and safety valve capacities when discharging water liquid.

RESPONSE

Refer to Section 5.4.13.2.

Q006.1 Item 6.3.2.11 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (Revision 1, October 1972) indicates the need to distinguish between true redundancy incorporated in a system and multiple components. To complement the SAR discussions in this regard, provide a summary of a systematic core cooling functional analysis of components required over the complete range of coolant pipe break inside the containment. The summary should be shown in the form of simple block diagrams beginning with the event (pipe break), branching out to the various possible sequences for the different size breaks, continuing through initial core cooling and ending with extended to long-term core cooling. When complete, the diagram should clearly identify each safety system required to function to cool the core for all coolant pipe breaks inside the containment during any plant operating state. The attached Figure 6-1 is provided as a guide.

RESPONSE

System reliability of the ECCS, including a discussion of redundancy compliance with the single failure criteria, is provided in Section 6.3.2.5. Functioning of the various ECCS components for various accidents, including large and small LOCAs, is discussed in Section 6.3.3. The actual LOCA analyses are discussed in Section 6.2 and 15.6.5.

Also refer to the Response to Question 015.0(1).

Q006.2 For each engineered safety feature identified in Question 6.1, list the auxiliaries required for its operation.

RESPONSE

Refer to Section 6.3.2.2 and the Response to Question 015.0(1).

Q010.01 Describe the device located on the suction side of the auxiliary feedwater pumps. This item is identified as SS001, SS002, and SS003 on Figure 10.4-9.

RESPONSE

The P&ID legend is provided on Figure 1.1-1.

Q015.0(1) For each transient and accident analyzed in Chapter 15, provide the following information:

(1) The step-by-step sequence of events from event initiation to the final stabilized condition. This listing should identify each significant occurrence on a time scale, including for example: flux monitor trip, insertion of control rods begin, primary coolant pressure reaches safety valve set point, safety valves open, safety valves close, containment isolation signal initiated, containment isolated, etc. All required operator actions should also be identified.

(2) The extent to which normally operating plant instrumentation and controls are assumed to function.

(3) The extent to which plant and reactor protection systems are required to function.

(4) The credit taken for the functioning of normally operating plant systems.

(5) The operation of engineered safety systems that is required.

RESPONSE

The sequence of events listed for each transient is provided in Tables in Chapter 15.0. The assumptions for instrumentation, controls, protection systems, and ESF systems are described for each transient analyzed in Chapter 15.0.

Figures of the step-by-step sequence of events for each transient are also provided in Chapter 15.0.

Q015.0(2) Section 15.2.4 of RESAR-3 UNCONTROLLED BORON DILUTION, analyzes the effects of a dilution at power. The analysis discusses the causes of the incident, and the automatic actions of the Reactor Protection System and the manual actions prompted by alarms and instrumentation that would mitigate the consequences of the accident.

However, there is a possible situation, involving the loss of offsite power, where a dilution incident may not be as readily apparent as that described in Section 15.2.4 and where no automatic Reactor Protection System action is available.

In order to assess the potential severity of a dilution accident after a loss of offsite power, provide the results of an analysis that assumes the anticipated equipment configurations in normal use prior to the event that results in the most severe consequences. The analysis should include a dilution operation in progress with the Chemical and Volume Control System mode selector switch being in the DILUTE position (or ALTERNATE DILUTE mode). The loss of offsite power is then assumed to occur with the minimum shutdown reactivity insertion due to control rods. Both diesel generators start and sequence the loss of offsite power loads.

The concerns are that the charging pumps again automatically start running after being loaded to the diesel generators and from electrical schematics of control circuits for the reactor makeup water pumps, that the reactor makeup water pumps would also again automatically start with the mode selector switch in DILUTE. Therefore, a dilution of the Reactor Coolant System is again in progress which could potentially result in a return to critical.

If the reactor makeup water batch integrator is assumed to malfunction by not automatically cutting off flow at the pre-selected value, provide the time available for manual action before the total shutdown margin is lost due to this dilution. If operator action is to be prompted by alarms, describe the features that will alert the operator to this specific action at a time when alarms from many plant systems are occurring simultaneously.

RESPONSE

This question is not applicable to WCGS since the reactor makeup water pumps cannot be supplied by the emergency diesel generators.

Q031.1 (3.10) Section 3.9.1.2 of RESAR-3 states that dynamic testing procedures concerning Westinghouse supplied safety-related mechanical equipment will be provided in the applicant's FSAR. It is our position that as a minimum you commit to conduct a seismic qualification program to conform to the criteria as contained in Attachment A. State your intent to employ the criteria as contained in Attachment A for all Westinghouse Category I mechanical equipment in order to confirm the functional operability of such equipment during and after a seismic event up to and including the SSE.

RESPONSE

Refer to Section 3.9(N).2.2.

Q031.2 (3.10) Section 3.9.2.4.1 of RESAR-3 states that the pump motor and vital auxiliary electrical equipment will be qualified by meeting the requirements of IEEE Standard 344-1971. Since the standard has undergone a major revision, state your intent to meet the requirements of the 1975 version of IEEE Standard 344. IEEE Standard 344-1975 includes requirements which are applicable to all plants with C.P. applications docketed after October 1972.

RESPONSE

Refer to Section 3.9(N).3.2.

Q031.3 (3.10) The seismic qualification criteria for electrical equipment as stated in Section 3.10 of the proposed Amendment 6 to RESAR-3 is not completely acceptable because it is only applicable to certain specific conditions when single frequency input to an individual axis is justifiable. A broader criterion to account for overall considerations should be provided. The major concern is the possible directional coupling and the concurrent multi-mode response. An acceptable response is to conduct a seismic qualification program as recommended by the 1975 version of IEEE-344 Standard. State your intent to use this recommended criteria.

RESPONSE

Refer to Section 3.10(N).

Q031.4 (3.11) The lists of safety-related equipment and components provided in Section 3.11.1 of RESAR-3 are not complete. Identify all individual components and complete the lists.

RESPONSE

Refer to Section 3.11(N).

Q031.5 (3.11) Section 3.11.2 of RESAR-3 does not give a complete and acceptable description of the qualification tests and analyses for each type of safety-related equipment and component. Provide this information for each item.

RESPONSE

Refer to Section 3.11(N).

Q031.6 (3.11) RESAR-3 Section 7.1.2.5. Describe how your design complies with IEEE Standard 323-1971, or IEEE Standard 323-1974, for all applications for which the construction permit safety evaluation report was issued July 1, 1974 or later. Identify and justify all exceptions.

RESPONSE

Refer to Section 3.11(N).

Q031.7 (7.1) In accordance with the implementation dates (noted in parentheses) and as they apply to your application, describe the extent to which the recommendations of the following regulatory guides will be met. Identify and justify any exception.

Regulatory Guide 1.22 (Safety Guide 22), "Periodic Testing of Protection System Actuation Functions" (Guide dated 2/17/72)

Regulatory Guide 1.29, "Seismic Design Classifications;" (Revision 1 dated August 1973)

Regulatory Guide 1.30 (Safety Guide 30), "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment;" (Guide dated August 11, 1972)

Regulatory Guide 1.40, "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants;" (Guide dated 3/16/73)

Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems;" (Guide dated May 1973)

Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems;" (Guide dated June 1973)

Regulatory Guide 1.62, "Manual Initiation of Protective Actions;" (Guide dated October 1973)

Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Water Cooled Nuclear Power Plants;" (Guide dated October 1973)

Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors;" (Guide dated November 1973)

Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants;" (Guide dated January 1974)

Regulatory Guide 1.75, "Physical Independence of Electric Systems." The physical identification of safety-related equipment should also be addressed in this section; (Guide dated February 1974)

Regulatory Guide 1.80, "Preoperational Testing of Instrument Air Systems;" (Guide dated June 1974) and

Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants." (Applicable to all plants with an SER issued after July 1, 1974).

RESPONSE

Refer to Appendix 3A.

Q031.8(1) (7.1) Provide a discussion and the results of an analysis showing how your design of the test and calibration features of the safety systems meets the requirements of Section 4.10 of IEEE Std 279-1971.

RESPONSE

Refer to Sections 7.1.2.5.2, 7.1.2.6.2, and 7.3.8.2 item (5) and Figures 7.3-2 and 7.3-3.

Q031.8(2) (7.2) Based on Figure 7.2-1, Sheet 7 of 17, of RESAR-3 we have concluded that the proposed design for the steamline differential pressure circuits does not conform to the requirements of IEEE Standard 279-1971. Specifically, during operation with a loop isolated, the logic for the operable steamlines is effectively changed to 2-out-of-2 which does not meet the single failure criterion. Our position is that in order to comply with IEEE Std 279-1971, the design should incorporate positive means of assuring that these circuits continue to meet the single failure criterion during operation with a coolant loop isolated. Discuss your intent to comply with this position and describe the necessary design changes, or justify any exceptions by discussing your reasons for concluding that such exceptions are in accordance with the requirements of IEEE Standard 279-1971. In addition as committed on Page 7.2-30 of RESAR-3, provide the results of an analysis that will determine whether automatic tripping of the steamline differential pressure bistables is required for N-1 loops operating.

RESPONSE

Refer to Figure 7.2-1 (Sheet 7) and Table 7.3-13.

Q031.9 (3.7.2) RESAR-3 Section 7.2.1.1.2(1)(d) and Figure 7.2-1 Sheet 3 address a power range high neutron flux rate "Positive" trip. This trip is used as protection against a rod ejection accident. The referenced Westinghouse Topical Report WCAP-7380-L (pages 2-8 and 3-12) provides a diagram and a description for the "Negative" flux rate trip but does not provide for the "Positive" flux rate trip. Provide a description and diagram covering "Positive" flux rate trip.

RESPONSE

WCAP-7380-L was replaced with WCAP-8255.

Refer to Section 7.2.4.

Q031.10 (7.2) The reactor trip system contains logic circuits that can initiate trips for the purpose of anticipating the approach to a limiting condition for operation. Specifically, these reactor trips are:

(1) Generation of a reactor trip by tripping the main coolant pump breakers,

(2) Generation of a reactor trip by tripping the turbine,

(3) Generation of reactor trip by underfrequency conditions on reactor coolant pump bus, and

(4) Generation of reactor trip by undervoltage conditions on reactor coolant pump bus.

Our position requires that all inputs to the reactor trip system be designed to meet IEEE Standard 279-1971, with an exception for anticipatory trips (trips not required for safety actions in the accident analysis - Chapter 15). The exception is that sensors for anticipatory trips are not required to be located in a qualified seismic Category I structure. Discuss your intent to comply with this position or justify any exceptions you may have in this regard. Your response should include a discussion of the testability of these circuits while the reactor is at power.

RESPONSE

(1) Refer to Section 7.2.1.1.2, item d.2.

(2) Refer to Section 7.2.1.1.2, item f.

(3) Refer to Section 7.2.1.1.2, item d.3.

(4) Refer to Section 7.2.1.1.2, item d.3.

Q031.11 (7.2, 7.3)

Testing of the reactor trip system and the engineered safety feature actuation system to verify that the "systems" response times are equal to or less than the values assumed in the accident analysis is discussed on Page 7.1-19, 7.2-24, and 7.3-13 of RESAR-3. In addition to the proposed response time testing during preoperational start-up testing and following the replacement of a component that affects response time, our position requires that these systems be designed to permit periodic verification that the response times are within the values assumed in the accident analysis. Discuss your intent to comply with this position or justify any exceptions.

It is stated in RESAR-3 on Page 7.3-26 that the response time specified in Paragraph 4.1 of IEEE Standard 338-1971 is not checked periodically as is the setpoint accuracy. Provide justification for the exception to this requirement.

RESPONSE

Refer to Section 7.1.2.6.2.

Q031.12 (7.3)

With regard to the motor operated accumulator isolation valves, we require that the proposed design include the following features in order to conform to the requirements of IEEE Std 279-1971:

(1) Automatic opening of the accumulator valves when either (a) the primary coolant system pressure exceeds a preselected value (to be specified in the Technical Specifications) or (b) a safety injection signal has been initiated. Both signals shall be provided to the valves.

(2) Visual indication in the control room of the open or closed status of the valve, actuated by sensors on the valve.

(3) An audible alarm, independent of Item (2), that is actuated by a sensor on the valve when the valve is not in the fully open position.

(4) Utilization of a safety injection signal to automatically remove (override) any bypass feature that may be provided to allow an isolation valve to be closed for short periods of time when the reactor coolant system is at pressure (in accordance with the provisions of the proposed Technical Specifications). Discuss your intent to comply with these requirements or justify any exceptions to these requirements.

RESPONSE

Refer to Section 7.6.4. and Figure 7.2-1 (Sheet 6).

Q031.13 (7.3) Based on the information provided in Section 7.3 of RESAR-3, we conclude that the proposed design for manual initiation of steam line isolation does not conform with the requirements of Section 4.17 of IEEE Standard 279-1971. In addition, there is not sufficient information on the design provision for manual initiation of containment isolation and containment depressurization to determine whether these functions are designed in accordance with Section 4.17 of IEEE Standard 279-1971. Our position is that a design which meets the following is an acceptable means of meeting the requirements of Section 4.17 of IEEE Standard 279-1971:

(1) Means should be provided for manual initiation of each protective action (e.g., reactor trip, containment isolation) at the system level, regardless of whether or not means are also provided to initiate the protective action at the component or channel level (e.g., individual control rod, individual isolation valve).

(2) Manual initiation of a protective action at the system level should perform all actions performed by automatic initiation such as starting auxiliary or supporting systems, sending signals to appropriate valves to assure their correct position, and providing the required action-sequencing functions and interlocks.

(3) The switches for manual initiation of protective actions at the system level should be located in the control room and be easily accessible to the operator so that action can be taken in an expeditious manner.

(4) The amount of equipment common to both manual and automatic initiation should be kept to a minimum. It is preferable to limit such common equipment to the final actuation devices and the actuated equipment. However, action-sequencing functions and interlocks (of Position 2) associated with the final actuation devices and actual equipment may be common providing individual manual initiation at the component or channel level is provided in the control room. No single failure within the manual, automatic, or common portions of the protection system should prevent initiation of protective action by manual or automatic means.

(5) Manual initiation of protective actions should depend on the operation of a minimum of equipment consistent with 1, 2, 3, and 4 above.

(6) Manual initiation of protective action at the system level should be so designed that once initiated, it will go to completion as required in Section 4.16 of IEEE Standard 279-1971.

Discuss your intent to comply with this position or justify any exceptions by discussing your reasons for concluding that such exceptions are in accordance with the requirements of IEE Standard 279-1971.

RESPONSE

Refer to Section 7.3.8.2, item b.7.

Q031.14 (7.4) General Design Criterion 37 requires, in part, that the emergency core cooling system be designed to permit testing the operability of the system as a whole. On Page 7.3-26 of RESAR-3, it is stated that the safety injection and residual heat removal pumps are made inoperable during the system tests. Our position is that in order to comply with the requirements of Criterion 37, these pumps must be included in the system test. Discuss your intent to comply with this position or justify any exception.

RESPONSE

Refer to Section 6.3.4.2.

Q031.15 (7.3, 6.3) Section 6.3.5.1 of RESAR-3 states that only "one temperature detector which provides heater control for the immersion heater, control room alarm and control room indication" is provided for the boron injection surge tank. Provide the results of an analysis which addresses the effect of a single failure in this system. This analysis should include possible boron dilution during recirculation. Also, it is our position that the monitoring system for the boron injection system meet IEEE Standard 279-1971. Discuss your intent to comply with this position or justify any exceptions you may have in this regard.

RESPONSE

Refer to Section 6.3.2.2.

Q031.16 (7.3.1) The description of the Emergency Safety Feature systems provided in Section 7.3.1 of RESAR-3 is incomplete in that it does not provide all of the information requested in Section 7.3.1 of the Standard Format for those safety-related systems, interfaces and components supplied by the applicant which match with the RESAR-3 scope systems. Provide all of the descriptive and design basis information requested in the Standard Format for these systems. In addition, provide the results of an analysis, as requested in Section 7.3.2 of the Standard Format, to demonstrate how the requirements of the General Design Criteria and IEEE Standard 279-1971 are satisfied and the extent to which the recommendations of applicable Regulatory Guides are satisfied. Identify and justify each exception.

RESPONSE

Refer to Section 7.3.8.

Q031.17 (7.3.1) Provide analyses showing that no adverse effects will occur or a discussion of such adverse effects that could occur as a result of power interruption to the Engineered Safety Features Actuation System at any time following the onset of a LOCA or other accident conditions in the plant.

RESPONSE

Refer to Section 7.3.

Q031.18 (7.4, 15.3.6) General Design Criterion 25 requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded from an accidental withdrawal of a single rod control cluster assembly (not ejection). In the accident analysis, presented in Section 15.3.6 of RESAR, it is stated that "no single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single rod control cluster assembly." However, Chapter 7.0 does not describe how the design prevents such an occurrence. Provide a detailed description of the control circuitry and discuss how the design meets the requirements of Criterion 25.

RESPONSE

Refer to Section 7.7.2.2 and Figure 7.7-15.

Q031.19 (7.4, 7.5 7.6) Provide a discussion which supplements those in Sections 7.4, 7.5 and 7.6 of RESAR-3 and which addresses the Standard Format information requirements for the safe shutdown systems, the safety-related display instrumentation and other safety systems and equipment outside the RESAR-3 scope which are assumed in the RESAR-3 and the PSAR Chapter 15 accident analyses.

RESPONSE

The safety-related systems are identified in Section 7.1.1. The safe shutdown safety-related system and other safety-related systems are discussed in Sections 7.4, 7.5, and 7.6.

Q031.20 (7.6.2) In addition to the design features discussed in Section 7.6.2 of RESAR-3, it is our position that the design of the RHR isolation valves satisfy the following:

(1) The interlocks shall utilize diverse equipment, and

(2) The interlocks shall be designed in accordance with the intent of IEEE Standard 279-1971.

The information presented in Section 7.6.2 of RESAR-3 does not address the requirements for diverse equipment and describes a degree of testability that conflicts with the requirements of IEEE Standard 1971. In addition, it is stated that the position indications for the RHR valves differ from those for the accumulator isolation valves but these differences are not identified. Discuss your intent to comply with the requirements that the design shall utilize diverse equipment and shall include complete on-line test capability without opening the isolation valves, or justify any exceptions. In addition, identify the differences in the position indications provided for the RHR valves compared to the accumulator valves and discuss the reasons for the differences.

RESPONSE

Refer to Section 5.4.7.

Q040.01 Figure 8.3-1 shows a "hold" symbol next to MCC PG 12J. Explain.

RESPONSE

See revised Figure 8.3-1.

Q040.02 Figure 8.3-2 has several loads listed as "later." Indicate the status of these loads.

RESPONSE

See revised Figure 8.3-2.

Q110.01 (3.10(B).2)	Section 3.10(B).2 addresses only Bechtel's scope of supply. Discuss your compliance with IEEE 344, 1975 and Regulatory Guide 1.100 for equipment outside Bechtel's scope of supply.

RESPONSE

Section 3.10 is presented in two parts: 3.10(B) and 3.10(N). Section 3.10(N) contains discussions on the compliance of the NSSS (Westinghouse) equipment to IEEE-344, 1975 and Regulatory Guide 1.100. All equipment subject to Regulatory Guide 1.100 is discussed in Section 3.10(B) or Section 3.10(N).

Q123.01 Identify whether SA-540 Class 1 or 2 material was used for closure bolting in the reactor coolant pumps. If SA-540 Class 1 or 2 materials were used for closure bolting in reactor coolant pumps, demonstrate the generic adequacy of the fracture toughness and demonstrate compliance with Paragraph I.C of Appendix G, to 10 CFR Part 50.

RESPONSE

SA-540 Class 1 or 2 material was not used for closure bolting in the reactor coolant pumps for WCGS. See Table 5.2-2.

Q123.02 Indicate whether the individuals performing the fracture toughness tests were qualified by training and experience and whether their competency was demonstrated in accordance with a written procedure. If the above information cannot be provided, state why the information cannot be provided and identify why the method used for qualifying individuals is equivalent to those of Paragraph III.B.4 Appendix G, 10 CFR Part 50.

RESPONSE

See Section 5.2.3.3.1.

Q123.03 Duplicate questions were received by SNUPPS and WCGS. See Q251.1.

RESPONSE

See Response to Q251.1

Q123.04 Duplicate questions were received by SNUPPS and WCGS. See Q251.2.

RESPONSE

See Response to Q251.2

Q123.05 Revise the FSAR to indicate that the conclusions of Westinghouse Topical Report WCAP 9292 are applicable to Wolf Creek SA-533 Grade A, Class 2 steel and SA 508 Class 2a steels.

RESPONSE

Refer to Section 5.2.3.3.1.

Q123.06 Duplicate questions were received by SNUPPS and WCGS. See Q251.3.

RESPONSE

See response to Q251.3.

Q123.07 Duplicate questions were received by SNUPPS and WCGS. See Q251.4.

RESPONSE

See response to Q251.4.

Q123.08 Duplicate questions were received by SNUPPS and WCGS. See Q251.5.

RESPONSE

See response to Q251.5.

Q123.09 Duplicate questions were received by SNUPPS and WCGS. See Q251.6.

RESPONSE

See response to Q251.6.

Q123.10 Duplicate questions were received by SNUPPS and WCGS. See Q251.7.

RESPONSE

See response to Q251.7.

Q123.11 Submit for review an inservice inspection program for the pump flywheels which complies with Paragraph C.4l of Safety Guide 14, October 27, 1971.

RESPONSE

See Appendix 3A.

Q210.1 Duplicate questions were received by SNUPPS and WCGS. See Q110.01.

RESPONSE

See Response to Q110.01.

Q210.2 The applicant states that all circumferential breaks in the RCS piping are assumed to result in a limited separation such that the maximum flow area is less than a full break area. The applicant must provide the design information assumed for each location where limited break areas are postulated including gap size, restraint stiffness, blowdown force, and maximum restraint deflection. The results of the time-history analysis (if used) should include the break area vs. time and mass flux rate vs. time which were used to calculate the subcompartment pressurization.

In addition, all restraint locations on the RCS piping must be shown.

RESPONSE

Refer to revised Sections 3.6.2 and 5.4.14.

Q210.3 In Section 1.8 of the Callaway SER (NUREG-0830), the staff identified a confirmatory item regarding the testing of pressure isolation valves. In Section 3.9.6 of the SER, the staff stated that the applicants have addressed the leak testing of only those check valves with an Event V configuration which form an interface between RCS pressure and low pressure coolant injection systems. The applicant's response for the Event V configuration is documented in a letter from N. Petrick to H. Denton dated September 11, 1981. However, the SER also stated that other low pressure interfacing systems exist with valve configurations whose failure could lead to an intersystem LOCA. These other systems include the accumulator discharge check valves, the boron injection system pressure isolation valves, and the motor operated valves in the RHR system. The SER stated, as a confirmatory item, that the staff will require that the leak-tight integrity of the pressure isolation valves in the above systems be verified by testing.

In order to complete the confirmatory item, it will be necessary for the applicants to identify all pressure isolation valves that will be included in their leak test program. The staff requires that these valves be included in the Callaway and Wolf Creek Technical Specifications. Limiting conditions for operation which will require corrective action and surveillance requirements which state the testing frequency should also be provided in the Technical Specifications. The applications should also submit four sets of Piping and Instrumentation Drawings (P&ID) for each system containing the pressure isolation valves to be tested. After reviewing the list of pressure isolation valves and provided we find it acceptably complete, we will consider the confirmatory item completed.

It should be emphasized that a proposed maximum allowable leakage limit of 10 gpm is not acceptable to the staff. The staff will require a maximum allowable leakage limit of 1.0 gpm in the Callaway and Wolf Creek Technical Specifications unless adequate justification is made for an exception.

RESPONSE

See the Technical Specifications and Figures 5.1-1, 5.4-7, and 6.3-1.

Q220.1 The staff has determined that Section 3.7(B).4.1 of the SNUPPS FSAR does not comply with the intent of R.G. 1.12, Rev. 1, as it claims. Nevertheless, it does comply, to a greater extent although not fully, with the positions of R.G. 1.12, Proposed Rev. 2, than that of R.G. 1.12, Rev. 1. The staff would accept that section of the FSAR if it is revised to comply with the positions of R.G. 1.12, Proposed Rev. 2, July, 1981.

<u>RESPONSE</u>

See Section 3.7(B).4.1.

Q220.2 Provide a discussion on how major cable tray test results were used in arriving at the 20% modal damping. The discussion should assure consistency of observed data and calculations used.

<u>RESPONSE</u>

See Section 3.7(B).3.16.

Q220.3 Why was cable tray test input loading applied at a 45 degree angle instead of simultaneous horizontal and vertical load input? What are the implications of this testing method upon the validity of the recommended 20% damping (e.g., with respect to statistical independency requirements of different directional inputs)?

<u>RESPONSE</u>

See Section 3.7(B).3.16.

Q220.4 Will sprayed-on fireproofing affect cable friction and thus the damping ratios?

<u>RESPONSE</u>

See Section 9.5.1.2.2.3.

Q220.5 The cable tray test conditions do not reflect the actual physical site situation. Provide the rationale for extending the test results to the actual design which is different from the test configuration.

RESPONSE

See Section 3.7(B).3.16.

Q220.6 Specify different conditions under which different modal damping ratios ranging from 7-20% are used. (cable tray)

RESPONSE

See Section 3.7(B).3.16.

Q220.7 It appears that the scope of the cable tray test and the number of tests may not support direct extension to SNUPPS (the appropriate project) cable tray design. Justify that the scope of test conducted is adequate for direct design application.

RESPONSE

See Section 3.7(B) 3.16.

Q220.8 Justify the use of 7% critical damping for conduit supports for all seismic input levels.

RESPONSE

See Section 3.7(B).3.16.

Q220.9 On Page 4 in last paragraph you stated that the method was selected in compliance with Standard Review Plan (SRP). Indicate which version of SRP you have referred to.

RESPONSE

See revised Section 3C.1.2.2.1.

Q220.10 The second sentence on top of Page 5 implies that the original FLUSH Analysis is unconservative and unrealistic. Clarify this statement.

RESPONSE

There is no implication that FLUSH results are unconservative. It is stated that a fixed base analysis is more conservative but still realistic when compared to the FLUSH results.

Q220.11 Under item C on Page 5, you stated that the presence of the soil surrounding the embedded portion of the structure was conservatively omitted. However, in staff's opinion your omission of the soil may result in a frequency shift and may, therefore, not be conservative. Your response to this staff's concern is requested.

RESPONSE

See revised Section 3C.1.2.2.1.

Q220.12 In the results of analyses for both fixed base and using FLUSH, there is substantial shift of maximum response. It is requested that response spectra enveloping the results of two analyses should be used unless your justification for not doing so is provided.

RESPONSE

See revised Section 3C.1.2.2.1.

Q220.13 On Page 9 in the second paragraph you indicated the consideration of torsional effects. Describe in detail how the torsional effects have been considered in the analysis.

RESPONSE

See revised Section 3C.1.2.2.2.

Q230.1 Provide a figure to illustrate the geographic regions used in the probability calculations discussed on FSAR Page 2.5-144.

RESPONSE

Figure 2.5-75 of the USAR illustrates the geographic regions used in the probability calculations.

Q230.2 Provide figure similar to FSAR Figure 2.5-82 comparing the SSE and (a) the scaled response spectra discussed on Pages 2.5-148 to 2.5-149 and (b) Nuttli's proposed spectra discussed on Page 2.5-149.

RESPONSE

See revised Section 2.5.2.5.

Q230.3 Current Staff Practice is to approach the development of response spectra by performing statistical analyses on the strong motion records for sites with similar foundation conditions. (See for example, Lawrence Livermore Laboratory, 1979, Draft, Seismic Hazard Analysis: Site Specific Response Spectra Results). Estimate the magnitudes of (a) the maximum random earthquake near the site and (b) the maximum event associated with the Nemaha Uplift. Accordingly, estimate the ground motion at the Wolf Creek site assuming (a) the maximum random event less than 25 km from the site, and (b) the maximum event associated with the Nemaha Uplift about 50 miles from the site.

Select response spectra from accelerograms for recording sites with foundation conditions similar to Wolf Creek. Choose those events that are within one-half the estimated magnitudes. For the data set compute 50 and 84 percentiles for the response spectra assuming the spectral ordinates are log normally distributed. On a plot similar to FSAR Figure 2.5-82 compare these spectra to the SSE.

RESPONSE

See revised Section 2.5.2.6.

Q230.4 Discuss the following recent studies and their significance to the Wolf Creek site:

1) Yarger, H. L., 1981, Aeromagnetic Survey of Kansas, EOS Transactions, v. 62, n. 17, 173-178.

2) Steeples, D. W., and M. E. Bickford, 1981, Piggyback Drilling in Kansas: An Example for the Continental Scientific Drilling Program, EOS Transactions, v. 62, n. 18, 473-476.

RESPONSE

See revised Section 2.5.1.1.5.1.19.

Q231.1 Prepare a new figure (or revise an existing figure) locating the noncapable shear zones, shear planes, and faults mapped at the site and described in the FSAR (Page 2.5-102). Also prepare a table listing the above deformations, the site location of the deformation, and the Dames & Moore report where the deformation mapping and description appears.

RESPONSE

See revised Section 2.5.1.2.4.1.

Q231.2 A number of lineaments, other than those numerically identified, are shown in Coffey County (the site county) on FSAR Figures 2.5-14a and 2.5-14b. Identify these unnumbered lineaments and present your interpretation of the origin/cause of each. Include in your discussion the relationship, if any, between each of the Coffey County lineaments (including those presently shown on FSAR Figures 2.5-14a and 2.5-14b) and the folds and faults identified on FSAR Figures 2.5-15 and 2.5-16.

RESPONSE

See revised Section 2.5.1.1.5.1.18.

Q231.3 Expand the LANDSAT lineament presentation (Revision 4, July 1981) to include a discussion of the relationship between the lineaments discussed, folds and faults (FSAR Figures 2.5-15 and 2.5-16), Precambrian surface folds and faults (FSAR Figure 2.5-14b), and earthquake epicenters.

RESPONSE

See revised Section 2.5.1.1.5.1.18.

Q231.4 Please provide a copy of the Dames & Moore report(s) discussing and portraying the Saddle Dam IV faulting. These reports are referenced in the D & M Second Interim Report of July 1979 (Dames & Moore, 1977; 1978a). Also provide a copy of the report(s) which includes the geologic map (and accompanying description) of the Drum Building excavation.

RESPONSE

Saddle Dam IV is described in Section 2.5.6.4.1.2.

Q231.5 Discuss the following recent studies and their significance to the Wolf Creek site:

1) H. Yarger et al. 1981, Bouguer gravity map of Southeastern Kansas, Kansas Geological Survey, Open-File Report.

2) Steeples, D.W., 1981, Microearthquake network activities, Fiscal Year 1980, Kansas Geological Survey, Report to the Kansas City District Corps of Engineers.

3) Steeples, D.W., 1981, Structure of the Salina-Forest City interbasin boundary from seismic studies, Kansas Geological Survey, prepared for the W.H. McNutt Memorial Lecture Series.

RESPONSE

See revised Section 2.5.1.5.1.18.

Q240.0 HYDROLOGIC & GEOTECHNICAL ENGINEERING BRANCH

Q240.1 (2.4.10) In Section 2.4.10 you state that the ESWS screen house was designed to withstand a high water elevation of 1100.2 feet, which corresponds to the maximum wave runup elevation from a wave height of 5.0 feet, with a period of 3.3 seconds. Using the PMF water surface elevation of 1095 feet, the combined wind set-up and runup must have been 5.2 feet. The staff's independent analysis at the ESWS screenhouse shows the maximum runup including set-up is 6.60 feet resulting in a high water elevation of 1101.60 feet. Our analysis is based on the following assumptions: 1) an effective fetch of 2.1 miles, 2) average fetch depth of 34 feet, 3) over land windspeed of 40 mph adjusted for over-water (50 mph), and 4) average depth along the south side of the structure of 17.8 feet. Either justify your wave runup calculations or use the staff's estimates and discuss the effects of the resulting higher wave runup elevation on the ESWS screenhouse.

RESPONSE

See Section 2.4.10.

Q240.2 (2.4.11.3) Table 2.4-25. The natural evaporation used to evaluate cooling lake drawdown are data for Fall Reservoir. Provide geographical coordinates of Fall Reservoir location. Since evaporation is a micro-climatically dependent phenomenon, provide sufficient justification (i.e., similarity of meteorological variables - wind speed, vapor pressure, etc.) for using Fall Reservoir natural evaporation in the analysis of cooling lake evaporation.

RESPONSE

See Section 2.4.11.3.2.

Q240.3 (2.4.11.3) Table 2.4-27. Provide a detailed description of your procedure for calculating forced evaporation from the cooling lake as presented in Table 2.4-26. Accompany the description with an example calculation including all data required to perform the example calculation.

RESPONSE

See Section 2.4.11.3.2

Q240.4 (2.4.11.6) During the August 13, 1981 site visit, you indicated that concrete pads were placed on the bottom of the ultimate heat sink and essential service water intake canal, and that sedimentation rates would be monitored by divers. Please discuss details of sampling methods, locations and frequency. Also, provide details of dredging procedures to restore capacity if and when it is reduced below the required capacity.

RESPONSE

See Section 2.4.11.6.

Q240.5 (9.2.5.3) It is stated in Section 9.2.5.3 that the UHS dam embankment structure will withstand overflow conditions that would result if the main cooling lake were to be drawn down below the UHS dam crest elevation. Please provide the maximum expected overflow velocities at the UHS dam during a postulated loss of the main cooling lake dam event and a discussion of the analysis including all pertinent assumptions. Provide evidence that the unprotected soil abutments of the UHS dam will not be eroded during the postulated event to the extent that there will be a loss of essential service water from behind the UHS dam.

Two cases were investigated to have an effect on the UHS for a postulated failure of the cooling lake main dam. Case I postulated the simultaneous failure of the cooling lake Main Dam and the Baffle Dike 'A' in front of the UHS. In Case II it was assumed that Baffle Dike 'A' fails subsequent to the main dam failure.

RESPONSE

See Section 9.2.5.3.

Q240.6 (9.2) Please provide a description of the trash collection and removal procedures from the service water and essential service trash racks.

RESPONSE

See Section 9.2.1.1.2 and 9.2.1.2.3.

Q240.7 What is the criteria used to determine which wells will be sealed and what is the status of well sealing?

RESPONSE

See Section 2.4.13.1.1.2.

Q240.8 Please provide a revised Figure 2.4-52 showing the cooling lake at its normal operating level and the WCGS property boundary superimposed on the well inventory within five miles of the plant.

RESPONSE

See Figure 2.4-52.

Q240.9 (2.4.2.3) Section 2.4.2.3.1 of the SNUPPS FSAR states that any rainfall in excess of design intensity (7.4 inches) will overflow the roof curb and the building walls to the site drainage system. Describe in more detail the roofs of safety related structures regarding their ability to pond water. State the maximum heights of any curbs or parapets on the roofs and the dimensions and locations of scuppers or other openings that will limit the depth of water during the PMP event.

RESPONSE

See Section 2.4.2.3.

Q240.10 State whether any permanent underdrains or ground water dewatering systems are installed, being constructed or planned at the plant site. If so, provide the information called for Branch Technical Position HMB/GSB, "Safety-Related Permanent Dewatering Systems."

RESPONSE

See Section 2.4.13.5.1.

Q241.1 In Figure 2.5-97a through 2.5-97e show the data points used in developing these curves. Also plot the mean and the standard deviation curves.

RESPONSE

See Section 2.5.4.7.

Q241.2 Provide a summary of the results of field density and moisture content tests used for quality control during construction of structural fill under and backfill around the Category I structures. Present the results as a statistical distribution plot or by other convenient method(s) to be able to verify that the specified compaction has been attained. Provide the above data for each type of fill separately for the Power Block Unit, the ESWS pumphouse, the ESWS discharge structure and the seismic Category I pipelines and electrical duct banks. NOTE: The ESWS Discharge Structure was removed from service after replacement of the ESWS underground piping.

RESPONSE

See Section 2.5.4.5.1.5.

Q241.3 Provide details of the six different types of backfill and the bedding materials used in the construction of ECCS seismic Category I piping and electrical duct banks including gradation and plasticity index requirements, and principal construction criteria.

RESPONSE

See Section 2.5.4.5.3.5.

Q241.4 For the ESWS discharge structure, submit drawings showing plans and typical cross-sections of the limits of excavation and types of fill and backfill materials. NOTE: The ESWS Discharge Structure was removed from service after replacement of the ESWS underground piping.

RESPONSE

See Section 2.5.4.5.4.

Q241.5 1) In Figure 2.5-47 show locations and limits of soft material, if any, that was replaced by competent material during construction.

2) For the ECCS pipeline, provide typical transverse cross section showing the excavation limits, pipe, bedding, and different kinds of backfill materials.

3) Provide typical longitudinal section and cross section details of excavation and backfill near the interface between the ECCS pipes and the structures.

4) What are the estimated total and differential settlements of the ECCS pipe and the structures at their interface due to both static and dynamic loads?

5) What is the estimated settlement of the ECCS piping due to both static and dynamic loads?

RESPONSE

See Section 2.5.4.10.3.1.

Q241.6 (2.5.4.7) Provide a copy of the Bechtel Topical Report BC-TOP-4A, referenced on Page 2.5-199 of the FSAR.

RESPONSE

Bechtel Topical Report, BC-TOP-4A, was approved by the NRC on October 31, 1974.

Q241.7 (2.5.4.10.1.3) Provide a plot of the magnitude and distribution of lateral earth and water pressures used in the design of subsurface walls and, on the same figure, plot the dynamic lateral pressures computed from the soil-structure interaction analyses due to the building and soil response under dynamic loading conditions. Provide such plots for the main powerblock structures, the ESWS pumphouse, and the ESWS discharge structure. NOTE: The ESWS Discharge Structure was removed from service after replacement of the ESWS underground piping.

RESPONSE

See Section 2.5.4.10.1.3.

Q241.8 (Figure 2.5-111) Revise FSAR Figure 2.5-111 to show the location of sections GG and HH.

RESPONSE

See USAR Figure 2.5-108 and 2.5-111.

Q241.9 (Figure 2.5-112) In Figure 2.5-112 show the following missing information:

1) The water levels and the piezometric surfaces used in the stability analyses for all conditions analyzed.

2) Show the minimum factor of safety and the corresponding critical sliding wedge.

RESPONSE

See Figure 2.5-112 of the USAR.

Q241.10 (Figure 2.5-113)

1) In Figure 2.5-113 show the following missing information:

   a) Subsurface soil profile and the soil parameters for each soil layer that were used in the slope stability analyses.

   b) Show the water levels and the piezometric surfaces used in the stability analyses for all conditions analyzed.

   c) Show the minimum factors of safety and the corresponding critical slip circles for each of the cases investigated.

2) Discuss the validity of using slip circle method of analysis, particularly for the side slopes of the pumphouse intake channel (3H:1V), considering that a) the hard rock layer is in the immediate vicinity of the toe of the slope, b) for the UHS slope you choose to use the sliding wedge method of analysis. Justify the validity of the slip circle method of analysis or investigate the stability of the slopes of the ESWS pumphouse intake channel using the sliding wedge method.

3) For the cross section presented in Figure 2.5-113 explain why the minimum factor of safety for the stability of (3H:1V) slope is higher than the minimum factor of safety for the stability of (5H:1V) slope.

RESPONSE

1) The information requested is shown on USAR Figure 2.5-113a through 2.5-113h. Section 2.5.5.2.2.2 had been revised to include a reference to these figures.

2) See Section 2.5.5.2.2.2.

3) See Section 2.5.5.2.2.2

Q241.11 (Figure 2.5-115) Show the critical slip circle and the corresponding minimum factor of safety for the cases investigated in the stability analyses presented on Figure 2.5-115. Also, correct Detail A that shows the fine filter layer between the coarse filter layer and the riprap layer.

RESPONSE

USAR Figures 2.5-115b through 2.5-115d show the critical slip circles and Factors of Safety for the cases investigated. Section 2.5.6.5.1.2 has been revised to include a reference to these figures. Detail A on Figure 2.5-115 (this was changed to USAR Figure 2.5-115a) has been corrected.

Q241.12 (2.5.6.8.4) Provide a description of the monitoring system that is being used to measure the movements of the UHS dam. Summarize the data collected to date and compare the results with the estimated movements of the UHS dam. Comment on the results of this comparison and its safety implication.

RESPONSE

See revised Section 2.5.6.8.4.

Q241.13 Provide a summary of the results of field density and moisture content tests performed in connection with quality control during construction of the UHS dam. Present the results as a statistical distribution plot or by other convenient method(s) to verify that the specified compaction has been attained.

Compare the compacted in-situ density and moisture content of the embankment fill with those of the test specimens from which the design strength parameters have been determined by laboratory testing. Based on the above comparison, comment on the validity of the physical and strength parameters used in the design.

RESPONSE

See revised Section 2.5.6.4.2.1.1.

Q241.14 Identify the local and federal agencies that have regulatory authority over the main dam, and the license or permit number(s); provide a brief description of the safety inspection program required and confirm your commitment to meet these requirements.

RESPONSE

See Section 2.5.6.8.1.

Q241.15 A seep was noticed in the grandular toe drain on the downstream side of the main dam during staff site visits in August and December 1981. At that time, the reservoir was not filled up to normal the operating level. This dam is a back-up structure for the safety-related UHS dam.

1) The possibility that the main dam embankment material may be a dispersive clay is of concern.

2) Provide a commitment to monitor the vertical and lateral deformation of the main dam and seepage through the dam during operation of the Nuclear Power Plant. Submit for review by the NRC details of the performance monitoring program presented in Section 2.5.6.8 of the FSAR.

3) Summarize the data collected to date and compare the results with estimated movements of the main dam. Comment on the results of this comparison and its safety implication.

RESPONSE

See Section 2.5.6.6.1.

Q241.16 The UHS dam embankment material was tested to determine the dispersive characteristics of the clay. The FSAR does not address this topic beyond the presentation of the laboratory test data. Provide the following:

1) Full details of your study, including any input from outside consultant, on this item.

2) Provide the test procedure, details of the data monitored and conclusions for the field test (filling only UHS pond) performed on the UHS dam.

3) Amend the FSAR to include the above information.

RESPONSE

1. See Section 2.5.6.4.1.4.1.14.

2. See Section 2.5.6.4.1.4.1.14.

3. See Section 2.5.6.4.1.4.1.14.

Q241.17 (2.5.4.5.1.5.12) Provide specification for the cohesive backfill material.

RESPONSE

See Section 2.5.4.5.1.5.1.2.

Q241.18 (2.5.5.2) Provide clear prints of Figures 2.5-108 and 2.5-111.

Show on Figure 2.5-108 the location of the sections analyzed for stability.

RESPONSE

See Section 2.5.5.2.2.2.

Q241.19 (2.5.5.2.2.1) Docket a write-up on the computer program used for the sliding wedge method of stability analysis. If you have not used a computer program, provide detailed write-up of the method of analysis.

RESPONSE

See Section 2.5.5.2.2.1.

Q241.20 (2.5.5.2)

1) What is the elevation of the water table for end-of-construction condition for UHS slope and Intake Channel Slopes? Is it el 1053.0 ft or el 1070.0 ft?

2) Justify using the water table elevation of 1070.0 ft rather than the normal cooling lake level of elevation 1087.0 ft for steady-state condition.

3) The drop in the water level for rapid drawdown condition should be from an initial elevation of 1087.0 ft to elevation of 1070.0 ft in the event of main dam failure, and to elevation of 1065.0 ft in the event of both main dam and UHS dam failure. Justify the water table elevations used in the stability analysis for rapid drawdown conditions presented in Figures 2.5-113d and 2.5-113h.

4) Revise Figures 2.5-113a through h to show the proper water levels and if required, revise the analysis to reflect the revised water table.

5) Provide analysis and factor of safety for the stability of the UHS slope (analyzed by Sliding Wedge Method) for the rapid drawdown condition.

6) Justify using total stress shear strength parameters for the residual soil in the analysis presented in Figure 2.5-113h.

Revise your analysis using effective stress strength parameters and proper water table.

7) Table 2.5-57 and analysis presented in Figures 2.5-113g and 2.5-113h are not compatible.

Revise Table 2.5-57.

RESPONSE

1. See Sections 2.5.5.2.2.1 and 2.5.5.2.2.2.

2. See Section 2.5.5.2.

3. Analyses have been presented in Section 2.5.5.2 for 5:1 slopes for rapid drawdown from elevations 1087 to 1070 ft. Drawdown to elevation 1065 ft would only occur if the UHS dam failed in which case there would be no water in the UHS and therefore a stability analysis is not needed.

4. See Figures 2.5-113a through h and Table 2.5-57.

5. Figure 2.5-112 has been revised to clarify these conditions.

6. Section 2.5.5.2 has been revised.

7. Figures 2.5-113a through h and Table 2.5-57 have been revised to reflect the revised analysis.

Q241.21 (2.5.5.2) The FSAR does not address the dynamic stability and liquefaction potential aspects of the UHS slopes and intake channel slopes. Amend the FSAR to include these items.

RESPONSE

See Subsections 2.5.5.2.3 and 2.5.5.2.4.

Q241.22

1) Provide settlement versus time plots for Category I structures based on data from the settlement monitoring program.

2) What are the maximum total and differential settlements measured to date and also expected in the future?

3) Compare the measured settlements with the anticipated settlements assumed in the analysis of these structures and their appurtenances, and evaluate the impact of any difference between the measured and anticipated settlements on the design and construction of these structures and appurtenances.

RESPONSE

See Section 2.5.4.10.1.2 for items 1 through 3.

Q241.23 Solution channels filled with clay were discovered in the Plattsmouth Limestone formation during geologic mapping of the UHS dam foundation excavation. This was not reported in the FSAR.

1) What was the areal extent and depth of these solution channels, and are there any continuous channels across the dam foundation?

2) How did you determine the presence or absence of these solution features within the limestone formations?

3) Was the soil in the solution cavities tested for the properties resistant to piping and for erosion under the design conditions?

4) Evaluate the effect of these solution channels on the safety of the UHS dam.

RESPONSE

1. A description of these features is provided in revised Section 2.5.1.2.5.3.

2. The subsurface exploration program for the UHS and the UHS dam are described in Section 2.5.6.2.1.

   A description of the extent and depth of the solution features observed in the UHS dam foundation is provided in Section 2.5.1.2.5.3.

3. No tests related to resistance to piping and erosion were performed on the material in the solution features. However, see Item 4 below.

4. The solution features discovered in the Plattsmouth Limestone during the mapping of the UHS dam foundation are discussed in Section 2.5.1.2.5.3.

Q241.24 Provide the following information on the UHS dam filling test:

1) What was the quantity of water pumped into the UHS dam during the 30-day monitoring period?

2) What was the quantity of water pumped from the downstream toe to maintain a water level of elevation 1955 feet?

3) What were the estimate seepages through the UHS dam and through the UHS dam foundations?

4) Compare the estimated vertical and lateral deformation of the UHS dam with "those measured during the filling and subsequent 30-day monitoring of the UHS dam." Evaluate the impact of any differences between the measured and estimated deformations on the safety of the UHS dam.

5) Provide a copy of the report "Final Report, Surveillance of Earthwork, UHS and UHS Dam" by Dames & Moore, 1981.

RESPONSE

1. See revised Section 2.5.6.8.4

2. See revised Section 2.5.6.8.4

3. See revised Section 2.5.6.8.4

4. See revised Section 2.5.6.8.4

5. A copy of the report was provided.

Q241.25 Provide copies of the following reports:

1) "Engineering Data Compilation for the Wolf Creek Lake" Sargent Lundy Report SL-3830

2) "Engineering Data Compilation for Water Control Structures at Wolf Creek Lake" Sargent and Lundy Report SL-3831

RESPONSE

The requested documents were provided to the NRC in letter KMLNRC 82-177, dated March 16, 1982.

Q241.26 The responses to the following inquires are the result of a meeting held with the NRC on March 19, 1982. These inquiries were never formally transmitted to KG&E by the NRC.

NRC Inquiry (1):

For the UHS dam (include riprap on top to elevation 1077.0, in your analysis).

a) pseudo - static; seismic coefficient 0.12, 0.15.

b) dynamic FEM analysis for SSRS.

RESPONSE

a. See Section 3C.1.2.3.1.

b. See section 3C.1.2.3.1.

NRC Inquiry (2):

UHS Slopes

a) pseudo - static analysis - seismic coefficient 0.15.

RESPONSE

a. See Section 3C.1.2.3.1.

NRC Inquiry (3):

Seismic Category I Buried Pipes and Electrical Duct Banks Comment on dynamic stability for SSRS loading.

RESPONSE

See Section 2.5.4.5.3.6.

Q251.1 To demonstrate compliance with the beltline material test requirements of Paragraph III.C.2 of Appendix G, 10 CFR Part 50:

a) Provide a schematic for the reactor vessel showing all welds, plates and/or forgings in the beltline. Welds should be identified by shop control number, weld procedure qualification number, the heat of filler metal, and type and batch of flux. Provide the chemical composition for these welds (particularly Cu, P, and S content).

b) Indicate the post-weld heat treatment used in the fabrication of the test welds.

c) Indicate the plates used to fabricate the test welds.

d) Indicate whether the test specimen for the longitudinal seams was removed from excess material and welds in the vessel shell course following completion of the longitudinal weld joint.

RESPONSE

See Figure 5.3-2, Table 5.3-7 and Section 5.3.1.1.

Q251.2 To demonstrate compliance with the fracture toughness requirements of Paragraph IV.A.1 of Appendix G, 10 CFR Part 50:

a) Provide the RT_{NDT} for all RCPB welds which may be limiting for operation of the reactor vessel.

b) Indicate whether there are any RCPB heat-affected zones which require CVN impact testing per paragraph NB-4335.2 of the 1977 ASME Code. Provide CVN impact test data for these heat-affected zones which may be limiting for operation of the reactor vessel.

c) Indicate that there are no ferritic RCPB base metals other than in vessels which require fracture toughness testing to NB-2300 of the ASME Code. If there are ferritic RCPB base metals other than in vessels which require fracture toughness testing to NB-2300 of the ASME Code, provide CVN impact and drop weight data for all materials which will be limiting for operation of the reactor vessel.

RESPONSE

See Section 5.3.1.5.

Q251.3 Provide actual pressure-temperature limits for Callaway Unit 1 (Wolf Creek) based upon the limiting fracture toughness of the reactor vessel material and the predicted shift in the adjusted reference temperature, RT_{NDT}, resulting from radiation damage. The pressure-temperature limits for the following conditions must be included in the technical specifications when they are submitted.

a) Preservice hydrostatic tests,

b) Inservice leak and hydrostatic tests,

c) Heatup and cooldown operations, and

d) Core operation.

RESPONSE

See the Technical Specifications.

Q251.4 Provide full CVN impact curves for each weld and plate in the beltline region. Provide the data in tabulated and graphical form.

RESPONSE

See Section 5.3.1.5 and Tables 5.3-8 through 11.

Q251.5 To demonstrate the surveillance capsule program complies with Paragraph II.C.3 of Appendix H:

a) Provide the withdrawal schedule for each capsule.

b) Provide the lead factors for each capsule.

c) Indicate the estimated reactor vessel end of life fluence at the 1/4 wall thickness as measured from the ID.

RESPONSE

See Section 5.3.1.6.

Q251.6 Identify the location of each material surveillance capsule and the materials in each capsule.

a) For each base metal and heat-affected zone surveillance specimen provide the specimen type, the orientation of the specimen relative to the principal rolling direction of the plate, the heat number, the component code number from which the sample was removed, the chemical composition especially the copper (Cu) and phosphorus (P) contents, the melting practice and the heat treatment received by the sample material.

b) For each weld metal surveillance specimen provide the weld identification from which the sample was removed, the weld wire type and heat identification, flux type and lot identification, weld process and heat treatment used for fabrication of the weld sample.

c) Provide a sketch which indicates the azimuthal location for each capsule relative to the reactor core.

RESPONSE

See Section 5.3.1.6.

Q251.7 Indicate the normal operating temperature of the flywheels and provide CVN impact and drop weight test data from each flywheel that indicates the RT_{NDT} of the flywheels are 100°F less than their normal operating temperatures.

RESPONSE

See Section 5.4.1.5.2.2.

Q260.0 QUALITY ASSURANCE BRANCH

Q260.1 (Table 17.2-3) Table 17.2-3 and its referenced Appendix 3A should incorporate the following:

Regulatory Guide	Rev.	Date	Appendix 3A	Table 17.2-3
1.8	1-R	5/77	OK	OK
1.26	2	6/75	Missing	Missing
1.29	3	9/78	Missing	Missing
1.30	-	8/72	OK	OK
1.33	2	2/78	OK	OK
1.37	-	3/73	OK	OK
1.38	2	5/77	OK	OK
1.39	2	9/77	OK	OK
1.58	1	9/80	8/73	8/73
1.64	2	6/76	OK	OK
1.74	-	2/74	OK	OK
1.88	2	10/76	OK	OK
1.94	1	4/76	Missing	OK
1.116	0-R	5/77	OK	OK
1.123	1	7/77	OK	OK
1.144	1	9/80	1/79	1/79
1.146	-	8/80	Missing	Missing

A commitment to 10 CFR 50.55a is also required.

The following is in reference to the KG&E discussion regarding the Regulatory Guide noted.

1.33 The discussion states that the recommendations of R.G. 1.33 are met through the specific ANSI daughter standards listed in Table 17.2-3. This could be construed to mean that the Regulatory Position of R.G. 1.33 is not met. Clarify.

1.38 The discussion states that KG&E may prescribe protective measures, in lieu of manufacturer's standards or minimum requirements. The standard says that the manufacturer's documented standard or minimum requirements shall be considered when classifying items, and the point of the discussion regarding this is not clear. Clarify.

1.39 The discussion states that KG&E procedures require general housekeeping practices to be maintained at the station during normal operations. Describe what is meant by "general housekeeping practices.

1.74 It is the staff position that certificates of conformance and certificates of compliance should be signed by a responsible party from the certifier's organization. Commit to meet this position or submit an alternative for our evaluation.

1.144 a) The first discussion paragraph regarding the classification of certain audit personnel as lead auditors implies that all KG&E auditors meet the requirements for lead auditors. This may require clarification based on commitment to R.G. 1.146.

b) The first sentence of the second discussion paragraph is unacceptable. The staff-position given in Section C.3b.(2) of R.G. 1.144 is a minimum requirement. More frequent audits, based on status and importance to safety, are acceptable. Clarify.

RESPONSE

See Table 17.2-3 and Appendix 3A.

Commitments regarding 10 CFR 50.55a are provided in Table 1.3-4.

Q260.2 Provide the qualification requirements for the Manager Quality Assurance. Section 17.2.1.4.1 states that the qualifications of the Manager Quality Assurance (Site) are at least equivalent to those specified in ANSI/ANS 3.1. Verify that this commitment is to the draft standard ANS 3.1 dated December 6, 1979, and identify the applicable part(s) of this draft standard.

RESPONSE

See Sections 17.2.1.4 and 17.2.1.4.1.

Q260.3 (Figure 13.1-1) Describe the significance of the dashed line from the QC Supervisor and Health Physicist on Figure 13.1-1. Provide the number of individuals planned to be assigned to the QC Supervisor shown on Figure 13.1-2.

RESPONSE

The revised Quality Organization is described in Section 13.1.2.4 and Figure 13.1-4. The Health Physicist is described in Section 13.1.2.2.4 and Figure 13.1-1.

Q260.4 (17.2.1.4 & 13.1.2.2) Provide a commitment that the Manager Quality Assurance, the Manager Quality Assurance (Site), and the QC Supervisor have not duties or reponsibilities unrelated to QA that would prevent their full attention to QA matters. Where is the Manager Quality Assurance (Site) located?

RESPONSE

See Section 17.2.1.4.

Q260.5 (17.2.2.3) Provide a commitment to notify NRC of changes (1) for review and acceptance in the accepted description of the FSAR QA program prior to implementation and (2) in organizational elements within 30 days after announcement.

RESPONSE

See Section 17.2.2.3.

Q260.6 FSAR Revision 1 deleted the statement that Table 3.2-1 of the Standard Plant FSAR is maintained current by the Manager Nuclear Services with changes to the table approved by the Manager Quality Assurance and Manager Nuclear Plant Engineering. Describe KG&E responsibilities regarding this table and discuss how these responsibilities are met. Also, it is not clear how Table 3.2-1 applies during the operations phase in regards to the column headed "Quality Assurance." While the Bechtel and Westinghouse QA programs were applicable during the design and construction phases, it is not clear how (or if) KG&E would use these programs during the operations phase. Clarify.

RESPONSE

See Section 17.2.2.2.

Q260.7 (17.2.2.4 & 17.2-1)	Item 2 on page 17.2-8 is headed "Operating Quality Assurance Program Manual." Although Table 17.2-1 is titled "Controlled Procedure Manuals," the Operating Quality Assurance Program Manual is not identified in the table. Clarify. Also discuss the Manager Quality Assurance's responsibility regarding this manual.

RESPONSE

The Operating Quality program previously described in the Operating Quality Assurance Program Manual has been replaced by certain Directives contained in the Wolf Creek Project Policy Manual. See Table 17.2-1a.

Q260.8 (17.2.2)	Section 17.2.0.3 indicates that computer codes are controlled by the OQAP. Describe how the QA program will be applied. Include a description of related organizational responsibilities for internal and external efforts.

RESPONSE

See revised Section 17.2.0.3.

Q260.9 (17.2.2)	Section 17.2.2.6 of the Wolf Creek FSAR discusses verification of QA program implementation through audits. Provide a commitment that KG&E management above the QA organization maintains frequent contact with the QA program through meetings and reports, including review of audit reports. Verify that in this way, and through preplanned and documented annual assessments, this management regularly assesses the scope, status, adequacy, and compliance of the QA program to 10 CFR Part 50 Appendix B.

RESPONSE

See Section 17.2.1.9.

Q260.10 (17.2.3)	The second sentence in Section 17.2.3.3 states that design changes shall be communicated to appropriate plant personnel when such changes may affect performance. Clarify that this means each individual's performance of his duties.

RESPONSE

See Section 17.2.3.7.

Q260.11 (17.2.3) Provide a commitment that action to correct errors found in design process and action to assure control of changes are documented.

RESPONSE

See Section 17.2.3.6.

Q260.12 (17.2.3) Clarify the first sentence of Section 17.2.3.3 which states: "Design requirements and changes thereto shall be...so that deviations from quality standards remain visible throughout the design process." (Underline added.)

RESPONSE

See Section 17.2.3.3.

Q260.13 (17.2.3) Section 17.2.3.5 indicates KG&E procedures will control design interfaces. Describe the controls.

RESPONSE

See Section 17.2.3.5.

Q260.14 (17.2.3) Section 17.2.3.6 of the Wolf Creek FSAR states: "Design verification shall be performed by personnel other than those who performed the original design and shall be accomplished prior to relying upon the component, system, or structure to perform its function." Concerning the personnel, provide a commitment that the verifier is qualified and is not directly responsible for the design or design change (i.e., neither the designer nor his immediate supervisor). Concerning the timing, provide a commitment that design verification is normally completed prior to release for procurement, manufacture, or installation or to another organization for use in other design activities. Where this timing cannot be met, justification for deferral should be documented and the unverified portion should be identified and controlled. Include such a commitment.

RESPONSE

See Section 17.2.3.6.

Q260.15 (17.2.3) In the area of design verification, clarify that procedures identify the responsibilities of the verifier, the areas and features to be verified, the pertinent considerations to be verified, and the documentation required. Also provide a commitment that specialized reviews are used when uniqueness or special design considerations warrant.

RESPONSE

See Sections 17.2.3.1 and 17.2.3.6.

Q260.16 (17.2.3) Clarify that design documents subject to procedural control include, but are not limited to, specifications, calculations, computer programs, system descriptions, SAR when used as a design document, and drawings including flow diagrams, piping, and instrument diagrams, control logic diagrams, electrical single line diagrams, structural systems for major facilities, site arrangements, and equipment locations.

RESPONSE

See Section 17.2.3.4.

Q260.17 (17.2.4) Provide a commitment that supplier QA programs are reviewed and found acceptable by KG&E's QA organization before initiation of activities affected by the program.

RESPONSE

See Section 17.2.4.5.

Q260.18 (17.2.4) Section 17.2.4.2 indicates KG&E's Quality Assurance Department is responsible for quality requirements for procurement. Verify that the QA Department review of procurement documents determines that the quality requirements are correctly stated, inspectable, and controllable; that there are adequate accept/reject criteria; and that the procurement documents have been prepared, reviewed, and approved in accordance with KG&E's QA program requirements.

RESPONSE

See Section 17.2.4.5.

Q260.19 (17.2.6)
Section 17.2.6.2 of the Wolf Creek FSAR identifies the types of documents which are controlled. Expand this list such that it includes the following:

a) Other design documents (e.g., calculations and analyses) including documents related to computer codes.

b) Instructions and procedures for such activities as fabrication, construction, modification, installation, test, and inspection.

c) As-built drawings.

d) Wolf Creek Project Policy Manual.

e) Wolf Creek Generating Station Procedures Manuals.

f) KG&E Procedures Manual.

g) FSAR.

h) Topical reports.

RESPONSE

See Section 17.2.6.2.

Q260.20 (17.2.6)
Discuss the role of the quality assurance organization in the review of and concurrence with documents under the control of the quality assurance program regarding the QA-related aspects.

RESPONSE

See Sections 17.2.2.2, 17.2.4.5, 17.2.6.6 and 17.2.7.7. Other review and approval activities conducted by the Quality Branch are described in the following Sections: 17.2.1.6, 17.2.2.5, 17.2.3.3, 17.2.4.4, 17.2.4.7, 17.2.4.11, 17.2.5.5, 17.2.7.2, 17.2.7.3, 17.2.7.6, 17.2.7.10, 17.2.9.2, 17.2.9.3, 17.2.10.2, 17.2.10.6, 17.2.15.2, 17.2.18.4, and 17.2.18.9.

Q260.21 (17.2.6)
Provide a commitment that the quality assurance organization reviews and concurs with instructions and procedures used for maintenance, modification, and inspection at Wolf Creek to determine,

a) The need for inspection, identification of inspection personnel, and documentation of inspection results.

b) That the necessary inspection requirements, methods, and acceptance criteria have been identified.

RESPONSE

See Section 17.2.6.6.

Q260.22 (17.2.7) Section 17.2.7.7 of the Wolf Creek FSAR addresses supplier monitoring in accordance with procedures. Verify that the procedures are documented, that they assure conformance to the purchase document requirements, that they identify organizational responsibilities, and that they specify the characteristics or processes to be witnessed, inspected, or verified, and accepted, the method of surveillance, and the documentation required. Clarify that the procedures are reviewed and approved by the quality assurance organization.

RESPONSE

See second paragraph of Section 17.2.7.7.

Q260.23 (17.2.7) Provide a commitment that the bases of supplier selection is documented and filed. Also clarify that when an LCVIP letter of confirmation or the CASE register is used to establish a supplier's qualification, the documentation will identify the "letter" or "audit" used.

RESPONSE

See Section 17.2.7.2.

Q260.24 (17.2.7) Provide a commitment that procurement of spare or replacement parts for safety-related structures, systems, and components is subject to present QA program controls, to applicable codes and standards, and to technical requirements equal to or better than the original technical requirements, or as required to preclude repetition of defects.

RESPONSE

See Section 17.2.4.11.

Q260.25 (17.2.7) Provide a commitment that suppliers' certificates of conformance are periodically evaluated by audits, independent inspections, or tests to assure they are valid.

RESPONSE

See Section 17.2.7.10.

Q260.26 (17.2.7) Section 17.2.7.6 states that the extent of acceptance methods and associated verification activities will vary as a function of the relative importance and complexity of the purchased item or service and the supplier's past performance. It is the staff's position that the extent of quality verification should also reflect the item's or service's importance to safety or relative safety importance.

Section 17.2.7.6 then goes on to state that procedures will provide for the acceptance of simple, off-the-shelf items based exclusively on receiving inspection with no quality verification documentation requirements. It is the staff's position that the involved design engineering organization and quality assurance organization should jointly determine the extent of inspection verification and the quality verification documentation requirements based on the item's end use.

Revise the FSAR to reflect this position.

RESPONSE

See Section 17.2.7.6.

Q260.27 Describe the involvement of KG&E's QA and QC organizations in the acceptance of items by post-installation test.

RESPONSE

See Section 17.2.7.6.

Q260.28 (17.2.7) Describe the involvement of KG&E's QA and QC organizations in the final acceptance of service.

RESPONSE

See Section 17.2.7.10.

Q260.29 (17.2.9) Describe the involvement of KG&E's QA and QC organizations in the control of special processes.

RESPONSE

See 17.2.9.2 and 17.2.9.3.

Q260.30 (17.2.9) Expand the list of processes given in Section 17.2.9.1 of the Wolf Creek FSAR so that the list is as complete as possible.

RESPONSE

See Section 17.2.9.1.

Q260.31 (17.2.9) Describe measures which assure the recording of evidence of acceptable accomplishment of special processes using only qualified procedures, equipment, and personnel.

RESPONSE

See Section 17.2.9.2.

Q260.32 (17.2.9.1) Identify the KG&E organization(s) responsible for qualifying special process equipment and for maintaining the qualification of such equipment. Discuss the records associated with qualifying special process equipment.

RESPONSE

See Section 17.2.9.2.

Q260.33 (17.2.10) It is not clear that KG&E personnel who perform inspections and process monitoring are part of the QC organization under the QC Supervisor. Clarify. Since QA personnel do not perform inspections and process monitoring, provide a commitment that procedures, personnel qualification criteria, and personnel independence from undue pressure of cost and schedule are reviewed and found acceptable by the QA organization prior to initiating the inspection or monitoring.

RESPONSE

See Section 17.2.10.

Q260.34 (17.2.10 & 17.2.11) Section 17.2.10b of the Wolf Creek FSAR indicates that inspections and NDE may be accomplished by "outside organizations." Describe how KG&E assures acceptable inspection/NDE procedures, qualification of the inspection/NDE personnel, and independence from undue cost and schedule pressures for these outside organizations. Provide the same information for testing activities performed by outside organizations.

RESPONSE

See Sections 17.2.4 and 17.2.7.

Q260.35 (17.2.11) Provide a commitment that procedures specify criteria for determining when a test is required or how and when tests are performed.

RESPONSE

See Section 17.2.11.2.

Q260.36 (17.2.12) The description of the control of measuring and test equipment in Section 17.2.12.2 of the Wolf Creek FSAR includes the following sentence: "Permanently installed process instrumentation is not included in this listing" (of controlled equipment). Describe the QA controls over permanently installed process instrumentation and discuss the differences between these controls and the controls described in Section 17.2.12.

RESPONSE

See Section 17.2.12.3.

Q260.37 (17.2.12) Provide a commitment that measuring and test equipment is labeled or tagged to indicate the due date of the next calibration.

RESPONSE

See Section 17.2.12.3.

Q260.38 (17.2.12) Discuss the documentation and management authorization required by KG&E when:

a) M&TE cannot be calibrated against standards that have an accuracy at least four times the required accuracy of the M&TE.

b) Calibrating standards do not have greater accuracy than standards being calibrated.

RESPONSE

See Section 17.2.12.3 and 17.2.12.4.

Q260.39 (17.2.13) Section 17.2.13.2 states that storage procedures may prescribe requirements "in lieu of" requirements contained in the manufacturer's recommendations. It appears that "supplementary to" or "in addition to" would be more appropriate than "in lieu of". Clarify.

RESPONSE

See Section 17.2.13.2.

Q260.40 (17.2.13) Describe provisions the storage of chemicals, reagents (including control of shelf life), lubricants, and other consumable materials.

RESPONSE

See Section 17.2.13.2

Q260.41 (17.2.14) Describe how KG&E controls the application and removal of inspection stamps, welding stamps, and status indicators such as tags, markings, labels, and other stamps.

RESPONSE

See Section 17.2.14.2.

Q260.42 (17.2.14) Section 17.2.14.4 states that KG&E will control the sequence of tests, inspections, and other operations in accordance with administrative procedures. Describe the procedure for such control. Such actions should be subject to the same controls as the original review and approval.

RESPONSE

See Section 17.2.14.4.

Q260.43 (17.2.14)	Clarify what is meant by the statement in 17.2.14.3 that procedures shall address methods for "initiating, maintaining, and releasing equipment control for maintenance, etc..."

RESPONSE

The second sentence of Section 17.2.14.3 has been revised for clarity.

Q260.44 (17.2.15)	Clarify Section 17.2.15.1 of the Wolf Creek FSAR that nonconformances also include inoperative and malfunctioning structures, systems, and components.

RESPONSE

See Section 17.2.15.1.

Q260.45 (17.2.15)	Describe QA controls over conditionally released nonconforming items. Identify reinspection criteria for repaired and reworked items and indicate how reinspection requirements and performance are documented. Identify individuals (by position title) or groups with authority to disposition nonconformances. Identify the individual (by position title) or group that performs the trend analysis discussed in Section 17.2.15.7.

RESPONSE

See Sections 17.2.15.2 and 17.2.15.7.

Q260.46 (17.2.15)	Provide commitment that nonconformances are corrected or resolved prior to initiation of the preoperational test program on the item.

RESPONSE

See Section 17.2.15.2.

Q260.47 (17.2.16)	Discuss the timeliness of actions taken to close out CARs and the followup action.

RESPONSE

See Section 17.2.16.3.

Q260.48 (17.2.18)	Discuss the "surveillance" portion of the KG&E audit system as mentioned in Section 17.2.18.1 of the FSAR.

RESPONSE

See Section 17.2.18.1.

Q260.49 Section 17.1.2.2 of the standard format (Regulatory Guide 1.70) requires the identification of safety-related structures, systems, and components controlled by the QA program. You are requested to supplement and clarify Table 3.2-1 of the Wolf Creek FSAR in accordance with the following:

a) The following items do not appear on FSAR Table 3.2-1. Add the appropriate items to the table and provide a commitment that the remaining items are subject to the pertinent requirements of the FSAR operational quality assurance program or justify not doing so.

a.1 Safety-related masonry walls (IE Bulletin 80-11).

RESPONSE

There are no safety-related masonry walls utilized in the Wolf Creek design.

Q260.49a.2 Biological shielding within the fuel building, auxiliary building, control building, and reactor building.

RESPONSE

Permanent biological shielding is constructed as part of safety- related buildings (refer to Section 8.1 and Table 3.2-1). Also see Section 12.1.4.

Q260.49a.3 Missile barriers within the fuel building, auxiliary building, control building, diesel-generator building, essential service water pump house.

RESPONSE

See Table 3.2-1, Sections 3.0 and 8.1. Also, permanent shields are part of the structures identified in Section 8.1 of Table 3.2-1.

Q260.49a.4 Spent fuel pool liner.

RESPONSE

See Table 3.2-1 and Section 8.2. (This item is not safety- related).

Q260.49a.5 Refueling machine.

RESPONSE

See Table 3.2-1 and Section 3.0.

Q260.49a.6 Spent fuel handling tool.

RESPONSE

See Table 3.2-1 and Section 3.0.

Q260.49a.7 Radiation shielding doors.

RESPONSE

See Table 3.2-1 and Section 8.2. (This item is not safety- related.) Also see Section 12.1.4.

Q260.49a.8 Radiation monitoring (fixed and portable).

RESPONSE

It is the Operating Agent's position that items 8-16 of Q260.49 (a) should not be included in Table 3.2-1, or be subject to the requirements of the operational Quality program. See Section 12.1.4.

Q260.49a.9 Radioactivity monitoring (fixed and portable).

RESPONSE

Refer to a.8 above.

Q260.49a.10 Radioactivity sampling (air, surfaces, liquids).

RESPONSE

Refer to a.8 above.

Q260.49a.11 Radioactive contamination measurement and analysis.

RESPONSE

Refer to a.8 above.

Q260.49a.12 Personnel monitoring internal (whole body counter) and external (TLD system).

RESPONSE

Refer to a.8 above.

Q260.49a.13 Instrument storage, calibration, and maintenance.

RESPONSE

Refer to a.8 above.

Q260.49a.14 Decontamination (facilities, personnel, equipment).

RESPONSE

Refer to a.8 above.

Q260.49a.15 Respiratory protection, including testing.

RESPONSE

Refer to a.8 above.

Q260.49a.16 Contamination Control.

RESPONSE

Refer to a.8 above.

Q260.49a.17 Radiation shielding (permanently installed).

RESPONSE

Refer to a.2 above.

Q260.49a.18 Accident-related meteorological data collection equipment.

RESPONSE

See Sections 2.3.3.2.3 and 2.3.3.5.1.

Q260.49a.19 Expendable and consumable items necessary for the functional performance of safety-related structures, systems, and components (weld rod, fuel oil, boric acid, snubber oil, etc.)

RESPONSE

See Section 17.2.13.2.

Q260.49a.20 Roof drains and parapets of buildings which house safety-related equipment.

RESPONSE

See Section 2.4.2.3.

Q260.49a.21 Site drainage system including grading, culverts, and channels.

RESPONSE

See note 14 of Table 3.2-1 and Section 2.4.2.2.

Q260.49a.22 Steam generators (primary and secondary).

RESPONSE

See Table 3.2-1 and Section 1.1.

Q260.49a.23 Steam generator piping located inside containment.

RESPONSE

See Sections 5.1 and 5.2 of Table 3.2-1.

Q260.49a.24 Valve operators for all safety-related valves.

RESPONSE

Valve operators are considered part of each safety-related valve. See Table 3.2-1 and fourth paragraph of Section 3.2.

Q260.49a.25 Motors for all safety-related pumps.

RESPONSE

Motors are considered part of each safety-related pump. See Table 3.2 and the fourth paragraph of Section 3.2.

Q260.49 b) The following items from FSAR Table 3.2-1 need expansion and/or clarification as noted. Revise the list as indicated or justify not doing so.

1) Identify the safety-related instrumentation and control systems to the same scope and level of detail as provided in Chapter 7 of the FSAR. (This can be done by footnote). Verify that this includes I & C for:

Q260.49b.1(a) Containment spray system.

RESPONSE

See Section 1.5 of Table 3.2-1 and the fourth paragraph of Section 3.2.

Q260.49b.1(b) Containment cooling system.

RESPONSE

See Section 1.6 of Table 3.2-1 and the fourth paragraph of Section 3.2.

Q260.49b.1(c) Containment hydrogen control system.

RESPONSE

See Section 1.8 of Table 3.2-1 and the fourth paragraph of Section 3.2.

Q260.49b.1(d) Containment pressure indication.

RESPONSE

See Section 9.0 of Table 3.2-1 and the fourth paragraph of USAR Section 3.2.

Q260.49b.1(e) Containment water level indication.

RESPONSE

See Section 9.0 of Table 3.2-1 and the fourth paragraph of Section 3.2.

Q260.49b.1(f) Containment hydrogen indication.

RESPONSE

See Section 1.8 of Table 3.2-1 and the fourth paragraph of USAR Section 3.2.

Q260.49b.2 For the systems shown below, expand the list in Table 3.2-1 to include the indicated components under the pertinent 10 CFR 50 Appendix B quality assurance requirements or verify that they are included as part of the components already listed.

Q260.49b.2.1.5 Containment spray system containment sump.

RESPONSE

See Section 8.1 of Table 3.2-1.

Q260.49b.2.1.6 Containment cooling system ductwork.

RESPONSE

See Section 1.6 of Table 3.2-1 and the fourth paragraph of Section 3.2.

Q260.49b.2.1.8 Containment hydrogen control system piping and valves.

RESPONSE

See Section 1.8 of Table 3.2-1 and the fourth paragraph of Section 3.2.

Q260.49c Enclosure 2 of NUREG-0737, "Clarification of TMI Action Plan Requirements" (November 1980) identified numerous items that are safety-related and appropriate for OL application and therefore should be on Table 3.2-1. These items are listed below. Add appropriate items to Table 3.2-1 and provide a commitment that the remaining items are subject to the pertinent requirements of FSAR operational QA program or justify not doing so.

Q260.49c.1 Plant safety-parameter display console.

RESPONSE

See Section 9 of Table 3.2-1.

Q260.49c.2 Reactor coolant system vents.

RESPONSE

See Section 1.1 of Table 3.2-1 and the fourth paragraph of Section 3.2.

Q260.49c.3 Plant shielding.

RESPONSE

Refer to a.2 above.

Q260.49c.4 Post accident sampling capabilities.

RESPONSE

The equipment used for inplant post-accident sampling is not safety-related and therefore is not included in Table 3.2-1. However, the portions of the system which are involved in maintaining containment integrity are procured and installed as safety-related equipment. See Section 1.7 of Table 3.2-1 and Section 12.1.4 and 8.1.

Q260.49c.5 Valve position indication.

RESPONSE

Position indication of each pressurizer safety valve and PORV is considered part of the valve. Therefore, this is included in Section 1.1 of Table 3.2-1 and see fourth paragraph of Section 3.2.

Q260.49c.6 Auxiliary feedwater system.

RESPONSE

See Section 5.4 of Table 3.2-1.

Q260.49c.7 Auxiliary feedwater system initiation and flow.

RESPONSE

See Section 5.4 of Table 3.2-1 and the fourth paragraph of Section 3.2.

Q260.49c.8 Emergency power for pressurizer heaters.

RESPONSE

See Section 1.1 of Table 3.2-1 and the fourth paragraph of Section 3.2.

Q260.49c.9 Dedicated hydrogen penetrations.

RESPONSE

Not applicable to Wolf Creek.

Q260.49c.10 Containment isolation dependability.

RESPONSE

See Section 1.7 of Table 3.2-1.

Q260.49c.11 Accident monitoring instrumentation.

RESPONSE

See Section 9.0 of Table 3.2-1.

Q260.49c.12 Instrumentation for detection of inadequate core-cooling.

RESPONSE

See the fourth paragraph of Section 3.2.

Q260.49c.13 Power supplies for pressurizer relief valves, block valves, and level indicators.

RESPONSE

See Section 1.1 of Table 3.2-1 and the fourth paragraph of Section 3.2.

Q260.49c.14 Automatic PORV isolation.

RESPONSE

See Section 1.1 of Table 3.2-1 and the fourth paragraph of Section 3.2.

Q260.49c.15 Automatic trip of reactor coolant pumps.

RESPONSE

Not applicable to Wolf Creek.

Q260.49c.16 PID controller.

RESPONSE

Not functional in Wolf Creek design.

Q260.49c.17 Anticipatory reactor trip on turbine trip.

RESPONSE

See Section 9.0 of Table 3.2-1 for the Reactor Protection System. The remainder of the system is non-IE but meets special criteria as defined in Section 7.2.1.1.2.f.

Q260.49c.18 Power on pump seals.

RESPONSE

Included as part of Section 2.3 of Table 3.2-1.

Q260.49c.19 Emergency plans (and related equip).

RESPONSE

Emergency plans are not systems, structures or components, are not considered safety-related and are therefore not included in Table 3.2-1. However, Emergency Plan effectiveness is verified through periodic drills and exercises as described in the Emergency Plans.

Q260.49c.20 Equipment and other items associated with the emergency support facilities.

RESPONSE

These are not considered safety-related and are therefore not included in Table 3.2-1. However, periodic checks of radiation measurement and communication equipment is required by written procedure. Appropriate engineering and reference documents (i.e., FSAR, prints, procedure manuals) will be placed in Wolf Creek emergency response facilities. The controls and update of reference documents will be handled in accordance with procedures. Emergency procedures will be subject to audit by individuals who are not directly responsible for procedure implementation. See Emergency Plans and Procedures.

Q260.49c.21 In-plant I_2 radiation monitoring.

RESPONSE

Inplant iodine monitoring is not considered safety-related and is therefore not included in Table 3.2-1. Provisions for monitoring of inplant iodine levels are incorporated within the scope of the Wolf Creek Health Physics Manual and procedures as described in Section 12.1.4.

Q260.49c.22 Control room habitability.

RESPONSE

See Section 7.1 of Table 3.2-1.

Q270.1 (SRP 3.11) Correlate the systems listed in Table 3.2-1 of the FSAR with the systems listed in Appendix B of the environmental qualification (EQ) program submittal of March 10, 1983. Provide justification for any system listed in Table 3.2-1 which is excluded from Appendix B (e.g., all components of the system are located in a mild environment, etc.). Identify the Class 1E function for all systems in Appendix B.

RESPONSE

Comparing Table 3.2-1 (USAR) to Appendix B (submittal) is inappropriate since the two listings were developed to different criteria and for different purposes.

It should also be noted that the listing of Appendix B includes all systems receiving Class 1E electrical power. No systems have been deleted due to their location (e.g., in a mild environment) as indicated by your questions.

Three systems identified in Appendix B are listed only because some portion of the system provides electrical isolation. The system identifiers are PN, RJ, and RK. These systems do not have any other Class 1E function. Note 1 of Appendix B clearly identifies this fact. Accordingly, no "X"s are provided for these systems.

Q270.2 (SRP 3.11) Identify, by categories listed in NUREG-0737, the components included in the qualification program in response to TMI Action Plan Requirements.

RESPONSE

See Section 3.11(B).1 and 18.0. It should be noted that much of the equipment required to satisfy NUREG-0737 concerns already existed in the plant design.

Q270.3 (SRP 3.11) The description of the criteria used for establishing environmental qualification does not reference Section II.B.2 of NUREG-0737 as the basis for establishing radiation dose from recirculating fluids. Discuss your compliance with the recommendations of this section of the Action Plan.

RESPONSE

Section 18.2.2 discusses in detail the WCGS position concerning Section II.B.2 of NUREG-0737.

Q270.4 (SRP 3.11) Provide a statement that 1E equipment located in areas which experience a significant increase in radiation during a LOCA has been reviewed for possible damage to solid state devices.

RESPONSE

See Section 3.11(B).1 and 3.11(B).2.1.f.

Q270.5 (SRP 3.11) Section 8.11 of the March 10, 1983 EQB program submittal indicates a minimized coverage of synergistic effects. Discuss what activity will be undertaken to identify known synergistic effects and how these will be factored into the EQ program.

RESPONSE

See Section 3.11(B).5.8.

Q270.6 (10 CFR 50.49) To demonstrate compliance with 10 CFR 50.49, the following information must be submitted before an operating license is granted:

a) In accordance with the scope defined in 10 CFR 50.49, provide:

- A list of all nonsafety-related electrical equipment located in a harsh environment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions by the safety-related equipment. A description of the method used to identify this equipment must be included. The nonsafety-related equipment identified must be included in the environmental qualification program.

- A statement that all safety-related electric equipment in a harsh environment, as defined in the scope of 10 CFR 50.49, is included in this list of equipment identified in the March 10, 1983 submittal (including equipment required for MELB, spent fuel rod drop accident, etc.).

- A list of all Category 1 and 2 post-accident monitoring equipment currently installed, or to be installed before plant operation, in response to Regulatory Guide 1.97, Revision 2. The equipment identified must be included in the environmental qualification program.

b) Provide information demonstrating qualification of all equipment in a harsh environment within the scope of 10 CFR 50.49, or provide justification for interim operation pending completion of qualification as required by 10 CFR 50.49. This material should be submitted to allow sufficient time for staff review and approval before issuance of an operating license.

RESPONSE

a) The WCGS design is based on utilizing only Class 1E powered electrical equipment to mitigate the consequences of the units identified in Section 2.3 of the submittal. See USAR Section 3.11(B).1 and Question 720.3 for additional information.

- Section 2.0 identifies that Appendix A includes all safety-related electrical equipment, regardless of the accident that required the equipment to be categorized as Class 1E. No Class 1E equipment is excluded from the list due to location or any other reason.

- Appendix 7A of the USAR identifies the WCGS position on Regulatory Guide 1.97. A categorized list of equipment is included in Appendix 7A. Section 8.2 of the submittal references the FSAR response and indicates that all Regulatory Guide 1.97 Category 1 instruments are included in the listing of Appendix A of the submittal. Additionally, all Category II electrical components powered by a Class 1E power source (as shown in Appendix A of the USAR) are also included.

b) Please refer to the submittal transmittal letter (SLNRC 83-0015, dated March 10, 1983) which states "...corrective actions will be taken to establish equipment qualification prior to fuel loading or justification will be provided for interim operation until corrective actions are completed." This information was submitted. See Section 3.11(B).3.

Q270.7 (SRP 3.11) Indicate your compliance with a one hour time margin for equipment with operability times less than 10 hours, or provide justification for reduced margins.

RESPONSE

See Section 3.11(B).5.2.

Q270.8 (SRP 3.11) Before the Safety-Related Mechanical (SRM) equipment audit items can be selected, you must indicate the qualification status of the SRM equipment. If qualification is not complete, briefly describe the tasks to be performed. Provide a list of SRM equipment which is considered qualified from which audit items can be selected. Your review of equipment should be essentially complete before items are selected.

RESPONSE

The Operating Agent considers the safety-related mechanical equipment to be qualified for its intended use. See Section 3.11(B).6.

Q270.9 (SRP 3.11) Table I Master Qualification Summary, Section II of the March 10, 1983 submittal, indicates that the qualification status has not been determined for 16 out of 74 qualification packages (3 packages - review is in progress, 13 packages - review has not started). The Equipment Qualification Branch considers the review incomplete until at least 85% of all equipment items have been categorized.

RESPONSE

This information was provided prior to receipt of the Operating License. See Section 3.11(B).6.

Q270.10 (SRP 3.11) A number of Qualification Summary Sheets state that qualification documentation is auditable but is incomplete, yet the equipment is considered qualified. Please explain this apparent contradiction.

RESPONSE

There is no contradiction involved. At the time the question was originally asked, when the submittal indicated that specific equipment documentation was auditable but incomplete and the equipment was considered qualified, then one of two conditions existed. Either a) the majority of the information was submitted and reviewed and the remaining documentation was considered proprietary, but the content was known and was at the vendor's facility available for audit, or b) the majority of the information was submitted and reviewed and the remaining documentation would only enhance the existing documentation. In either case, the vendor was contacted to determine the content of the missing information before the equipment was considered qualified.

It should also be noted that a review of the qualification summaries indicated only one case in which the documentation was incomplete, but the equipment was considered qualified. The incomplete documentation was an enhancement, but the vendor was requested to supply the documentation. The appropriate documentation has been received, and Revision 1 of the qualification summary has been changed to reflect the documentation being complete.

Q270.11 (SRP 3.11) The justification given to reconcile test failures, tests not performed and inconsistencies between test parameter levels and plant requirements seem strained in a number of instances (e.g., E028, E029, E093, E062, M 223A, etc.). Please review the basis for determining qualification and, if appropriate, strengthen the justifications or re-evaluate the qualification status.

RESPONSE

Specifications E028 and E093 are not considered qualified. Accordingly, the qualification summaries for these specifications do not indicate that they are qualified. For the remaining identified specifications (and all others), it should be noted that only the summary is submitted. Additional data leading to the conclusion reached is available in the associated utility files. Due to the extensive conservatism built into the qualification review program, we feel that the justifications are not strained. No changes of qualification status are necessary.

Q270.12 (SRP 3.11) Provide an example of the equipment surface temperature calculations referenced in Section 6.2.2 of the EQ submittal which allows credit for specific equipment surface temperature response for MSLB environments.

RESPONSE

See Section 3.11(B).1.

Q270.13 (SRP 3.11) Provide an example of the equipment specific analysis referenced in Section 6.3.1 of the EQ submittal to demonstrate how radiation dose reductions were obtained.

RESPONSE

See Section 3.11(B).1.2.3.

Q270.14 (SRP 3.11) Provide information on the specific maintenance/surveillance programs to be applied to 1) Cables located inside containment, 2) Limitorque valve operators, 3) Amphenol electrical penetrations, 4) Motor control center relays and circuit breakers, and 5) Barton pressure transmitters.

RESPONSE

See 3.11(B).5.6.

Q270.15 (SRP 3.11) The temperature profiles shown for postulated HELBs outside containment do not meet the screening criterion of saturation temperature at the calculated pressure. Please provide an example of the analysis used to determine the environmental conditions resulting from a line break outside containment.

RESPONSE

See Section 3.11(B)-1.

Q270.16 (SRP 3.11) The applicant is requested to identify the systems listed in FSAR Table 3.2-1 which include Instrumentation and Control (I&C) equipment. This may be done by modifying Table 3.2-1 to include Instrumentation and Control as subsets or portions of the systems identified.

RESPONSE

See Note 14 of Table 3.2-1.

Q270.17 (SRP 3.11) Describe the criteria used to determine the I&C systems and components important to safety to be covered by the equipment qualification program.

RESPONSE

See response to Question 270.6 (a). Additionally, USAR Section 7.1.1, Identification of Safety-Related Systems, identifies the criteria for the selection of I&C equipment as being safety related.

Q270.18 (SRP 3.11) Describe the method used to identify each specific I&C component covered.

RESPONSE

See Note 14 of Table 3.2-1.

Q271.1 In accordance with the requirements of GDC 2 and 4 all safety-related equipment is required to be designed to withstand the effects of earthquakes and dynamic loads from normal operation, maintenance, testing and postulated accident conditions. GDC 2 further requires that such equipment be designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of earthquake loads.

The criteria to be used by the staff to determine the acceptability of your equipment qualification program for seismic and dynamic loads are IEEE Std. 344-1975 as supplemented by Regulatory Guides 1.100 and 1.92, and Standard Review Plan Sections 3.9.2, 3.9.3 and 3.10. State the extent to which the equipment in your plant meets these requirements and the above requirements to combine seismic and dynamic loads. For equipment that does not meet these requirements justification will be needed for the use of other criteria.

RESPONSE

All safety-related equipment is designed to withstand the effects of earthquake and dynamic loads. The extent to which the powerblock equipment meets the requirements of the questioned documents is provided in the USAR Sections referenced below.

IEEE Std. 344-1975: 3.10(B), 3.10(N)

Regulatory Guide 1.100: 3.10(B), Appendix 3A

Regulatory Guide 1.92: Appendix 3A, 3.7(B), 3.7(N)

Standard Review Plan (SRP) 3.9.2: 3.9.2(B), 3.9.2(N)

SRP 3.9.3: 3.9.3(B), 3.9.3(N)

SRP 3.10: 3.10(B), 3.10(N)

In addition, the extent to which powerblock equipment meets the recommendations of Regulatory Guide 1.29, "Seismic Design Classification" is provided in Section 3.2 and Appendix 3A.

Q271.2 (271.3)

To confirm the extent to which the equipment important to safety meets the requirements of General Design Criterion 2 and 4, the Seismic Qualification Review Team (SQRT) will conduct a plant site review. For selected equipment, SQRT will review the combined required response spectra (RRS) or the combined dynamic response, examine the equipment configuration and mounting, and then determine whether the test or analysis which has been conducted demonstrates compliance with the RRS if the equipment was qualified by test, or the acceptable analytical criteria if qualified by analysis.

In order to select equipment types for a detailed review it is necessary to obtain a list of <u>all equipment important to safety</u>. Equipment should be divided first by system then by component type. Attachment #1 shows a tabular format which should be followed to present the status summary of seismic and dynamic qualification of all equipment important to safety. Attachment #2 shows suggested categories of component type to be listed in Attachment #1. Provide a complete set of floor response spectra identifying their applicability to the equipment listed in Attachment #1.

After the information on Attachment #1 is received, a selection will be made of the equipment to be reviewed by the site audit. Specific information on equipment selected for audit should be presented as shown on Attachment #3 which should be provided to the NRC staff two weeks prior to the plant site visit. The applicant should make available at the plant site for SQRT review all the pertinent documents and reports of the qualification of the selected equipment. After the visit, the applicant should be prepared to submit certain selected documents and reports for further staff review.

The purpose of the site audit is to confirm the acceptability of the seismic and dynamic qualification of all equipment important to safety based on the review of a few selected pieces. If a number of deficiencies are observed or significant generic concerns arise, the deficiencies should be removed for <u>all equipment important to safety</u> subject to confirmation by a follow-up audit of randomly selected items before the fuel loading date.

RESPONSE

A list of all safety-related equipment was provided to the NRC by SLNRC 82-06 dated February 4, 1982. The list was updated by SLNRC 83-026 dated May 9, 1983.

Q280.1 Provide a table that lists all equipment including instrumentation and vital support system equipment required to achieve and maintain hot and/or cold shutdown. For each equipment listed:

a) Differentiate between equipment required to achieve and maintain hot shutdown and equipment required to achieve and maintain cold shutdown,

b) Define each equipment's location by fire area,

c) Define each equipment's redundant counterpart,

d) Identify each equipment's essential cabling (instrumentation, control, and power). For each cable identified: (1) Describe the cable routing (by fire area) from source to termination, and (2) Identify each fire area location where the cables are separated by less than a wall having a three-hour fire rating from cables for any redundant shutdown system, and

e) List any problem areas identified by item 1.d.(2) above that will be corrected in accordance with Section III.G.3 of Appendix R (i.e., alternate or dedicated shutdown capability).

RESPONSE

The final fire hazards analysis, USAR Appendix 9.5B, identifies all redundant post-fire safe shutdown components and circuits on a fire area by fire area basis, and demonstrates that either the required separation exists or that alternate means are available to perform the safe shutdown function.

Section 7.4 provides a safe shutdown discussion and lists of systems and components required for hot standby and cold shutdown.

Table 3.11(B).3, identifies all the equipment required for safe shutdown, differentiates between hot and cold shutdown requirements, and identifies the location of each component.

Q280.2 Provide a table that lists Class 1E and Non-Class 1E cables that are associated with the essential safe shutdown systems identified in item 1 above. For each cable listed: (*Note).

*NOTE

Option 3a is considered to be one method of meeting the requirements of Section II.G.3 Appendix R. If option 3a is selected the information requested in items 2a and 2c above should be provided in general terms and the information requested by 2b need not be provided.

a) Define the cables' association to the safe shutdown system (common power source, common raceway, separation less than IEEE Standard-384 guidelines, cables for equipment whose spurious operation will adversely affect shutdown systems, etc.),

b) Describe each associated cable routing (by fire area) from source to termination, and

c) Identify each location where the associated cables are separated by less than a wall having a three-hour fire rating from cables required for or associated with any redundant shutdown system.

RESPONSE

As stated in Section 8.1.4.3, in complying with Regulatory Guide 1.75, associated circuits are separated and identified as if they are safety-related.

The final fire hazards analysis, Appendix 9.5B, demonstrates that adequate separation is provided for post-fire safe shutdown systems.

Q280.3 Provide one of the following for each of the circuits identified in item 2c above:

a) The results of an analysis that demonstrates that failure caused by open, ground, or hot short of cables will not affect it's associated shutdown system, (*Note)

b) Identify each circuit requiring a solution in accordance with Section III.G.3 of Appendix R, or

c) Identify each circuit meeting or that will be modified to meet the requirements of Section III.G.2 of Appendix R (i.e., three-hour wall, 20 feet of clear space with automatic fire suppression, or one-hour barrier with automatic fire suppression).

*NOTE

Option 3a is considered to be one method of meeting the requirements of Section II.G.3 Appendix R. If option 3a is selected the information requested in items 2a and 2c above should be provided in general terms and the information requested by 2b need not be provided.

RESPONSE

As stated in Section 8.1.4.3, there are no associated circuits whose failure would affect safe shutdown systems.

Q280.4 To assure compliance with GDC 19, we require the following information be provided for the control room. If credit is to be taken for an alternate or dedicated shutdown method for other fire areas (as identified by item 1e or 3b above) in accordance with Section III.G.3 of new Appendix R to 10 CFR Part 50, the following information will also be required for each of these plant areas.

a) A table that lists all equipment including instrumentation and vital support system equipment that are required by the primary method of achieving and maintaining hot and/or cold shutdown.

b) A table that lists all equipment including instrumentation and vital support system equipment that are required by the alternate, dedicated, or remote method of achieving and maintaining hot and/or cold shutdown.

c) Identify each alternate shutdown equipment listed in item 4b above with essential cables (instrumentation, control, and power) that are located in the fire area containing the primary shutdown equipment. For each equipment listed provide one of the following:

1) Detailed electrical schematic drawings that show the essential cables that are duplicated elsewhere and are electrically isolated from the subject fire areas, or

2) The results of an analysis that demonstrates that failure (open, ground, or hot short) of each cable identified will not affect the capability to achieve and maintain hot or cold shutdown.

d) Provide a table that lists Class 1E and Non-Class 1E cables that are associated with the alternate, dedicated, or remote method of shutdown. For each item listed, identify each associated cable located in the fire area containing the primary shutdown equipment. For each cable so identified provide the results of an analysis that demonstrates that failure (open, ground, or hot short) of the associated cable will not adversely affect the alternate, dedicated, or remote method of shutdown.

RESPONSE

A discussion of safe shutdown and a list of systems necessary for safe shutdown are in Section 7.4. Section 7.4 also describes the capability of the auxiliary shutdown panel for safe shutdown from outside the control room.

The final fire hazards analysis, USAR Appendix 9.5B, considers primary, alternate, and associated circuits and demonstrates that any single fire will not prevent the safe shutdown of the plant.

Q280.5 The residual heat removal system is generally a low pressure system that interfaces with the high pressure primary coolant system. To preclude a LOCA through this interface, we require compliance with the recommendations of Branch Technical Position RSB 5-1. Thus, this interface most likely consists of two redundant and independent motor operated valves with diverse interlocks in accordance with Branch Technical Position ICSB 3. These two motor operated valves and their associated cable may be subject to a single fire hazard. It is our concern that this single fire could cause the two valves to open resulting in a fire-initiated LOCA through the subject high-low pressure system interface. To assure that this interface and other high-low pressure interfaces are adequately protected from the effects of a single fire, we require the following information:

a) Identify each high-low pressure interface that uses redundant electrically controlled devices (such as two series motor operated valves) to isolate or preclude rupture of any primary coolant boundary.

b) Identify each device's essential cabling (power and control) and describe the cable routing (by fire area) from source to termination.

c) Identify each location where the identified cables are separated by less than a wall having a three-hour fire rating from cables for the redundant device.

d) For the areas identified in item 5c above (if any), provide the bases and justification as to the acceptability of the existing design or any proposed modifications.

RESPONSE

The reactor coolant system high-low pressure interfaces that rely on redundant electrically controlled devices for isolation include the RHR letdown isolation valves.

The fire hazards analysis, Appendix 9.5B, demonstrates that no single credible fire could cause the spurious opening of these valves in a manner that would breach the primary coolant boundary.

Q280.6 Notification of Appendix R to 10 CFR Part 50 as a Licensing Requirement.

Appendix R to 10 CFR Part 50 will also be used as guidance for our review of your fire protection program. Your compliance with the requirement set forth in Appendix R as modified by accepted exceptions will be made a license condition. Identify any exceptions your program takes to the requirements of Appendix R as well as BTP ASB 9.5-1, and describe your alternative for providing an equivalent level of fire protection.

RESPONSE

Table 9.5E-1 provides the requested comparisons and identifies the exceptions of the Wolf Creek Generating Station to 10 CFR 50 Appendix R. Table 9.5B-1 provides the WCGS Fire Protection comparisons to APCSB 9.5-1 Appendix A.

Q281.1 Indicate the total amount of protective coatings and organic materials (including conduit covered and uncovered cable insulation) used inside the containment that do not meet the requirements of ANSI N101.2 (1972) and Regulatory Guide 1.54. Evaluate the generation rates vs. time of combustible gases that can be formed from these unqualified organic materials under DBA conditions. Also evaluate the amount (volume) of solid debris that can be formed from these unqualified organic materials under DBA conditions that can reach the containment sump. Provide the technical basis and assumptions used for this evaluation.

RESPONSE

See Section 6.2.5.2.3 c and d.

Q281.2 Regarding the fuel pool cooling and cleanup system, indicate the sampling frequency and criteria for filter and/or ion exchanger resin replacement. Items to be addressed should include (1) decontamination factor, (2) radiation level, and (3) differential pressure.

RESPONSE

See Section 9.1.3.2.3.2.

Q281.3 Describe the provisions to meet the requirements of post-accident sampling of the primary coolant and containment atmosphere. The description should address all the requirements outlined in Section II.B.3 of Enclosure 3 in NUREG-0737 (Clarification TMI Action Plan Requirements) and should include the appropriate P & ID's. In addition, if gas chromatography is used for reactor coolant analysis, special provisions (e.g., pressure relief and purging) should be provided to prevent high-pressure carrier gas from entering the reactor coolant. With respect to clarification (4) in Section II.B.3 of NUREG-0737, if the chloride concentration in the reactor coolant samples exceeds the limit in the Technical Specification, verification that oxygen is less than 0.1 PPM will be mandatory. Provide also either (a) a summary description of procedures for sample collection, sample transfer or transport, sample analysis and analytical accuracy or (b) copies of procedures for sample collection, sample transfer or transport, sample analysis and analytical accuracy.

RESPONSE

See Section 18.2.3.

Q282.1 (4.5.1)

To evaluate the compatibility of the control rod drive structural materials with the reactor coolant water, provide the list of materials and specifications which are used for each component of the control rod drive mechanism. The information in the FSAR does not adequately identify the materials.

RESPONSE

The requested information is located in Table 5.2-2.

Q282.2 (10.3.5)

Provide the following on your secondary water chemistry control and monitoring programs:

a) Sampling schedule for the critical parameters and of control points for these parameters for the cold startup mode of operation;

b) Procedures used to measure the values of the critical parameters;

c) Procedure for recording and management of data;

d) Procedures defining corrective actions* for off-control point chemistry conditions; and

e) A procedure identifying (1) the authority responsible for the interpretation of the data and (2) the sequence and timing of administrative events required to initiate corrective action.

Verify that the steam generator secondary water chemistry control program incorporates technical recommendations of the NSSS. Any significant deviations from NSSS recommendations should be noted and justified technically.

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*Branch Technical Position MTEB 5-3 describes the acceptable means for monitoring secondary side water chemistry in PWR steam generators including corrective actions for off-control point chemistry conditions. However, the Staff is amenable to alternatives, particularly to Branch Technical Position B.3.b(9) of MTEB 5-3 (96 - hour time limit to repair or plug confirmed condenser tube leaks).

RESPONSE

These items have been covered by plant procedures.

The steam generator secondary water chemistry control program incorporates the technical recommendations of Westinghouse.

As stated in Section 10.3.5.1, the requirements of MTEB 5-3 are met.

Q310.1 Figure 2.1-7 shows an abandoned A.T.&S.F. railroad line passing through the Wolf Creek site. Please explain the status of the line. Discuss any easements which may exist relative to this railroad line.

RESPONSE

Refer to Section 2.2.1.4.

Q310.2 The population of Burlington in the year 2010, as shown in Figure 2.1-13, is difficult to read. Please provide the population estimates for Burlington for the years 1990, 2000, 2010 and 2020.

RESPONSE

See Figures 2.1-10 through 2.1-14.

Q310.3 Discuss any recreational areas within the Wolf Creek site boundary.

RESPONSE

See Section 2.1.2.5.

320.0 OFFICE OF STATE PROGRAMS

The Nuclear Regulatory Commission amended 10 CFR Part 2, Rules of Practice for Domestic Licensing Proceedings and 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities, effective March 31, 1982, to eliminate entirely requirements for financial qualifications review and findings for electric utilities that are applying for construction permits or operating licenses for production or utilization facilities (47 FR 13750, March 31, 1982).

Accordingly, the 320 Series questions and responses were no longer required and were deleted.

Q331.0 RADIOLOGICAL ASSESSMENT BRANCH

Q331.1 (12.1.2.5b) Section 12.1.2.5b addresses a neutron shield design at the RPV in containment. Please specify the neutron and gamma dose equivalent rates that will exist at specific locations within the various levels of containment prior to shield installation and after the shield is installed. A figure or table showing respective dose rates would be a suitable format. Describe your plan for neutron personnel dosimetry whenever an entry is made while the reactor is at power, the frequencies at which entries are made, and the number of people making these entries.

RESPONSE

See Section 3.8.3.1.4.

Q331.2 (12.2.1.3) Radiation levels in excess of 100 R/hr can occur in the vicinity of spent fuel transfer tubes; therefore, all accessible portions of the transfer tubes must be shielded during fuel transfer. Please address the manner in which shielding, access control and radiation monitoring will be incorporated into the radiation protection program to prevent either occupants or transient workers from receiving very high exposures during transfer of spent fuel from the reactor to the spent fuel pool through the fuel transfer tubes. Use of removable shielding for this purpose is acceptable. Provide appropriate figures (e.g. plan and elevation) that show the shielding arrays for all direct gamma radiation and streaming pathways from the spent fuel during the transfer. On the same figure show the location of any administrative controls by barriers, signs, audible and visual alarms, locked doors, etc. All accessible portions of the transfer tubes that cannot be adequately shielded shall be clearly marked with a sign stating that potentially lethal fields are possible during fuel transfer.

RESPONSE

See Section 9.1.2.2, and Figures 3.8-48 and 12.3-2.

Q331.3 (12.2.1.2.3) Describe the procedure for extracting a sample from the Nuclear Sampling System of RCS, RHR and CVCS with as low as is reasonably achievable exposures to personnel withdrawing the sample. In your response include use of shielding, area monitoring, portable survey meters, hand contact with sample containers, dose rate levels in sampling area, dose rate level of sample container, etc. Consider samples taken during normal operations, anticipated operational occurrences and accidents. The response to this question should satisfy the requirements of NUREG-0578 item 2.1.8.a, Post Accident Sampling, with regard to Radiation Protection.

RESPONSE

See Section 12.2.1.2.3.

Q331.4 (Table 12.2-7) Table 12.2-7 indicates the radionuclide concentration in the spent fuel pool (SFP) water. Relevant reactor operating experience shows that the ^{60}Co activity, from crud transferred to the SFP from the interchange of the primary coolant water during refueling, is several orders of magnitude greater than that shown in the table even after purification by the SFP clean-up system. Please justify the values given in the table for ^{60}Co, ^{58}Co, ^{134}Cs, and ^{137}Cs and show that these values will be retained after several years of reactor operation. Provide an estimate of the dose rate above the SFP during a refueling operation and for the period thereafter. Include in the estimate the effect on the dose rate of any radioactive equipment that might be stored therein.

RESPONSE

See Table 12.2-7, Section 9.1.2.2.

Q331.5 (12.3.4.2.2.2.2.) Please clarify how iodine radioactivity levels can be "inferred from the particulate and noble gas radioactivity levels" when monitoring the exhaust from the radwaste and auxiliary buildings as addressed in Sections 12.3.4.2.2.2.2 and 12.3.4.2.2.2.4.

RESPONSE

See revised Sections 12.3.4.2.2.2.2 and 12.3.4.2.2.2.4.

Q360.1 EFFLUENT TREATMENT

Q360.1 (11.4) Table 11.4-3 (sheet 2) of the SNUPPS FSAR indicates that the estimated annual volume of dry and compacted waste is based upon Table 2-49 of WASH-1258. The estimated volume was 3,380 ft^3. Page 11.4-8 of the SNUPPS FSAR states that the filled drums are sealed and moved to the dry waste storage area in the radwaste building, where they are stored until they are shipped offsite. Figure 1.2-3 of the SNUPPS FSAR shows that the storage area has a storage capacity of 722 drums, if stacked three high, and 1055 drums, if stacked five high. Data made available since the publication of WASH-1258 have made that document inappropriate for waste projections. The dry waste volumes estimated by WASH-1258 are much lower than those being generated at operating reactors. NRC staff calculations, which are based on data from semi-annual effluent reports, show that the volume of dry wastes generated are independent of reactor size and amount to approximately 10,000 ft^3 (compacted) annually, which is a factor of three greater than the estimates presented in the SNUPPS FSAR. Also, the growing uncertainty of the availability of burial space has made the availability of adequate storage space at the reactor facility an important issue.

Based upon the material presented above, provide information verifying that the storage space at Callaway will be sufficient to handle the storage of drummed waste in accordance with the requirements of Branch Technical Position, ETSB 11-3 (Rev. 1), item III (Waste Storage).

RESPONSE

See revised Section 11.4.

Q360.2 (11.4) Page 11.4-12 the SNUPPS FSAR discusses shielded storage areas for "high-level" solidified radwaste and "low-level" solid radwaste. The term "high-level" is inappropriate and should be revised. "High-level" generally refers to reprocessing wastes resulting from the first cycle of solvent extraction. More recently, use of the term has been extended to cover spent reactor fuel. See 10 CFR Part 50, Appendix F, item 2.

RESPONSE

The terms "high-level" and "low-level" were eliminated and replaced by primary and secondary, respectively, in Section 11.4 to differentiate drummed solid wastes that require radiation shielding from those that do not.

Q420.1 Loss of Non-Class IE Instrumentation and Control Power System Bus During Power Operation (IE Bulletin 79-27)

If reactor controls and vital instruments derive power from common electrical distribution systems, the failure of such electrical distribution systems may result in an event requiring operator action concurrent with failure of important instrumentation upon which these operator actions should be based. This concern was addressed in IE Bulletin 79-27. On November 30, 1979, IE Bulletin 79-27 was sent to operating license (OL) holders, the near term OL applicants (North Anna 2, Diablo Canyon, McGuire, Salem 2, Sequoyah, and Zimmer), and other holders of construction permits (CP), including Callaway 1 and Wolf Creek. Of these recipients, the CP holders were not given explicit direction for making a submittal as part of the licensing review. However, they were informed that the issue would be addressed later.

You are requested to address these issues by taking IE Bulletin 79-27 Actions 1 thru 3 under "Actions to be Taken by Licensees". Within the response time called for in the attached transmittal letter, complete the review and evaluation required by Actions 1 thru 3 and provide a written response describing your reviews and actions. This report should be in the form of an amendment to your FSAR and submitted to the NRC Office of Nuclear Reactor Regulations as a licensing submittal.

RESPONSE

See Section 8.1.4.3.

Q420.2 Engineered Safety Features (ESF) Reset Controls (IE Bulletin 80-06)

If safety equipment does not remain in its emergency mode upon reset of an engineered safeguards actuation signal, system modification, design change or other corrective action should be planned to assure that protective action of the affected equipment is not compromised once the associated actuation signal is reset. This issue was addressed in IE Bulletin 80-06 (enclosed). For facilities with operating licenses as of March 13, 1980, IE Bulletin 80-06 required that reviews be conducted by the licensees to determine which, if any, safety functions might be unavailable after reset, and what changes could be implemented to correct the problem.

For facilities with a construction permit including OL applicants Bulletin 80-06 was issued for information only.

The NRC staff has determined that all CP holders, as a part of the OL review process, are to be requested to address this issue. Accordingly, you are requested to take the actions called for in Bulletin 80-06 Actions 1 thru 4 under "Actions to be Taken by Licensees". Within the response time called for in the attached transmittal letter, complete the review verifications and description.

RESPONSE

See Section 7.3.

Q420.3 Qualification of Control Systems (IE Information Notice 79-22)

Operating reactor licensees were informed by IE Information Notice 79-22, issued September 19, 1979, that certain non-safety grade or control equipment, if subjected to the adverse environment of a high energy line break, could impact the safety analyses and the adequacy of the protection functions performed by the safety grade equipment. Enclosed is a copy of IE Information Notice 79-22, and reprinted copies of an August 30, 1979 Westinghouse letter, and a September 10, 1979 Public Service Electric and Gas Company letter which address this matter. Operating Reactor licensees conducted reviews to determine whether such problems could exist at operating facilities.

We are concerned that a similar potential may exist at light water facilities now under construction. You are, therefore, requested to perform a review to determine what, if any, design changes or operator actions would be necessary to assure that high energy line breaks will not cause control system failures to complicate the event beyond your FSAR analysis. Provide the results of your review, including all identified problems and the manner in which you have resolved them to NRR.

The specific "scenarios" discussed in the above referenced Westinghouse letter are to be considered as examples of the kinds of interactions which might occur. Your review should include those scenarios, where applicable, but should not necessarily be limited to them. Applicants with other LWR designs should consider analogous interactions as relevant to their designs.

RESPONSE

See Section 3.11(B).2.1.

Q420.4 The analyses reported in Chapter 15 of the FSAR are intended to demonstrate the adequacy of safety systems in mitigating anticipated operational occurrences and accidents.

Based on the conservative assumptions made in defining these design-basis events and the detailed review of the analysis by the staff, it is likely that they adequately bound the consequences of single control system failures.

To provide assurance that the design basis event analyses adequately bound other more fundamental credible failures you are requested to provide the following information:

1) Identify those control systems whose failure or malfunction could seriously impact plant safety.

2) Indicate which, if any, of the control systems identified in (1) receive power from common power sources. The power sources considered should include all power sources whose failure or malfunction could lead to failure or malfunction of more than one control system and should extend to the effects of cascading power losses due to the failure of higher level distribution panels and load centers.

3) Indicate which, if any, of the control systems identified in (1) receive input signals from common sensors. The sensors considered should include, but should not necessarily be limited to, common hydraulic headers or impulse lines feeding pressure, temperature, level or other signals to two or more control systems.

4) Provide justification that any simultaneous malfunctions of the control systems identified in (2) and (3) resulting from failures or malfunctions of the applicable common power source or sensor are bounded by the analyses in Chapter 15 and would not require action or response beyond the capability of operators or safety systems.

RESPONSE

See Section 7.4.

Q422.01 Please provide the Administrative Controls Section of the Technical Specifications which describes the PSRC supervisory and technical personnel referenced in Section 13.4.1.1.

RESPONSE

The description of the Plant Safety Review Committee is provided in the Administrative Controls Section of the Wolf Creek Generating Station, Unit No. 1, Technical Specifications.

The description of the Plant Safety Review Committee is provided in the Quality Program Mannal.

Q430.1
(8.3)
RSP

Operating experience at certain nuclear power plants which have two cycle turbocharged diesel engines manufactured by the Electromotive Division (EMD) of General Motors driving emergency generators have experienced a significant number of turbocharger mechanical gear drive failures. The failures have occurred as the result of running the emergency diesel generators at no load or light load conditions for extended periods. No load or light load operation could occur during periodic equipment testing or during accident conditions with availability of offsite power. When this equipment is operated under no load conditions insufficient exhaust gas volume is generated to operate the turbocharger. As a result the turbocharger is driven mechanically from a gear drive in order to supply enough combustion air to the engine to maintain rated speed. The turbocharger and mechanical drive gear normally supplied with these engines are not designed for standby service encountered in nuclear power plant application where the equipment may be called upon to operate at no load or light load condition and full rated speed for a prolonged period. The EMD equipment was originally designed for locomotive service where no load speeds for the engine and generator are much lower than full load speeds. The locomotive turbocharged diesel hardly ever runs at full speed except at full load. The EMD has strongly recommended to users of this diesel engine design against operation at no load or light load conditions at full rated speed for extended periods because of the short life expectancy of the turbocharger mechanical gear drive unit normally furnished. No load or light load operation also causes general deterioration in any diesel engine.

To cope with the severe service the equipment is normally subjected to and in the interest of reducing failures and increasing the availability of their equipment EMD has developed a heavy duty turbocharger drive gear unit that can replace existing equipment. This is available as a replacement kit, or engines can be ordered with the heavy duty turbocharger drive gear assembly.

To assure optimum availability of emergency diesel generators on demand. Applicant's who have in place, or order or intend to order emergency generators driven by two cycle diesel engines manufactured by EMD should be provided with the heavy duty turbocharger mechanical drive gear assembly as recommended by EMD for the class of service encountered in nuclear power plants. Confirm your compliance with this requirement.

RESPONSE

WCGS diesel generators are not manufactured by EMD; they are Fairbanks Morse diesel engines.

As discussed in response to USAR Question 430.3 and 9.5.8.2.3, specific guidance has been provided by the diesel manufacturer on procedures for operating the engines at light or no load.

Q430.2
(8.3)

Provide a detail discussion (or plan) of the level of training proposed for your operators, maintenance crew, quality assurance, and supervisory personnel responsible for the operation and maintenance of the emergency diesel generators. Identify the number and type of personnel that will be dedicated to the operations and maintenance of the emergency diesel generators and the number and type that will be assigned from your general plant operations and maintenance groups to assist when needed.

In your discussion identify the amount and kind of training that will be received by each of the above categories and the type of ongoing training program planned to assure optimum availability of the emergency generators.

Also discuss the level of education and minimum experience requirements for the various categories of operations and maintenance personnel associated with the emergency diesel generators.

RESPONSE

See Section 13.2.2.14

Q430.3
(8.3)
RSP

Periodic testing and test loading of an emergency diesel generator in a nuclear power plant is a necessary function to demonstrate the operability, capability and availability of the unit on demand. Periodic testing coupled with good preventive maintenance practices will assure optimum equipment readiness and availability on demand. This is the desired goal.

To achieve this optimum equipment readiness status the following requirements should be met:

a) The equipment should be tested with a minimum loading of 25 percent of rated load. No load or light load operation will cause incomplete combustion of fuel resulting in the formation of gum and varnish deposits on the cylinder walls, intake and exhaust valves, pistons and piston rings, etc., and accumulation of unburned fuel in the turbocharger and exhaust system. The consequences of no load or light load operation are potential equipment failure due to the gum and varnish deposits and film in the engine exhaust system.

b) Periodic surveillance testing should be performed in accordance with the applicable NRC guidelines (R. G. 1.108), and with the recommendations of the engine manufacturer. Conflicts between any such recommendations and the NRC guidelines, particularly with respect to test frequency, loading and duration, should be identified and justified.

c) Preventive maintenance should go beyond the normal routine adjustments, servicing and repair of components when a malfunction occurs. Preventive maintenance should encompass investigative testing of components which have a history of repeated malfunctioning and require constant attention and repair. In such cases consideration should be given to replacement of those components with other products which have a record of demonstrated reliability, rather than repetitive repair and maintenance of the existing components. Testing of the unit after adjustments or repairs have been made only confirm that the equipment is operable and does not necessarily mean that the root cause of the problem has been eliminated or alleviated.

d) Upon completion of repairs or maintenance and prior to an actual start, run, and load test a final equipment check should be made to assure that all electrical circuits are functional, i.e., fuses are in place, switches and circuit breakers are in their proper position, no loose wires, and test loads have been removed, and all valves are in the proper position to permit a manual start of the equipment. After the unit has been satisfactorily started and load tested, return the unit to ready automatic standby service and under the control of the control room operator.

Provide a discussion of how the above requirements have been implemented in the emergency diesel generator system design and how they will be considered when the plant is in commercial operation, i.e., by what means will the above requirements be enforced.

RESPONSE

a) See Section 9.5.8.2.3 System Operation (Emergency Diesel Engine Combustion Air Intake and Exhaust System).

b) WCGS is in compliance with the requirements of Regulatory Guide 1.108. Refer to Section 8.1.4.3 for details.

c) See Section 8.3.1.1.3.

d) See Section 8.3.1.1.3.

Q430.4 (8.3) RSP

The availability on demand of an emergency diesel generator is dependent upon, among other things, the proper functioning of its controls and monitoring instrumentation. This equipment is generally panel mounted and in some instances the panels are mounted directly on the diesel generator skid. Major diesel engine damage has occurred at some operating plants from vibration induced wear on skid mounted control and monitoring instrumentation. This sensitive instrumentation is not made to withstand and function accurately for prolonged periods under continuous vibrational stresses normally encountered with internal combustion engines. Operation of sensitive instrumentation under this environment rapidly deteriorates calibration, accuracy and control signal output.

Therefore, except for sensors and other equipment that must be directly mounted on the engine or associated piping, the controls and monitoring instrumentation should be installed on a free standing floor mounted panel separate from the engine skids, and located on a vibration free floor area. If the floor is not vibration free, the panel shall be equipped with vibration mounts.

Confirm your compliance with the above requirement or provide justification for noncompliance.

RESPONSE

See Section 8.3.1.1.3.

Q430.5 The information regarding the onsite communications system (Section 9.5.2) does not adequately cover the system capabilities during transients and accidents. Provide the following information:

a) Identify all working stations on the plant site where it may be necessary for plant personnel to communicate with the control room or the emergency shutdown panel during and/or following transients and/or accidents (including fires) in order to mitigate the consequences of the event and to attain a safe cold plant shutdown.

b) Indicate the maximum sound levels that could exist at each of the above identified working stations for all transients and accident conditions.

c) Indicate the types of communication systems available at each of the above identified working stations.

d) Indicate the maximum background noise level that could exist at each working station and yet reliably expect effective communication with the control room using:

1) the page party communications systems, and

2) any other additional communication system provided that working station.

e) Describe the performance requirements and tests that the above onsite working stations communication systems will be required to pass in order to be assured that effective communication with the control room or emergency shutdown panel is possible under all conditions.

f) Identify and describe the power source(s) provided for each of the communications systems.

g) Discuss the protective measures taken to assure a functionally operable onsite communication system. The discussion should include the considerations given to component failures, loss of power, and the severing of a communication line or trunk as a result of an accident or fire.

RESPONSE

a) Refer to Section 9.5.2.

b) Refer to Section 9.5.2.

c) Refer to revised Table 9.5.2-1.

d) Refer to Section 9.5.2.

e) Refer to Section 9.5.2.

f) Refer to Section 9.5.2.

g) Refer to Section 9.5.2.

Q430.6 Identify the vital areas and hazardous areas where emergency lighting is needed for safe shutdown of the reactor and the evacuation of personnel in the event of an accident. Tabulate the lighting system provided in your design to accommodate those areas so identified. Include the degree of compliance to Standard Review Plan 9.5.1 regarding emergency lighting requirements in the event of a fire.

RESPONSE

Refer to Section 9.5.3.

Q430.7 (9.5.4) Describe the instruments, controls, sensors and alarms provided for monitoring the diesel engine fuel oil storage and transfer system and describe their function. Discuss the testing necessary to maintain and assure a highly reliable instrumentation, controls, sensors and alarm system and where the alarms are annunciated. Identify the temperature, pressure and level sensors which alert the operator when these parameters are exceeded the ranges recommended by the engine manufacturer and describe what operator actions are required during alarm conditions to prevent harmful effects to the diesel engine. Discuss the system interlocks provided. (SRP 9.5.4, Part III, Item 1).

RESPONSE

All applicable instruments, controls, sensors and alarms for the diesel fuel oil storage and transfer system are shown on USAR Figures 9.5.4-1 and 9.5.6-1, Sheets 1 and 2. See Section 9.5.4.

Q430.8 (9.5.4) The diesel generator structures are designed to seismic and tornado criteria and are isolated from one another by a reinforced concrete wall barrier. Describe the barrier (including openings) in more detail and its capability to withstand the effects of internally generated missiles resulting from a crankcase explosion, failure of one or all of the starting air receivers, or failure of any high or moderate energy line and initial flooding from the cooling system so that the assumed effects will not result in loss of an additional generator. (SRP 9.5.4, Part III, Item 2).

RESPONSE

See Section 3.5.2.5.

Q430.9 (9.5.4) Figure 9.5.4-1 and the FSAR text state that the fuel oil storage tank fill and vent lines are non-seismic. We require these lines to be designed seismic Category I and Quality Group C. Conform your compliance with this position. Also describe the design provisions made to protect the fuel oil storage tank fill and vent lines from damage by tornado missiles. (SRP 9.5.4, Part II).

RESPONSE

The fuel oil storage tank vent and all lines are non-seismic above grade and are seismic Category I below grade (refer to USAR Figure 9.5.4-1). See Section 3.5.3.1.

Q430.10 (9.5.4) Discuss the means for detecting or preventing growth of algae in the diesel fuel storage tank. If it were detected, describe the methods to be provided for cleaning the affected storage tank. (SRP 9.5.4, Part III, Item 4).

RESPONSE

See Section 9.5.4.2.1.

Q430.11 (3.2) (9.5.4) (9.5.5) (9.5.7) (9.5.8)

The FSAR text and Table 3.2-1 states that the components and piping systems for the diesel generator auxiliaries (fuel oil system, cooling water, lubrication, air starting, and intake and combustion system) that are mounted on the auxiliary skids are designed seismic Category I and are ASME Section III Class 3 quality. The engine mounted components and piping are designed and manufactured to DEMA standards, and are seismic Category I. This is not in accordance with Regulatory Guide 1.26 which requires the entire diesel generator auxiliary systems be designed to ASME Section III Class 3 or Quality Group C. Provide the industry standards that were used in the design, manufacture, and inspection of the engine mounted piping and components. Also show on the appropriate P&ID's where the Quality Group Classification changes from Quality Group C.

RESPONSE

Only those components and piping supplied with the standard diesel engine and which either make up an integral part of the engine or whose design and reliability have been proven through years of previous diesel engine service are not Quality Group C. All other piping, tubing, and components are ASME Section III, Class 3. See Table 3.2-1.

The USAR figures for the diesel engine auxiliary systems differentiate between seismic and non-seismic portions of the systems and identify those portions of the systems provided by the diesel engine manufacturer.

The standards used in the design, manufacture, and inspection of the Non-Quality Group C components are the manufacturer's standards, developed from his manufacturing and testing experience. By nature of its design and construction, the engine- mounted piping is considered to provide equivalency to ANSI B31.1 standards.

Q430.12 (9.5.4)

Discuss what precautions have taken in the design of the fuel oil system in locating the fuel oil day tank and connecting fuel oil piping in the diesel generator room with regard to possible exposure to ignition sources such as open flames and hot surfaces. (SRP 9.5.4, Part III, Item 6).

RESPONSE

See Section 9.5.4.

Q430.13 (9.5.4) (9.5.5) (9.5.6) (9.5.7) (9.5.8)

Identify high and moderate energy lines and systems that will be installed in the diesel generator room. Discuss the measures that will be taken in the design of the diesel generator facility to protect the safety related systems, piping and components from the affects of high and moderate energy line failure to assure availability of the diesel generators when needed. (SRP 9.5.4, Part III, Item 8, SRP 9.5.5, Part III, Item 4, SRP 9.5.6, Part III, Item 8; SRP 9.5.7, Part III, Item 3; SRP 9.5.8, Part III, Item 6c).

RESPONSE

See Section 3.5 and 3.6.

Q430.14 (9.5.4)

In section 9.5.4 of the FSAR you state that accumulated sediment and moisture may be withdrawn, prior to adding a new fuel oil, through the sample nozzle to minimize the possibility of degrading the overall quality of the new fuel in the unlikely event that would require replenishment of fuel oil without interrupting operation of the diesel generator. This is unacceptable since the sample nozzle would only permit removal of accumulated moisture but not the sediment. Discuss what provisions that will be made in the design of the fuel oil storage fill system to minimize the creation of turbulence of the sediment in the bottom of the storage tank. Stirring of this sediment during addition of new fuel has the potential of causing the overall quality of the fuel to become unacceptable and could potentially lead to the degradation of failure of the diesel generator. Two methods of minimizing this problem are suggested. 1) Design a fuel oil storage tank fill system that will minimize turbulence in the tank. 2) Cross connect the fuel oil storage tank of each diesel in a manner that will permit supply of fuel oil to either engine from either tank. In this manner one tank could be filled while the other tank supplies fuel to the operating D/G. After filling the tank fuel would not be drawn from the tank for a period of time to permit settling of sediment.

RESPONSE

Refer to Section 9.5.4.2.1.

Q430.15 (9.5.4) You state in Section 9.5.4.3 that diesel oil is normally delivered to the site by tanker truck and if road transportation is unavailable, it can be delivered onsite by railroad tanker. Discuss your sources where diesel quality fuel oil will be available and the distance required to be traveled from the source to the plant. Also discuss how fuel oil will be delivered onsite under extremely unfavorable environmental conditions including maximum probable flood conditions.

RESPONSE

See Section 9.5.4.2.3.

Q430.16 (9.5.4) You state in Section 9.5.4.2 that the diesel generator fuel oil storage tank is provided with an individual fill and vent line. Indicate where these lines are located (indoor or outdoor) and the height these lines are terminated above finished ground grade. If these lines are located outdoors discuss the provisions made in your design to prevent entrance of water into the storage tank during adverse environmental condition including maximum probable flood conditions.

RESPONSE

See Section 9.5.4.2.2.

Q430.17 (9.5.5) Discuss the design margin (excess heat removal capability) included in the design of major components and subsystems of the D/G cooling water system (SRP 9.5.5, Part III, Item I).

RESPONSE

See Section 9.5.5.2.2.

Q430.18 (9.5.5) Provide the results of the failure mode and effects analysis to show that failure of a piping connection between subsystems (engine water jacket, lube oil cooler, governor lube oil cooler, and engine air inter-cooler) does not cause total degradation of the diesel generator cooling water system. (SRP 9.5.5, Part III, Item 1a).

RESPONSE

See Section 9.5.5.3.

Q430.19 (9.5.5) Indicate the measures to preclude long-term corrosion and organic fouling in the diesel engine cooling water system that would degrade system cooling performance, and the compatibility of any corrosion inhibitors or antifreeze compounds used with the materials of the system. Indicate if the water chemistry is in conformance with the engine manufacturers recommendations. (SRP 9.5.5, Part III, Item 1c.)

RESPONSE

See Section 9.5.2.2.

Q430.20 (9.5.5) You stated in Section 9.5.5.2.3 the diesel engine cooling water is treated as appropriate to minimize corrosion. Provide additional details of your proposed diesel engine cooling water system chemical treatment, and discuss how your proposed treatment complies with the engine manufacturers recommendations. (SRP 9.5.5, Part III, Item 1c).

RESPONSE

See Section 9.5.2.2.

Q430.21 (9.5.5) Describe the instrumentation, controls, sensors and alarms provided for monitoring of the diesel engine cooling water system and describe their function. Discuss the testing necessary to maintain and assure a highly reliable instrumentation, controls, sensors, and alarm system, and where the alarms are annunciated. Identify the temperature, pressure, level, and flow (where applicable) sensors which alert the operator when these parameters exceed the ranges recommended by the engine manufacturer and describe what operator actions are required during alarm conditions to prevent harmful effects to the diesel engine. Discuss the systems interlocks provided. (SRP 9.5.6, Part III, Item 1c).

RESPONSE

See Section 9.5.5.5.

Q430.22 (9.5.5) In Section 9.5.8.2 of the FSAR, you state that "To reduce the possibility of accumulation of combustion and lube oil products in the exhaust system at the lower loads, the engine will be operated at 50 percent or higher loads for short periods at stipulated time intervals as recommended by the engine manufacturer. Provide the time duration of the "short periods" and the manufacture's recommended "time intervals". We require that this "light load or no load operation" procedure be made part of plant operating procedures. Confirm your compliance with this position.

RESPONSE

Refer to Response to 430.3.

Light load or no load operation is addressed in plant operating procedures.

Q430.23 (9.5.6) Provide a discussion of the measures that have been taken in the design of the standby diesel generator air starting system to preclude the feeling of the air start valve or filter with moisture and contaminants such as oil carryover and rust. (SRP 9.5.6, Part III, Item 1).

RESPONSE

See Section 9.5.6.2.1.

Q430.24 (9.5.6) Describe the instrumentation, controls, sensors and alarms provided for monitoring the diesel engine air starting system, and describe their function. Describe the testing necessary to maintain a highly reliable instrumentation, control, sensors and alarm system and where the alarms are annunciated. Identify the temperature, pressure and level sensors which alert the operator when these parameters exceed the ranges recommended by the engine manufacturer and describe any operator actions required during alarm conditions to prevent harmful affects to the diesel engine. Discuss system interlocks provided. Revise your FSAR accordingly. (SRP 9.5.6, Part III, Item 1).

RESPONSE

See Section 9.5.6.5.

Q430.25 (9.5.6) Expand your description of the diesel engine starting system. The FSAR text should provide a detail system description of what is shown on Figure 9.5.6-1. The FSAR text should also describe: 1) components and their function, 2) instrumentation, controls, sensors and alarms, and 3) a diesel engine starting sequence. In describing the diesel engine starting sequence include the number of air start valves used and whether one or both air start systems are used.

RESPONSE

The diesel engine air start system components and their functions are described in Section 9.5.6.2.2.

Refer to Section 9.5.6.5 for information relating to above (part 2).

System operation is discussed in Section 9.5.6.2.3.

Q430.26 Provide the source of power for the diesel engine air starting system compressors and motor characteristics, i.e., motor hp, operating voltage, phase(s), and frequency. Revise your FSAR accordingly.

RESPONSE

Refer to Table 9.5.6-1 for the response to this question.

Q430.27 (9.5.7) For the diesel engine lubrication system in Section 9.5.7 provide the following information: 1) define the temperature differentials, flow rate, and heat removal rate of the interface cooling system external to the engine and verify that these are in accordance with recommendations of the engine manufacturer; 2) discuss the measures that will be taken to maintain the required quality of the oil, including the inspection and replacement when oil quality is degraded; 3) describe the capability for detection and control of system leakage. (SRP 9.5.7, Part II, Item 8a, 8b, 8c, Part III, Item I.)

RESPONSE

1) Requested information for lube oil cooler is given in Table 9.5.7-1. Design information given in Table 9.5.7-1 is manufacturer's data.

2) See Section 9.5.7.2.1.

3) See Section 9.5.7.2.3.

Q430.28 (9.5.7) What measures have been taken to prevent entry of deleterious materials into the engine lubrication oil system due to operator error during recharging of lubricating oil or normal operation. (SRP 9.5.7, Part III, Item 1c).

RESPONSE

See Section 9.5.7.2.

Q430.29 (9.5.7) Describe the instrumentation, controls, sensors and alarms provided for monitoring the diesel engine lubrication oil system and describe their function. Describe the testing necessary to maintain a highly reliable instrumentation, control, sensors and alarm system and where the alarms are annunciated. Identify the temperature, pressure and level sensors which alert the operator when these parameters exceed the ranges recommended by the engine manufacturer and describe any operator action required during alarm conditions to prevent harmful effects to the diesel engine. Discuss systems interlocks provided. Devise your FSAR accordingly. (SRP 9.5.7, Part III, Item 1c).

RESPONSE

See Section 9.5.7.5.

Q430.30 (9.5.7) Expand your description of the diesel engine lube oil system. The FSAR text should include a detail system description of what is shown on Figure 9.5.7-1. The FSAR text should also describe: 1) components and their function, and 2) a diesel generator starting sequence for a normal start and an emergency start. Revise your FSAR accordingly.

RESPONSE

Refer to USAR Sections 9.5.7.2.1 through 9.5.7.2.3.

Q430.31 (9.5.7) Provide the source of power for the diesel engine prelube oil pump, lube oil transfer pump, clean lube oil transfer pump and used lube oil tank transfer pump, and motor characteristics, i.e., motor hp, operating voltage, phase(s) and frequency. Also provide the pump capacity and discharge head. Revise your FSAR accordingly.

RESPONSE

The WCGS diesel engine is equipped with a main lube oil pump, an auxiliary lube oil (keep warm) pump, a rocker lube oil pump, and a rocker prelube pump. Refer to USAR Table 9.5.7-1 for the requested information.

Q430.32 In Section 9.5.7.2 of the FSAR you state that prelubrication of the rocker arm assembly during standby conditions is done periodically in accordance with the engine manufacturer's recommendations. Provide the following:

a) (RSP) We require that the electric prelube pump automatically prelube the rocker arm assembly and that alarms be provided which alert the operator of pump failure to start on automatic prelubrication.

b) Provide the manufacturer's periodic prelubrication recommendations.

c) Discuss how the lubricating oil in the rocker arm assembly lubrication system is cooler during engine operation and kept warm to enhance engine starting during standby operation.

RESPONSE

See Section 9.5.7.2.1 and 9.5.7.2.3.

Q430.33 (9.5.8) Describe the instrumentation, controls, sensors and alarms provided in the region of the diesel engine combustion air intake and exhaust system which alert the operator when parameters exceed ranges recommended by the engine manufacturer and describe any operator action required during alarm conditions to prevent harmful effects to the diesel engine. Discuss systems interlocks provided. Revise your FSAR accordingly. (SRP 9.5.8, Part III, Item 1 & 4).

RESPONSE

See Section 9.5.8.5.

Q430.34 (9.5.8) Provide the results of an analysis that demonstrates that the function of your diesel engine air intake and exhaust system design will not be degraded to an extent which prevents developing full engine rated power or cause engine shutdown as a consequence of any meteorological or accident condition. Include in your discussion the potential and effect of other gases that may intentionally or accidentally be released on site, on the performance of the diesel generator. (SRP 9.5.8, Part III, Item 3).

RESPONSE

See Section 9.5.8.2.3.

Q430.35 (9.5.8) Discuss the provisions made in your design of the diesel engine combustion air intake, D/G supply ventilation system, and exhaust system to prevent possible clogging, during standby and in operation, from abnormal climatic conditions (heavy rain, freezing rain, dust storms, ice and snow) that could prevent operation of the diesel generator on demand. (SRP 9.5.8, Part III, Item 5).

RESPONSE

See Section 9.5.8.2.3.

Q430.36 (9.5.8) Figure 1.2-1 of the Callaway (and Wolf Creek) FSAR shows the ESF transformers located near the control/ diesel generator building complex. An ESF transformer fire with the right meteorological conditions could degrade engine operation by the products of combustion being drawn into the D/G ventilation system which supplies D/G combustion air. Discuss the provisions of your design (site characteristics, ventilation system and building design, etc.) which preclude this event from occurring.

RESPONSE

See Section 9.5.8.2.3.

Q430.37 (9.5.8) Experience at some operating plants has shown that diesel engines have failed to start due to accumulation of dust and other deleterious material on electrical equipment associated with starting of the diesel generators (e.g., auxiliary relay contacts, control switches - etc.) Describe the provisions that have been made in your diesel generator building design, electrical starting system, and combustion air and ventilation air intake design(s) to preclude this condition to assure availability of the diesel generator on demand.

Also describe under normal plant operation what procedure(s) will be used to minimize accumulation of dust in the diesel generator room; specifically address concrete dust control. In your response also consider the condition when Unit 1 is in operation and Unit 2 is under construction (abnormal generation of dust).

RESPONSE

See Section 9.5.8.2.2.

Q430.38 (9.5.8) (RSP) Section 9.5.8.2.2 and 3.2.2 of the FSAR state that the portions of the EDEAIES outside the D/G building are non-seismic and Quality Group D. This is unacceptable. We require that these portions of the system also be designed seismic Category I and Quality Group C. In addition we required also that the exhaust stacks located outside the D/G building be tornado missile protected. Separation by distance does not constitute adequate protection. Confirm your compliance with these positions.

RESPONSE

See Section 3.5.

Q430.39 (10.1) Provide a general discussion of the criteria and bases of the various steam and condensate instrumentation systems in Section 10.1 of the FSAR. The FSAR should differentiate between normal operation instrumentation and required safety instrumentations.

RESPONSE

The criteria and bases of the various steam and condensate instrumentation are to monitor system variables to provide maximum plant availability, automatic control of equipment and identification of abnormal conditions. Sections 7.3, 7.4, and 7.5 describe the required safety instrumentation associated with Section 10.1. The remaining steam and condensate instrumentation systems included in Section 10.1 are nonsafety-related and are used for normal operation.

Q430.40 (10.2) The FSAR discusses the main steam stop and control, and reheat stop and intercept valves. Show that a single failure of any of the above valves cannot disable the turbine overspeed trip functions. (SRP 10.2, Part III, Item 3).

RESPONSE

Section 10.2.2.3.2 describes the component redundancy which precludes single failure of any main stop, control, intermediate stop, and intercept valve from resulting in rotor speed exceeding design overspeed. All the above valves have independent operating controls and mechanisms.

Q430.41 (10.2) In the turbine generator section discuss: 1) the valve closure times and the arrangement for the main steam stop and control and the reheat stop and intercept valves in relation to the effect of a failure of a single valve on the overspeed control functions; 2) the valve closure items and extraction steam valve arrangements in relation to stable turbine operation after a turbine generator system trip; 3) effects of missiles from a possible turbine generator failure on safety related systems or components. (SRP 10.2, Part III, Items 3, 4.)

RESPONSE

See Section 10.2.2.2. Main stop and control valves, intermediate stop, and intercept valves' closure times are provided. Extraction nonreturn valves are free swinging and close on decreasing flow as described in Section 10.2.2.2. Valve arrangements and single failure effects plus stable turbine operation after a trip are described in Sections 10.2.2.2 and 10.2.2.3.2, Table 10.2-1, and Figure 10.4-6. Turbine missiles are discussed in Section 3.5.1.3.

Q430.42 (10.2) Discuss the effects of a high and moderate energy piping failure or failure of the connection from the low pressure turbine to condenser on nearby safety-related equipment or systems. Discuss what protection will be provided the turbine overspeed control system equipment, electrical wiring and hydraulic lines from the effects of a high or moderate energy pipe failure so that the turbine overspeed protection system will not be damaged to preclude its safety function. (SRP 10.2, Part III, Item 3).

RESPONSE

The turbine overspeed protection system is not safety-related. The ultimate protection from turbine missiles is discussed in Section 3.5.1. No high/moderate energy pipe break or hazards analysis is performed for nonsafety-related turbine building piping or components. See Section 10.2.2.3.2.

Figures 1.2-32 and 1.2-33 show the physical separation between redundant stop/control valves and intermediate stop/intercept valves. Fail safe design of the ETS hydraulic system and the trip power circuitry provide additional turbine overspeed protection. Failure of the low pressure turbine/condenser connection will draw air into the condenser and increase turbine backpressure until trip occurs as stated in Section 10.2.2.3.4.

Q430.43 (10.2) Describe with the aid of drawings, the bulk hydrogen storage facility including its location and distribution system. Include the protective measures considered in the design to prevent fires and explosions during operations such as filling and purging the generator, as well as during normal operations.

RESPONSE

See Section 2.2.1.2.4.2.

Q430.44 (10.4.1) Provide a tabulation in your FSAR showing the physical characteristics and performance requirements of the main condensers. In your tabulation include such items as; 1) the number of condenser tubes, material and total heat transfer surface, 2) overall dimensions of the condenser, 3) number of pauses, 4) hot well capacity, 5) special design features, 6) minimum heat transfer, 7) normal and maximum steam flows, 8) normal and maximum cooling water temperature, 9) normal and maximum exhaust steam temperature with no turbine by-pass flow and with maximum turbine by-pass flow, 10) limiting oxygen content in the condensate in cc per liter, and 11) other pertinent data. (SRP 10.4.1, Part III, Item 1).

RESPONSE

Table 10.4-1 has been revised to include the requested information.

Q430.45 (10.4.1) Discuss the measures taken; 1) to prevent loss of vacuum, and 2) to prevent corrosion/erosion of condenser tubes and components. (SRP 10.4.1, Part III, Item 1).

RESPONSE

Measures taken to prevent loss of vacuum and the Section describing them include:

a) Hydrostatic test of condenser shell (10.4.1.4).

b) Water seal for the LP turbine/condenser connection expansion joint with level indication (10.4.1.2).

c) Operation of condenser vacuum pumps (10.4.2).

d) Control room indication of circulating water pump status (Section 10.4.5).

Measures taken to prevent corrosion/erosion of condenser tubes and components:

a) Provision of 304 stainless steel tubes in the impingement areas of all tube bundles (Table 10.4-1).

b) Feedwater/circulating water chemistry control (Section 10.3.5 and 10.4.5).

Q430.46 (10.4.1) Indicate and describe the means of detecting and controlling radioactive leakage into and out of the condenser and the means for processing excessive amounts. (SRP 10.4.1, Part III, Item 2).

RESPONSE

The means of detecting, controlling, and processing radioactive leakage into and out of the condenser resulting from a steam generator tube leak are discussed in Chapter 11.0. The means for detecting and controlling radioactive leakage into and out of the condenser are described in Sections 11.5.2.2.2.2, 11.5.2.2.2.3, 11.5.2.2.3.4, and 11.5.2.3.2.1. Processing of excessive radioactive leakage is discussed in Sections 11.2.2 and 11.3.2.

Q430.47 (10.4.1) Discuss the measures taken for detecting and controlling and correcting condenser cooling water leakage into the condensate stream. (SRP 10.4.1, Part III, Item 2).

RESPONSE

The measures taken for detecting, controlling, and correcting condenser cooling water leakage into the condensate stream are discussed in Section 10.4.1.

Q430.48 (10.4.1) Provide the permissible cooling water inleakage and time of operation with inleakage to assure that condensate/feedwater quality can be maintained within safe limits. (SRP 10.4.1, Part III, Item 2).

RESPONSE

The information is provided in Section 10.4.6, Condensate Cleanup System.

Q430.49 (10.4.1) In Section 10.4.1.4 you have discussed tests and initial field inspection but not the frequency and extent of inservice inspection of the main condenser. Provide this information in the FSAR. (SRP 10.4.1, Part II).

RESPONSE

See Section 10.4.1.4.

Q430.50 (10.4.1) Indicate what design provisions have been made to preclude failures of condenser tubes or components from turbine by-pass blowdown or other high temperature drains into the condenser shell. (SRP 10.4.1, Part III, Item 3).

RESPONSE

See Section 10.4.1.2.3.

Q430.51 (10.4.1) Discuss the effect of loss of main condenser vacuum on the operation of the main steam isolation valves (SRP 10.4.1, Part III, Item 3).

RESPONSE

Loss of main condenser vacuum does not trip the main steam isolation valves. Loss of main condenser vacuum trips the turbine and blocks turbine by-pass. Turbine trip at power levels above 50 percent results in a reactor trip as described in Section 7.2. The effects of potential failure modes on the NSSS and turbine system are addressed in Sections 15.1.4, 15.2.3, and 15.2.5.

Q430.52 (10.4.4) Provide additional description (with the aid of drawings) of the turbine by-pass system (condenser dump valves and atmosphere dump valves) and associated instruments and controls. In your discussion include: 1) the size, principle of operation, construction and set points of the valves, 2) the malfunctions and/or modes of failure considered in the design of the system.

RESPONSE

Condenser Dump Valves

Section 10.4.4.2.1, 10.4.4.2.2 and Figure 10.3-1, Sheet 3 provide a description of the turbine bypass system and the condenser dump valves. Section 7.7.1.8 and Figures 7.2-1, Sheet 10 and 10.3-1, Sheet 3 describe the associated instruments and controls. The malfunctions and failure modes considered in system design and their effect on the NSSS and turbine system are addressed in Sections 15.1.4 and 15.2.3.

Steam Generator Atmospheric Relief Valves

Section 10.3.2.2, Table 10.3-2 and Figure 10.3-1, Sheet 1 provide a description of the steam generator atmospheric relief valves. The valves are opened by pneumatic pressure and closed by spring action as stated in Section 10.3.2.2. Section 7.4.1.2 and Figures 7.2-1, Sheet 10 and 10.3-1, Sheet 1 describe the associated instruments and controls. The malfunctions and failure modes considered in the system design are addressed in Section 7.4.1.2 and Section 15.1.4.

Q430.53 (10.4.4) Section 10.4.4 of the FSAR describes the turbine by-pass system and states that the TBS dumps steam to the condenser through condenser spargers. Figure 10.3.1, Sheet 3 in the FSAR shows the turbine by-pass as described in Section 10.4.4. It also shows six 3 inch lines branching off the TBS lines upstream of the TBS valves. These lines are labelled "To Condenser Sparger" and seem to have normally open valves. Explain the purpose of these lines and the status of these valves.

RESPONSE

The purpose of these lines is to supply steam to the condenser hotwell spargers used for deaeration of the condensate, as described in Sections 10.3.5 and 10.4.1.2.3. The valves in phantom on Figure 10.3.1, Sheet 3 are shown on P&ID M-02AD01 (Figure 10.4.2, Sheet 1) as normally closed.

Q430.54 (10.4.4) In Section 10.4.4.4 you have discussed tests and initial field inspection but not the frequency and extent of inservice testing and inspection of the turbine by-pass system. Provide this information in the FSAR. (SRP 10.4.4, Part II).

RESPONSE

See Section 10.4.4.4.

Q430.55 Provide the results of an analysis indicating that failure of the turbine by-pass system high energy line will not have an adverse effect or preclude operation of the turbine speed control system or any safety-related components or system located close to the turbine by-pass system. (SRP 10.4.4, Part III, Item 4).

RESPONSE

See response to Question 430.42. There is no safety-related equipment in the vicinity of the turbine by-pass system, as stated in Section 10.4.4.3.

Q440.1 Please provide a scheduled completion date for the plant administrative procedures which are referred to in Section 13.5.1.

RESPONSE

This information is provided in Section 13.5.1.2.

Q440.2 Please indicate that you intend to include procedures for design change processing, retest after design changes, and control of plant documents and records in the plant administrative procedures.

RESPONSE

See Section 13.5.1.2.

Q440.3 (Q440.1C) The analyses of a locked reactor coolant pump rotor and a sheared reactor coolant pump shaft in the FSAR assumes the availability of offsite power throughout the event. In accordance with Standard Review Plan 15.3.3 and GDC 17, we require that this event be analyzed assuming turbine trip and coincident loss of offsite power to the undamaged pumps.

Appropriate delay times may be assumed for loss of offsite power if suitably justified.

Steam generator tube leakage should be assumed at the rates specified in the Technical Specifications.

The event should also be analyzed assuming the worst single failure of a safety-system active component. Maximum technical specification primary system activity and steam generator tube leakage should be assumed. The analyses should demonstrate that offsite doses are less than 10 CFR 100 guidelines values.

RESPONSE

See Section 15.3.3 for additional information.

Q440.106 (5.2.2) In reviews of certain other Westinghouse-designed plants, a failure of a D.C. power bus was identified which could both initiate an overpressure event at low temperature (by isolating letdown) and fail closed one of the PORVs. A postulated single failure (closed) of the other PORV would fail mitigating systems for this event. Address this scenario for the SNUPPS design.

RESPONSE

See Section 5.2.2.3 for additional information.

Q440.207 The NRC wanted to know if the solid water condition between RHR suction valves could, because of heating, expand and cause system damage or valve inoperability.

RESPONSE

RHR suction valve seat leakage is expected to prevent system damage or valve inoperability resulting from contained fluid thermal expansion.

Q450.0 (6.4) In your description of the control room habitability system, include the provisions for emergency food, water and medical supplies.

RESPONSE

See Table 6.4-1, Position 15.

Q450.1 (6.4) In the evaluation of toxic gas protection, document the degree of leak-tightness of the control room isolation dampers.

RESPONSE

The total leak-tightness of the control room and its potential leakage paths are discussed in USAR Section 9.4.1.2.3 under EMERGENCY OPERATION. Also see Section 9.4.1.2.2.

Q450.2 (6.4) Provide a description and drawing showing the locations of control room outside air inlets relative to potential radiation releases.

RESPONSE

See Section 9.4.1.2.3.

Q450.3 In your analysis of toxic gas protection for Control Room Personnel, provide the number and type of respiratory devices, the type of operator training for respiratory use, the estimated time for donning or deploying the equipment, the length of time the equipment can be used, and the equipment testing and maintenance provisions.

RESPONSE

See Table 6.4-1, Item 13.

Q450.4 (6.4) List the areas, equipment and materials in the zone serviced by the control room emergency ventilation system.

RESPONSE

The control room ventilation systems are described in Sections 6.4.2 and 9.4.1.

Q450.5 (6.4) Discuss how the control room design precludes the buildup of noxious gases from control room equipment such as gases from batteries.

RESPONSE

See Section 6.4.2.4.

Q450.6 (6.4) In Section 6.4.5, the testing and inspection of the control room habitability systems is described. In particular, the last paragraph states: "The control room is classified as Type B per Regulatory Guide 1.78. Since the air exchanger rate exceeds 0.06 air exchanges per hour for the control room, periodic testing of the control room pressurization system is not required per the exclusion provisions of the Regulatory Guide."

Apparently, there is some confusion as to the applicability of Regulatory Guide 1.95 (and 1.78) to the control room ventilation design for radiological protection. For a control room outside air makeup rate during emergency pressurization less than 0.25 volume change per hour (as in Callaway), SRP Section 6.4 recommends the following:

a) acceptance test to verify adequate pressure,

b) supporting calculations to verify adequate air flow, and

c) periodic verification testing.

If this guidance is not followed, justify the departures.

RESPONSE

a) See USAR Section 14.2.12.1.45.

b) See Section 6.4.2.3 and 9.4.1.2.3.

c) See Section 6.4.5.

Q450.7 (6.5.2) In Section 6.5.2.2.3 of the SNUPPS FSAR, it stated that the containment spray system recirculation flow is manually initiated. It is the staff's position that the containment spray switchover be automatic. Justify your departure from this position.

RESPONSE

See Section 6.5.2.2.3.

Q450.8 (15.4.8(A)) With respect to rod ejection accident, provide the transient time for the depressurization of the primary system to the termination of primary to secondary leakage.

RESPONSE

See Section 15.4.8.1.1.

Q450.9 (15.6.3) The following information is currently missing from the Callaway FSAR and is needed to complete our review. For the steam generator tube rupture accident provide the following figures:

a) SGTR break flow rate vs Time

b) SGTR integrated tube leak mass vs Time

c) Primary system pressure vs Time

d) Secondary system pressure vs Time

e) PORV flow rate vs Time

f) MS safety valve flow rate per steamline vs Time

g) Atmospheric dump valve flow rate vs Time

h) Steam generator steaming rate vs Time

i) Reactor coolant temperature vs Time

j) Feedwater flow rate into the steam generators vs Time

k) Water level in the affected steam generator relative to the top of the tube bundle vs Time.

Also, provide the mass of secondary coolant in a steam generator.

RESPONSE

Refer to Section 15.6.3.

Q450.10 (6.5.2) (RSP)

The SNUPPS FSAR indicates that the mode of initiation of switchover of the containment spray system suction from the Refueling Water Storage Tank to the containment sump is manual. The staff finds that this practice departs from that currently deemed acceptable. SRP Section 6.5.2 (II. Acceptance Criteria, Item 2.a) states "The Containment spray system should be designed...and should be capable of continuous operation thereafter until the design objectives of the system have been achieved. In all cases the operating period should not be less than two hours." Manual initiation of the switchover does not guarantee continuous operation for two hours and does not provide assurance that the design objectives of the spray system are achieved for delayed fission product releases from the core. It is the staff's position that we require a design modification which will change from manual to automatic the switchover of the containment spray system from the RWST to the containment sump. State your intent regarding compliance with our position.

RESPONSE

See Section 6.5.2.2.3.

Q451.0 ACCIDENT EVALUATION BRANCH

Q451.1 Please provide hour-by-hour meteorological data for the periods 6/1/73 - 5/31/75 and 3/5/79 - 3/4/80 on magnetic tape using the enclosed guidance on format and tape attributes.

RESPONSE

This data was forwarded to the NRC on 6/1/81.

Q451.2 Describe the status of the onsite meteorological measurements program since 3/4/80 and provide additional data for the period 3/5/80 - 3/4/81, if available.

RESPONSE

See Section 2.3.3.

Q451.3 Table 2.3-37 (Rev. 1, 2/81) of the FSAR indicates that extremely unstable (Pasquill Type A), moderately stable (Pasquill Type F), and extremely stable (Pasquill Type G) conditions have persisted for long durations (e.g., greater than 12 hours) at the WCGS site. Apparently, extremely unstable conditions persisted for a 24-hour period during the Phase 2 program. Persistence of these stability classes for periods greater than 12 hours in duration is very unusual. Discuss the causes of persistent stability conditions for periods greater than 12 hours for classes A, F, and G. Identify the synoptic conditions during the observed periods of persistent stability for periods greater than 12 hours and discuss the possibility of instrument malfunction.

RESPONSE

See Section 2.3.2.1.7.

Q451.4 Table 2.3-29 (Rev. 1, 2/81) of the FSAR indicates a lower data recovery for joint frequency distributions of wind speed and wind direction by atmospheric stability for the period 3/5/79 - 3/4/80 than for the previous two years of data collection (6/1/73 - 5/31/75) despite increased attention to the onsite meteorological program. The major difference between the Phase 1 (6/1/73 - 5/31/75) program and the Phase 2 program (3/5/79 - 3/4/80) appears to be the type of data recording system, with the Phase 2 system consisting solely of analog charts. Discuss the reasons for the lower data recovery and indicate whether complete reliance on an analog recording system could be a major factor in reduced data recovery. Identify periods of extended instrument outage (e.g., for 24 hours or more) during the Phase 2 program and the cause of the outage. Indicate the corrective measures taken to minimize extended outages in the future. Describe the data availability (e.g., remote display in the control room or elsewhere) and data reduction procedures to be used for the meteorological measurements program during plant operation.

RESPONSE

See Section 2.3.3.7.2.

Q451.5 Section 2.3.2.2 (Rev. 1, 2/81) of the FSAR (see also Revision 1, 4/81 to the Environmental Report Section 5.1.4) presents an analysis of the atmospheric impacts of the heat dissipation facilities using the model FOGALL. This analysis replaces the previous analysis based on the model POND.

a) Describe the improvements in the analysis using FOGALL compared to the analysis using POND.

b) Describe the validation (or verification) of FOGALL for analyzing atmospheric impacts of a 5090 acre cooling lake.

c) Describe the meteorological measurements program to be used to evaluate actual meteorological impacts of the heat dissipation system once the cooling lake is filled and the plant is operational.

RESPONSE

a) See Section 2.3.2.2.

b) See Section 2.3.2.2.

c) See Section 2.3.2.2.

Q451.6 Section 2.3.2.2 (Rev. 1, 2/81) of the FSAR also discusses the effect of the cooling lake on atmospheric transport and diffusion and concludes "for winds less than about 6 mph flowing from or into this sector [south-southwest to south-southeast] (and less than 2 mph in any sector over the lake) modifications in the atmospheric stability of the diffusion properties of the air may be expected." Winds less than about 6 mph blowing from or into the south-southwest to south-southeast sector occur about 13% of the time. Discuss the modifications to transport and dispersion characteristics during these conditions and indicate if the calculations in Sections 2.3.4 and 2.3.5 of the FSAR should be changed to reflect the modified dispersion conditions.

RESPONSE See Section 2.3.2.2.

Q451.7 Tables 2.3-59 and 2.3-60 of the FSAR (Rev. 1, 2/81) present terrain/recirculation correction factors to be applied to a straight-line Gaussian dispersion model to better characterize temporal variations in meteorological conditions. These correction factors were estimated based on the results of a variable-trajectory puff advection model using one year of hour-by-hour meteorological data from the Wolf Creek site. Substantial reductions (up to a factor of 100 lower than the straight-line model) are suggested for distances approaching 80 km. For several directions, correction factors of zero are suggested, implying that no release from the site would affect a particular receptor location. Discuss the reasonableness and appropriateness of correction factors for receptors greater than 8 km from the source developed by use of a variable trajectory model with only a single source of meteorological data as input. Indicate the merit of a correction factor calculated to be zero.

RESPONSE

See Section 2.3.5.1.4.

Q451.8 The expected number of lightning strikes to ground per year in a square mile area surrounding the site could be as high as 46 (p. 2.3-8 of the FSAR). Provide seasonal and annual estimates of lightning strikes to safety-related structures at the site, considering the "attractive area" of the structures. A suggested reference for this type of analysis is J. L. Marshall, Lightning Protection, 1973.

RESPONSE

See Section 2.3.1.2.5.

Q451.9 The tornado statistics presented in Section 2.3.1.2.6 are based on a regional data base that ended in 1971. Identify any tornadoes that have occurred in the vicinity of the site since 1971, and provide estimates of the intensity (maximum wind speed) and path area of each.

RESPONSE

See Section 2.3.1.2.6.

Q451.10

a) Describe the procedures used for determining "the worst temperature period" and "the worst evaporation period" (Table 2.3-9 A and B) used for the analysis of the ultimate heat sink.

b) Regulatory Guide 1.27 (Rev. 2) recommends that the meteorological conditions used for analysis of the ultimate heat sink be selected from a recent 30-year period. Only 16 years of data from Chanute Flight Service Station were used in this evaluation (p. 2.3-12). Explain why 16 years of data (1949 through 1964) is considered representative of regional climatological conditions for analysis of the ultimate heat sink.

RESPONSE

a) See Section 9.2.5.3.

b) See Section 2.3.1.2.10.

Q451.11 Review of the hour-by-hour meteorological data provided on magnetic tape in responses to question 451.1 indicates a number of concerns. First, the tape has been mislabeled so that the intervals for measurement of vertical temperature gradient are incorrectly identified. Second, a sizable fraction of the recorded temperature gradient measurements exceed the auto-convective lapse rate. Third, occasionally the temperature difference measured between the 10m and 60m levels is considerably different than that measured between the 10m and 85m levels. For example, on Julian day 160 1979, the temperature difference between the 10m and 60m levels indicated a moderately unstable (Pasquill Type "B") condition while a slightly stable (Pasquill Type "E") condition was indicated by the temperature difference between the 10m and 85m levels. Finally, 45% of moderately stable (Pasquill Type "F") and 30% of extremely stable (Pasquill Type "G") conditions occur with wind speeds greater than 3m/sec. Similarly, 60% of extremely unstable (Pasquill Type "A") conditions occur with wind speeds greater than 3m/sec. Occurrences of extremely unstable, moderately stable, and extremely stable conditions usually predominate during low wind speeds (i.e., less than 1.5m/sec).

a) Provide a new magnetic tape of corrected hour-by-hour meteorological data for the 3 year period of record in the format requested in question 451.1. All invalid data (see b and c below) should be properly identified.

b) Provide a description of the quality control checks used to identify invalid hourly data. Discuss the validity of occurrences of temperature gradients exceeding the auto-convective lapse rates and the occurrences of considerably different stability conditions indicated by temperature gradients measured between the 10m and 60m levels and those measured between the 10m and 85m levels.

c) Discuss the validity of the relatively large number occurrences of extremely unstable, moderately stable, and extremely unstable conditions with wind speeds greater than 3m/sec.

RESPONSE

a) The revised data was submitted.

b) See Section 2.3.3.7.2.

c) See Section 2.3.2.7.2.

471.0 RADIOLOGICAL ASSESSMENT BRANCH

Q471.1 Please describe your plan to provide onsite backup coverage in the event of the absence of the site Health Physicist and outline the qualifications (or make reference to them in the appropriate section of the FSAR) of the individual who will act as the backup. It is our position that this individual have a B.S. degree in science or engineering, and two years health physics experience, one year of which should be nuclear power plant experience, with six months of this experience being onsite. It is our position that this experience be health physics experience.

RESPONSE

See Section 12.5.1.

Q471.2 Section 13.1.2.3 "Shift Crew Composition" states that this area will be addressed in the Technical Specification. The staff requires that an H.P. technician will be onsite at all times, in accordance with NUREG-0654 "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants", after the reactor is at power. Please state your intentions for having your technical specification include a H.P. technician as part of the shift crew. The qualifications of the H.P. technician are described in ANSI 18.1.

RESPONSE

See the WCGS Technical Specifications.

Q471.3 In accordance with 12.5.3 several procedures including respiratory protection, decontamination, glove boxes, tents, etc. will be used to reduce possibility of personnel exposure to airborne activity. Please discuss your radiation protection provision for installation of temporary flexible ducting and monitoring equipment at the site of maintenance operations and repair activities, if a high potential for airborne radioactivity exists, to assure that 10 CFR Part 20.103 limits are not excluded, that 10 CFR 20.103(b) actions are taken, and that exposure are maintained ALARA during the operation.

RESPONSE See Section 12.5.3.

Q471.4 Table 12.5-2 "Portable Health Physics Equipment" show quantities of instrumentation not adequate to meet the anticipated needs of a radiation protection program for a nuclear power plant. The staff position is that sufficient numbers of instrumentation be available in operating condition to accommodate the need to monitor such large numbers of operations that may be required in radiation areas and high radiation areas throughout the plant during major maintenance and refueling outages and/or accidents. In arriving at a total number, consideration should also be given to the survey instruments that may be in a calibration, maintenance or inoperative-on-the-shelf status during the outage and/or accidents. Additionally, the inventory should include the requirements for selected ranges, sensitivities, types of radiation to be monitored, accuracy required and types of monitoring to be performed. Ten instruments that read-out in the R/hr range of measurements, as shown in Table 12.5.2, would probably not satisfy the above criteria based on the findings at operating nuclear power plants. Therefore, the table should be revised to reflect these criteria in order to provide the radiation protection instrumentation inventory requirements of the plant.

RESPONSE

See Table 12.5-2.

Q490.1 Since the issuance of Construction Permits for SNUPPS plants, several significant changes have taken place that will affect our review of Section 4.2, "Fuel System Design." The most fundamental changes deal with the format and content of Section 4.2 as they relate to the Standard Review Plan; the other changes deal with technical issues that have arisen recently. All of these changes are discussed below.

Standard Review Plan

The basic fuel sections of the Standard Format (Rev. 3), the Standard Review Plan (Rev. 1, 1978), and the SNUPPS FSAR are all the same: 4.2.1 Design Bases, 4.2.2 Description and Design Drawings, and 4.2.3 Design Evaluation. Unfortunately, 4.2.1 of the Standard Format (and, hence, of the SNUPPS FSAR) does not clearly call for a quantitative (usually numerical) statement of all design bases as does the Standard Review Plan. Similarly, the other sections of the Standard Format and the SNUPPS FSAR mix up design bases, design descriptions, and design evaluations, but that information is sorted out clearly in the Standard Review Plan.

Because of improvements in clarity and completeness in this 1978 version of the Standard Review Plan, we will conduct our review and prepare the SER according to the SRP. Our questions, then, will not be open-end, but they will simply ask for the residual information called for in the SRP but not present in the SNUPPS FSAR. There are, thus, two options at this stage of the review.

Option 1 - You could revise Section 4.2 of the SNUPPS FSAR to follow the details of the SRP (remember, the basic organization structure would be unchanged). This would automatically bring out all of the information that is needed.

Option 2 - A cross reference could be provided to link each item in the SRP with a paragraph in the SNUPPS FSAR. This method would leave Section 4.2 of the SNUPPS FSAR in its present format, but might lead to additional questions since all of the information is not present.

We recommend Option 1. Revision 1 of the SRP, to which we refer, was formally issued more than two years ago. Therefore, we do not view this change as either precipitous or disruptive. Furthermore, it is likely that you will have to identify and justify all deviations from the SRP under the provisions of a proposed rule (Federal Register 45, p. 67099, October 9, 1980) since your SER will be issued after January 1, 1982.

We urge you to provide the information that would be needed to demonstrate compliance with the SRP at your earliest convenience. To help you anticipate an imminent revision to SRP-4.2, the following comments are provided.

Revision 1 - This revision was issued in October 1978 and contains all of the basic requirements that you need to address. It will not be changed significantly by the planned revision.

Revision 2 - This revision is planned for April 1981 and is the revision alluded to in the notice of proposed rulemaking on SRP compliance. In SRP-4.2 this revision will (a) add acceptance criteria for mechanical response to seismic and LOCA loads, and (b) make editorial change largely confined to adding and correcting citations to regulations and regulatory guides that are already addressed in Rev. 1. The acceptance criteria for mechanical response were recently implemented as part of the resolution of Unresolved Safety Issue, Task A-2 and are given in Appendix E of NUREG-0609. Therefore, you can base the SNUPPS FSAR revisions on SRP-4.2 Rev. 1 (current version) plus Appendix E of NUREG-0609, and last-minute changes in referencing can be made in April prior to your submittal of the additional fuel-related information.

Recent Technical Issues

The following is a list of current technical issues that have frequently been noted as outstanding issues in recent SERs and that should be given special attention in the SNUPPS FSAR.

1. Supplemental ECCS analysis with NUREG-0630.
2. Combined seismic and LOCA loads analysis.
3. Enhanced fission gas release analysis at high burnups.
4. Fuel rod bowing and analysis.
5. Fuel assembly control rod guide tube wear analysis.
6. Fuel assembly design shoulder gap analysis.
7. End-of-life fuel rod internal pressure analysis.

RESPONSE

A. See Section 4.2, 4.2.3, 15.4 and 15.6.

B. See Table 4.1-1, 4.3-1, Section 4.2.2.1, Figures 4.2-1 through 4.2-15 and Section 4.2-3.

Recent Technical Issues

With regard to the seven current technical issues presented in question 490.1, it is WCGS's understanding that many of the generic issues have been resolved in connection with NRC staff reviews of similar plants with fuel assembly designs and fuel fabrication specifications that are the same as those for SNUPPS. The following paragraphs address these issues.

1. Supplemental ECCS analysis with NUREG-0630

   Section 6.2.5 describes the ECCS.

2. Combined seismic and LOCA loads analysis

   The combination of seismic effects and loads due to a double ended loss-of-coolant accident are discussed in Section 4.2.3.

3. Enhanced fission gas release analysis at high burnups

   The subject of fission gas release is discussed in Westinghouse topical report WCAP-8720/8785 (Reference 5 in Section 4.2.)

4. <u>Fuel rod bowing analysis</u>

   The subject of fuel rod bowing is discussed in Section 4.2.3 as well as Westinghouse topical report WCAP-8691/8692 (Reference 11 of Section 4.2.)

5. <u>Fuel assembly control rod guide tube wear analysis</u>

   Westinghouse topical report WCAP-8278/8279 (Reference 10 of Section 4.2) presents flow test results for fretting wear at contact points between the control rods and control rod guide thimbles. Additional experimental data has been submitted to the NRC by Westinghouse (see W letters NS-TMA-1936, 1992, and 2102), and a post-irradiation examination program has been established to address this specific subject (see NUREG-0717).

6. <u>Fuel assembly design shoulder gap analysis</u>

   Appropriate rod-to-nozzle gap is provided in the WCGS fuel to accommodate thermal expansion and irradiation-induced growth of the fuel rods relative to the overall fuel assembly structure. Westinghouse's ability to model fuel rod growth has been confirmed by comparison with measurements from 15 x 15 and 17 x 17 in-reactor data, and also is in good agreement with established experimental results as discussed in Reference 1.

7. <u>End-of-life fuel internal pressure analysis</u>

   The internal fuel rod pressure criteria are described in approved Westinghouse topical report WCAP-8963/8964 (Reference 7 to Section 4.2.)

<u>References</u>

1. Balfour, J.B., Destefan, J., Melehan, M.G., and Cerni, S. "Evaluation and Performance of Westinghouse 17 x 17 Fuel," presented at the ANSI Topical Meeting on LWR Fuel Performance held April 30 through May 2, 1979.

Q492.2 The effects of fuel rod bowing must be included in the thermal-hydraulic design. The predicted extent of rod bow (gap closure) versus exposure and the effect of rod bowing on DNBR must be addressed. Use of the staff report "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," February 16, 1977, represents an acceptably conservative treatment of rod bowing.

RESPONSE

See Section 4.3.3.3.1d.

Q492.3 Operating experience on two pressurized water reactors (not of the Westinghouse design) indicate that significant reduction in core flow rate can occur over a relatively short period of time as a result of crud deposition on the fuel rods. In establishing the Technical Specifications for Callaway and Wolf Creek we will require provisions to assure that the minimum design flow rates are not exceeded. Therefore, provide a description of the flow measurements capability for Callaway and Wolf Creek as well as a description of the procedures to measure flow and the actions to be taken in the event of an indication of lower than design flow.

RESPONSE

See Section 4.4.4.7.

Q492.4 The NRC approval of the THINC-IV code, for use in the thermal-hydraulic design, indicates that the pressure gradient at the core exit must be modeled. Provide a revised THINC-IV calculation at the steady state reactor design conditions including the modeling of the core exit radial pressure gradient. Provide the following specific information from that calculation:

1. minimum DNB ratio (value and location)
2. hot channel flow vs. axial position
3. hot channel enthalpy vs. axial position
4. hot channel void fraction vs. axial position
5. the assumed core exit pressure gradient.

RESPONSE

On October 25, 1977, Westinghouse met with the NRC to discuss the effects of nonuniform upper plenum pressure distribution as part of the NRC staff's review of RESAR-414. The Westinghouse material presented at that meeting was transmitted to the NRC via letter NS-CE-1591, dated November 2, 1977, from C. Eicheldinger (Westinghouse) to J. F. Stolz (NRC). This letter addresses the THINC-IV information requested by question 492.4, and is applicable to all Westinghouse 4-loop plants, including the SNUPPS units.

In addition, this issue was pursued further by the NRC during the McGuire FSAR review. The McGuire fuel is identical to the SNUPPS fuel, and the same thermal-hydraulic models and correlations were used. As a result of this review, the staff concluded that this issue was adequately resolved. This conclusion is equally applicable to WCGS.

Q492.5 Insufficient information has been provided to justify the design power level of 2389 Mwt (70% of full power) during three-loop operation. Temperature differences in the active cold legs of a few degrees could exist during three-loop operation. Therefore a radial power tilt and an increase in enthalpy rise factor could result. As a result, we request that a complete detailed description of the following items be provided:

1. The method of determining the temperature distribution among the cold legs and the associated radial power tilt;

2. The method of accounting for differences (if any) in the three-loop thermal-hydraulic design;

3. The instrumentation available and monitoring procedures during three-loop operation;

4. The DNBR Technical Specification and how it will be implemented for three-loop operation;

5. The reactor protective system setpoints related to DNBR protection and how they are generated;

6. The effects of anticipated operational occurrences on the cold leg temperature distributions and how this effect is included in the design.

RESPONSE

This question is not applicable to the SNUPPS Plants, since they do not currently plan to operate in the N-1 mode.

Q492.6 Please state your intent regarding the use of the Westinghouse optimized fuel assembly in your plant. If the use of this design is being considered, provide a discussion of the status and schedule for any revised submittals.

RESPONSE

WCGS does not currently plan to incorporate Westinghouse optimized fuel for the first fuel cycles.

Q492.7 Please state your intent regarding the use of the Westinghouse "Improved Thermal Design Procedure" described in WCAP-8567, dated July, 1975. If you intend to use these methods, responses to the following questions will be required:

(a) Provide a block diagram depicting sensor, process equipment, computer, and readout devices for each parameter channel used in the uncertainty analysis. Within each element of the block diagram, identify the accuracy, drift, range, span, operating limits and setpoints. Identify the overall accuracy of each channel transmitter to final output and specify the minimum acceptable accuracy for use with the new procedure. Also identify the overall accuracy of the output value and maximum accuracy requirements for each input channel of this final output device.

(b) Discuss the method(s) for incorporating environmental effects (e.g., noise, EMI) on instrument channels into the uncertainty analysis.

(c) Provide data to verify that the plant instruments will perform with a high degree of confidence, within their design accuracies. This information may be obtained from operating history of identical instruments installed in other plants. This request pertains to the instruments affecting the uncertainties in the design procedure (as identified in question 1 above), the overtemperature T trip, the high flow trip, the low pressure trip and the pump voltage trip.

(d) Provide the ranges of applicability of sensitivity factors.

(e) Demonstrate that the linearity assumption of equation 3-8 in WCAP-8567 is valid when the WRB-1 correlation is used.

RESPONSE

The Westinghouse Improved Thermal Design Procedure is not currently planned to be used.

Q492.8 Standard format and content of Safety Analysis Reports, Regulatory Guide 1.70, states that in Chapter 4 of the SAR

> "...the applicant provide an evaluation and supporting information to establish the capability of the reactor to perform its safety functions throughout its design lifetime under all normal operation modes..."

Are the analyses presented in Section 4.4 representative of the initial core only or have future cycles been analyzed? Provide a discussion of how power distributions for future cycles are considered in the FSAR analyses. Is there any assurance that the Callaway Units (Wolf Creek) can operate at the licensed power level without excessive DNB trips throughout future cycles? Will revisions to the design methodology be required in order to maintain sufficient thermal margin?

RESPONSE

The goal of the reload safety evaluation is to confirm the validity of the existing safety analysis. The existing safety analysis is defined as the reference safety analysis and is intended to be valid for all plant cycles. Thus safety analysis input parameter values are selected to bound the values expected in all subsequent cycles. This bounding analysis concept is the key to the Westinghouse reload safety analysis methodology. When all reload safety-related parameters for a given accident are bounded, the reference safety analysis is valid. On the other hand, when a reload parameter is not bounded, further evaluation is necessary. The purpose of this further evaluation is to confirm that the margin of safety defined in the basis for any technical specification is not reduced. This reload safety evaluation methodology is applied whenever the input parameter values for a reference safety analysis are available. In summary, Westinghouse reload safety evaluation methodology consists of:

1. A systematic evaluation to determine whether the reload parameters are bounded by the values used in the reference safety analysis.

2. A determination of the effects on the reference safety analysis when a reload parameter is not bounded to ensure that specified design bases are met.

When the above process identifies either a need for a license amendment or a change in the plant Technical Specifications, the Operating Agent will make the appropriate notification to the NRC.

Q492.9 The staff has reviewed the applicants' response to the requirements of Item II.F.2 of NUREG-0737 and found that the applicants have not provided the documentation required by Item II.F.2. Therefore, the staff will require that the applicants provide the documentation required by Item II.F.2 of NUREG-0737.

RESPONSE

See revised Section 18.2.13.

Q492.10 Justify that the single upper head penetration meets the single failure requirement of NUREG-0737 and show that it does not negate the redundancy of the two instrument trains.

<u>RESPONSE</u>

See Section 18.2.13.2.

Q492.11 Describe the location of the level system displays in the control room with respect to other plant instrument displays related to ICC monitoring, in particular, the saturation meter display and the core exit thermocouple display.

<u>RESPONSE</u>

See Section 18.2.13.2.

Q492.12 Describe the provisions and procedures for on-line verification, calibration and maintenance.

<u>RESPONSE</u>

See Section 18.2.13.2.

Q492.13 Describe the diagnostic techniques and criteria to be used to identify malfunctioning components.

<u>RESPONSE</u>

See Section 18.2.13.2.

Q492.14 Estimate the in-service life under conditions of normal plant operations and describe the methods used to make the estimate, and the data and sources used.

<u>RESPONSE</u>

See Section 18.2.13.2.

Q492.15 Explain how the value of the system accuracy (given as +/- 6%) was derived. How were the uncertainties from the individual components of the system combined? What were the random and systematic errors assumed for each component? What were the sources of these estimates?

<u>RESPONSE</u>

See Section 18.2.13.2.

Q492.16 Assume a range of sizes for "small break" LOCA's. What are the relative times available for each size break for the operator to initiate action to recover the plant from the accident and prevent damage to the core? What is the dividing line between a "small break" and a "large break"?

RESPONSE

See Section 18.2.13.2.

Q492.17 Describe how the system response time was estimated. Explain how the response times of the various components (differential pressure transducers, connecting lines and isolators) affect the response time.

RESPONSE

See Section 18.2.13.2.

Q492.18 There are indications that the TMI-2 core may be up to 95% blocked. Estimate the effect of partial blockage in the core on the differential pressure measurements for a range of values from 0 to 95% blockage.

RESPONSE

See Section 18.2.13.2.

Q492.19 Describe the effects of reverse flows within the reactor vessel on the indicated level.

RESPONSE

See Section 18.2.13.2.

Q492.20 What is the experience, if any, of maintaining D/p cells at 300% overrange for long periods of time?

RESPONSE

See Section 18.2.13.2.

Q492.21 Five conditions were identified which could cause the DP level system to give ambiguous indications. Discuss the nature of the ambiguities for 1) accumulator injection into a highly voided downcomer, 2) when the upper head behaves like a pressurizer, 3) upper plenum injection, and 4) periods of void redistribution.

RESPONSE

See Section 18.2.13.2.

Q492.22 No recommendations are made as to the uncertainties of the pressure or temperature transducers to be used, but the choice appears to be left to the owner or AE. What is the upper limit of uncertainties that should be allowed? Describe the effect of these uncertainties on the measurement of level. What would be the effect on the level measurement should these uncertainties be exceeded?

RESPONSE

See Section 18.2.13.2

Q492.23 Only single RTD sensors on each vertical run are indicated to determine the temperatures of the impulse lines. Where are they to be located? What are the expected temperature gradients along each line under normal operating conditions and under a design basis accident? What is the worst case error that could result from only determining the temperature at a single point on each line?

RESPONSE

See Section 18.2.13.2.

Q492.24 What is the source of the tables or relationships used to calculate density corrections for the level system?

RESPONSE

See Section 18.2.13.2.

Q492.25 The microprocessor system is stated to display the status of the sensor input. Describe how this is indicated and what this actually means with respect to the status of the sensor itself and the reliability of the indication.

RESPONSE

See Section 18.2.13.2.

Q492.26 Describe the provisions for preventing the draining of either the upper head or hot leg impulse lines during an accident. What would be the resultant errors in the level indications should such draining occur?

RESPONSE

See Section 18.2.13.2.

Q492.27 Discuss the effect on the level measurement of the release of dissolved, noncondensible gases in the impulse lines in the event of a depressurization.

RESPONSE

See Section 18.2.13.2.

Q492.28 In some tests at Semi-scale, voiding was observed in the core while the upper head was still filled with water. Discuss the possibility of cooling the core-exit thermocouples by water draining down out of the upper head during or after core voiding with a solid upper head.

RESPONSE

See Section 18.2.13.2.

Q492.29 Describe the behavior of the level measurement system when the upper head is full, but the lower vessel is not.

RESPONSE

See Section 18.2.13.2.

Q492.30 One discussion of the microprocessor system states that water in the upper head is not reflected in the plot. Does this mean that there is no water in the upper head or that the system is indifferent to water in the upper head under these conditions?

RESPONSE

See Section 18.2.13.2.

Q492.31 Describe the details of the pump flow/Dp calculation. Discuss the possible errors.

RESPONSE

See Section 18.2.13.2.

Q492.32 Have tests been run with voids in the vessel? Describe the results of these tests.

RESPONSE

See Section 18.2.13.2.

Q492.33 Estimate the expected accuracy of the system after an ICC event.

RESPONSE

See Section 18.2.13.2.

Q492.34 Describe how the conversion of RTD resistance to temperature is made in the analog level system.

RESPONSE

See Section 18.2.13.2.

Q640.0 PROCEDURES AND TEST REVIEW BRANCH

Q640.1 (14.2.7) Certain exceptions to regulatory guides as listed in Appendix 3A are not acceptable or require further justification.

Provide the following information:

1) Regulatory Guide 1.68

Describe existing tests that verify acceptable plant response for a loss of turbine-generator coincident with a loss of offsite power, or delete this exception and include the appropriate test description.

RESPONSE

See Sections 14.2.12.1.74, 14.2.12.1.75, 14.2.12.3.36 and 14.2.12.3.39. The ability of the plant to respond to a loss of offsite power is demonstrated. Additional testing is performed on the main generation system to verify the operability and controls of the system. The combination of this testing provides more information than could be obtained by performing the required test.

2) Regulatory Guide 1.80

State which tests demonstrate that safety-related valves fail-safe on loss-of-instrument air.

RESPONSE

The failure position of safety-related valves is verified within the test procedure associated with the system to which the valve belongs. Also see Section 14.2.12.1.90.

3) Regulatory Guide 1.118

The discussion states that nuclear instrumentation sensors are exempt from time response testing since their worst case response time is not a significant portion of the total overall system response (i.e., less than 5%). Given that this exemption is no longer permitted by IEEE-338 (1977 version), delete this exception or provide expanded technical justification for not conducting time response testing.

RESPONSE

See Section 7.1.2.6.2 and Appendix 3A.

640.0WC PROCEDURES AND TEST REVIEW BRANCH

Q640.1WC (14.2.2.4) Subsection 14.2.2.4.4 states that GE will be responsible for providing personnel experienced in the startup and operation of the turbine generator and related auxiliary equipment. Expand Subsection 14.2.2.4.4 to explain in greater detail what direct support GE will provide (ex., supply and install turbine-generator, instruct KG&E personnel in the conduct of testing and operation, recommend procedures for starting, operating, and shutting down equipment).

RESPONSE

See Section 14.2.2.4.4.

Q640.2 (14.2.10.2) Your initial criticality description should be expanded to include:

1) A source range count of at least 1/2 count per second should be visible on the startup channels prior to commencing the startup.

2) The signal to noise ratio should be known to be greater than 2.

3) Criticality predictions for boron concentration and control rod positions should be provided, and criteria and actions to be taken should be established if actual plant conditions deviate from predicted values.

4) The approach to criticality should be slow enough to limit start up rate at criticality to less than 1 decade per minute.

RESPONSE

1) The procedure requires greater than 1/2 counts per second.

2) See Section 14.2.12.3.9.

3) See Section 14.2.10.2.

4) Reactivity insertion rates on approach to criticality are so low that startup rate at criticality is not a function of the rate of approach to criticality. Sufficient precautions are included in the startup test procedures to preclude exceeding a 1-decade-per-minute startup rate at criticality.

Q640.2WC (14.2.2.6) Subsection 14.2.2.6 refers to Section 13 regarding the qualifications of key personnel involved in the initial testing program. Subsection 13.1.3.1 references Regulatory Guide 1.8. Our current position is that the individuals involved in preoperational or startup testing should hold the qualifications stated in Regulatory Position 3 of proposed Revision 2 to Regulatory Guide 1.8, February 1979 (issued for comment). State that your minimum qualification requirements will be in accordance with this regulatory position or provide justification for requiring any lesser qualifications.

RESPONSE

This area of review was covered by the NRC Management Structure and Technical Resources Review Team during the week of 1/18/82 at KG&E.

Q640.3 (14.2.11) Section 14.2.11 of SNUPPS states that insofar as practicable, test requirements will be completed prior to exceeding 25-percent power for all plant structures, systems and components that are relied upon to prevent, limit or mitigate the consequences of postulated accidents. According to Table 14.2-5 the following startup tests are performed after exceeding 25-percent power:

1) S070012 - Rod Drop and Plant Trip

2) S07AB01 - Automatic Steam Generator Level Control

3) S07SF05 - Automatic Reactor Control System

4) S07SF07 - Startup Adjustments of Reactor Control System

Perform these tests at 25% power or less, or provide technical justification for not fulfilling the testing requirements of Section 14.2.11.

RESPONSE

See Table 14.2-5.

Q640.3WC (14.2.5) Section 14.2.5 states that during Power Ascension Testing, review and approval of initial startup test procedure results is completed for each of the plateaus. The first plateau is at 30%. In Section 14.2.11 of SNUPPS, a 25% power level is referenced. This is given as the power level which will not be exceeded until major plant test requirements are completed satisfactorily. Modify Section 14.2.5 to clarify how the applicable startup test results will be reviewed prior to exceeding 25% power as referenced in Section 14.2.11 of SNUPPS.

RESPONSE

See Section 14.2.11.

Q640.4 (14.2.11) Section 14.2.11 of SNUPPS states that startup test procedures will be available for NRC review at least 60 days prior to fuel loading. Table 14.2-5 indicates that twenty of thirty-eight startup tests will be in the procedure preparation, review and approval stage at that time. Modify Table 14.2-5 to indicate by a note or legend alteration that complete procedures will be available for review in the time frame stated in Section 14.2.11.

RESPONSE

See Table 14.2-5.

Q640.4WC (14.2.7) Appendix 3A states in the Section on Regulatory Guide 1.58 that an alternative method for qualifying nuclear power plant inspection, examination and testing personnel will be used. Insufficient detail is available to determine whether or not the alternative qualification program provides the same quality training. Expand the description of the alternative qualification method in Appendix 3A or delete this exception to Regulatory Guide 1.58. Note: Regulatory Positions C.5, 6, 7, 8 and 10 of Regulatory Guide 1.58 (Rev. 1, 9/80) apply to the Wolf Creek nuclear station.

RESPONSE

This area of review was covered by the NRC Management Structure and Technical Resources Review Team during the week of 1/18/82 at KG&E.

Q640.5 (14.2.12) Provide a commitment to include in your test program the design features to prevent or mitigate anticipated transients without scram (ATWS) that may now, or in the future, be incorporated into your plant design (Subsection 15.8).

RESPONSE

See Section 15.8.

Q640.5WC (14.2.8.2) Subsection 14.2.8.2 of SNUPPS refers to Section 14.2.8 of the Site Addendum for additional site specific information. SNUPPS-WC contains no such information. Provide the following:

1) Specify which individual at Wolf Creek will be responsible for incorporating reactor operating and testing experiences of similar power plants during the Initial Test Program.

2) Subsection 14.2.8.1 of SNUPPS only references development of preoperational test procedures. Provide information on how information or other plant's experiences will be used in the preparation of Phase II-IV testing.

RESPONSE

See Section 14.2.8.

Q640.6 (14.2.12) List those tests that will only be performed on the first SNUPPS unit. In addition cite the criteria that will be used during subsequent unit testing programs to ensure that follow-on units perform in an identical manner regarding those tests to be deleted.

RESPONSE

See Section 14.2.8.

Q640.6WC (14.2.12) Certain terminology used in the individual test descriptions does not clearly indicate the source of the acceptance criteria to be used in determining test adequacy. An acceptable format for providing acceptance criteria for test results includes any of the following:

- o Referencing technical specifications
- o Referencing specific sections of the FSAR
- o Referencing vendor technical manuals
- o Providing specific quantitative bounds (only if the information cannot be provided in any of the above ways).

Modify the individual test description subsection presented below or, if applicable, add a paragraph to Subsection 14.2.12 that provides an acceptable description of each of the nuclear terms.

1) Within design specification
   14.2.12.1.1
   1.2
   1.3
   2.1
   2.2
   2.3

2) In accordance with design
   14.2.12.1.1

3) Responds properly
   14.2.12.1.2
   2.1
   2.2
   2.3

RESPONSE

See response to Question 640.10 which provides a description of the terminology used.

Q640.7 (14.2.12.3) Identify any of the post-fuel loading tests described in Section 14.2.12.3. which are not essential towards the demonstration of conformance with design requirements for structures, systems, components, and design features that meet any of the following criteria:

1) Will be relied upon for safe shutdown and cooldown of the reactor under normal plant

conditions and for maintaining the reactor in a safe condition for an extended shutdown period.

2) Will be relied upon for safe shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions, and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions.

3) Will be relied upon for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications.

4) Are classified as engineered safety features or will be relied upon to support or assure the operation of engineered safety features within design limits.

5) Are assumed to function or for which credit is taken in the accident analysis for the facility (as described in the Final Safety Analysis Report).

6) Will be utilized to process, store, control, or limit the release of radioactive materials.

RESPONSE

All post-fuel loading tests essential to demonstrate conformance with design requirements for structures, systems, components, and design features for the criteria specified in Question 640.7, items (1) through (6) are included in Section 14.2.12.3.

Q640.7WC (14.2.12) Verify that the ultimate heat sink cooling pond (Subsection 9.2.5) is tested to demonstrate adequate NPSH and the absence of vortexing over range of basin level from maximum to the minimum calculated 30 days following LOCA.

RESPONSE

See Sections 9.2.1.2.2.2 and 14.2.12.1.2.

Q640.8 (14.2.12.3) The objectives specified for several tests are in-appropriate. In general, appropriate test objectives are:

- o to measure
- o to calibrate
- o to obtain data
- o to document
- o to verify performance

Provide appropriate objectives for the following tests:

14.2.12.3.1
3.2
3.3
3.8
3.22
3.33
3.35

RESPONSE

See Sections 14.2.12.3.1, 14.2.12.3.2, 14.2.12.3.3, 14.2.12.3.8, 14.2.12.3.22, and 14.2.12.3.33. Section 14.2.12.3.35 has been deleted.

Q640.8WC (14.2) Table 14.2-1 (Sheet 4) of SNUPPS states that for S-X3GD01, S-X3EF01, and S-X3NG01 the X in the test numbers will be a U or a K, depending on the test site. In SNUPPS-WC, Section 14.2.12, the tests are listed as S-13GD01, S-3EF01, and S-3NG01. Modify Section 14.2.12 of SNUPPS-WC or Table 14.2-1 of SNUPPS to eliminate this discrepancy (the test numbers on the non-safety related tests should also be corrected).

RESPONSE

See the test abstracts in Section 14.2.12. The test abstracts, as identified in Section 14.2.12, are numbered per the method used at WCGS.

Q640.9 (14.2.12.3) It is unacceptable to reference test instructions for test prerequisites. Provide acceptable prerequisites for the following tests:

14.2.12.3.1
3.4
3.5
3.6
3.7
3.8.2.a

3.13
3.14
3.21
3.22
3.23
3.24
3.25.2.a
3.26
3.27
3.29
3.30
3.31
3.32
3.33
3.34.2.b
3.35

RESPONSE

See Sections 14.2.12.3.1, 14.2.12.3.4, 14.2.12.3.5, 14.2.12.3.6, 14.2.12.3.7, 14.2.12.3.8, 14.2.12.3.13, 14.2.12.3.14, 14.2.12.3.21, 14.2.12.3.22, 14.2.12.3.23, 14.2.12.3.24, 14.2.12.3.25, 14.2.12.3.26, 14.2.12.3.27, 14.2.12.3.29, 14.2.12.3.30, 14.2.12.3.31, 14.2.12.3.32, 14.2.12.3.33, and 14.2.12.3.34. Section 14.2.12.3.35 has been deleted.

Q640.10 (14.2.12) Certain terminology used in the individual test descriptions does not clearly indicate the source of the acceptance criteria to be used in determining test adequacy. An acceptable format for providing acceptance criteria for test results includes any of the following:

- o Referencing technical specifications
- o Referencing specific sections of the FSAR
- o Referencing vendor technical manuals
- o Providing specific quantitative bounds (only if the information cannot be provided in any of the above ways).

Modify the individual test description subsection presented below or, if applicable, add a paragraph to Subsection 14.2.12 that provides an acceptable description of each of the unclear terms.

1) Within design specifications

14.2.12.1.3
1.4
1.5
1.7
1.9
1.10
1.11
1.12
1.15 (2 times)
1.18 (2 times)
1.21 (2 times)
1.23 (2 times)
1.24
1.25 (2 times)
1.26 (2 times)
1.27
1.28 (3 times)
1.29 (3 times)
1.30
1.32 (2 times)
1.33 (4 times)
1.34 (3 times)
1.36
1.37 (3 times)
1.39
1.41 (3 times)
1.42 (2 times)
1.43
1.44 (2 times)
1.45 (2 times)
1.46
1.47
1.48
1.49
1.50 (2 times)
1.51 (2 times)
1.52
1.53
1.59
1.60 (2 times)
1.61 (2 times)
1.62
1.64 (6 times)
1.65
1.66 (2 times)
1.68 (2 times)
1.71

1.72
2.1
2.2 (2 times)
2.3 (2 times)
2.4
2.5
2.6 (2 times)
2.7
2.8
2.10
2.11 (2 times)
2.14 (2 times)
2.15
2.16
2.19
2.22 (2 times)
2.25
3.15
3.18 (2 times)
3.20 (2 times)

<u>RESPONSE</u>

The acceptance criteria provided in the individual test descriptions meet the requirements of Regulatory Guide 1.70, Revision 3, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. It is not the intent of the test descriptions to provide a source of the acceptance criteria or specific quantitative values to be utilized to determine test adequacy. The acceptance criteria provided is a summary of the acceptance criteria provided in the individual test procedures, which contain the specific criteria against which success or failure of the test procedure is judged.

2) In accordance with design, in accordance with system design

14.2.12.1.1 (2 times)
1.6 (2 times)
1.8
1.44
1.45
1.46
1.48
1.51
1.54
1.55
1.56 (2 times)
1.57

1.58 (2 times)
1.59
1.63 (2 times)
1.64 (4 times)
1.65 (2 times)
1.66
1.68
1.69
1.70
1.71 (2 times)
1.72
1.73
2.15
2.16

RESPONSE

See the response to item (1).

3) In accordance with design specification, in accordance with system design specification

14.2.12.1.39
2.1
2.9
2.11
2.12
2.13
2.20
2.21 (2 times)
2.24
2.26

RESPONSE

See the response to item (1).

4) Design
14.2.12.1.10
1.11
1.17
1.35
1.42
1.65 (3 times)
1.67 (5 times)
1.70
1.80
2.17
2.18

3.15
3.17
3.37

RESPONSE

See the response to item (1).

5) Within design limits, without exceeding design limits, within the limits predicted by design analyses, within design requirements

14.2.12.1.16 (2 times)
1.29
1.32
1.35
1.37
1.41
1.62
1.64
1.73
1.78
1.79
3.16

RESPONSE

See the response to item (1).

6) Within allowable limits, within required limits

14.2.12.1.22
1.38
1.62

RESPONSE

See Sections 14.2.12.1.22, 14.2.12.1.38, and 14.2.12.1.62.

7) Required

14.2.12.1.10
1.22
1.64 (10 times)
1.65 (2 times)
1.85

RESPONSE

See Sections 14.2.12.1.10, 14.2.12.1.22, 14.2.12.1.64, 14.2.12.1.65, and 14.2.12.1.85.

8) Rated

14.2.12.1.62
1.64 (2 times)
1.65
1.82 (3 times)

RESPONSE

See Sections 14.2.12.1.62, 14.2.12.1.64, 14.2.12.1.65, and 14.2.12.1.82.

9) Responds, responds properly, properly respond

14.2.12.1.12
1.34
1.36
1.48
1.49
1.51

RESPONSE

See Sections 14.2.12.1.12, 14.2.12.1.34, 14.2.12.1.36, 14.2.12.1.48, 14.2.12.1.49, and 14.2.12.1.51.

10) In accordance with test instructions, is provided in test instructions, meets the requirements of the test instructions, consistent with the acceptance criteria given in the test procedure, agrees with the acceptance criteria given in the test procedure, as required by the test instructions

14.2.12.1.74
1.75
1.76
3.2
3.6
3.7
3.8
3.11
3.13

3.14
3.23
3.30
3.31
3.32
3.33

RESPONSE

See Sections 14.2.12.1.74, 14.2.12.3.2, 14.2.12.3.6, 14.2.12.3.7, 14.2.12.3.8, 14.2.12.3.11, 14.2.12.3.13, 14.2.12.3.14, 14.2.12.3.23, 14.2.12.3.30, 14.2.12.3.31, 14.2.12.3.32, and 14.2.12.3.33.

11) Shall not exceed code-allowable stresses, must not exceed their code-allowable limits at the test or design conditions

14.2.12.1.80
1.81
3.37 (2 times)

RESPONSE

The phrases "code-allowable stresses" and "code-allowable limits" are specific and consistent with the requirements in FSAR Section 3.0. This is a design verification program and specifying the codes as acceptance criteria is consistent with the design criteria.

12) Set point tolerances

14.2.12.1.2

RESPONSE

The phrase "set point tolerances" is referring to the lift point (set point) and band (tolerances) at which the main steam safety valves lift. Specific values are provided in the Test Procedure S-03AB02.

13) Acceptable

14.2.12.1.14
1.64 (2 times)
2.17
2.18

RESPONSE

See Sections 14.2.12.1.14, 14.2.12.1.64, 14.2.12.2.17, and 14.2.12.2.18.

14) Adequate

14.2.12.1.37
1.83

RESPONSE

See Sections 14.2.12.1.37 and 14.2.12.1.83.

15) Approximate

14.2.12.1.14
1.80
3.37

RESPONSE

See Sections 14.2.12.1.14, 14.2.12.1.80, and 14.2.12.3.37.

16) Predicted

14.2.12.1.14

RESPONSE

See Section 14.2.12.1.14.

17) Verified

14.2.12.1.14
1.22

RESPONSE

See Sections 14.2.12.1.14 and 14.2.12.1.22.

18) Fails safe

14.2.12.1.73

RESPONSE

See Section 14.2.12.1.73.

19) Operate satisfactorily per design

14.2.12.1.83

RESPONSE

See Section 14.2.12.1.83.

20) Impair design functions

14.2.12.1.83

RESPONSE

See Section 14.2.12.1.83.

21) Slightly above

14.2.12.1.20

RESPONSE

See Section 14.2.12.3.19 and 14.2.12.3.20.

Q640.11 (14.2.12) Our review of your initial test program description disclosed that the operability of several of the systems and components listed in Regulatory Guide 1.68 (Rev. 2), Appendix A, may not be demonstrated. Expand your FSAR to include appropriate test descriptions (or identify existing descriptions) that address the following items from Appendix A, or provide technical justification for any exceptions to the guide in Subsection 14.2.7:

1) Preoperational Testing

1.a.(2)(i) RCS safety valves

RESPONSE

Component testing is not within the scope of the Preoperational Test Program; therefore, no test abstracts are provided. See Section 3.9 (N) 3.2.1, Pump and Valve Operability Program.

1.b.(1) Control rod drive system test

RESPONSE

This test cannot adequately be performed prior to core loading. The system is tested prior to operation as described in Section 14.2.12.3.25.

1.e.(5) Steam extraction system

RESPONSE

The operability of the steam extraction system is verified in the Plant Performance Test, S-090007. See Section 14.2.12.2.27.

1.e.(6) Turbine stop, control, and intercept valves

RESPONSE

See new test descriptions, Section 14.2.12.2.28, Turbine Trip Test (S-04AC02), and Section 14.2.12.2.29, Turbine System Cold Test (S-04AC03).

1.e.(10) Feedwater heater and drain systems

RESPONSE

See new test description, Section 14.2.12.2.33, Secondary Vent and Drain System Preoperational Test Procedure S-04AF01.

1.h Test of protective devices such as leaktight covers, structures, or housings provided to protect Engineered Safety Features from flooding

RESPONSE

The equipment location of safety-related equipment is such that no credit is taken for the above-mentioned protective devices except that credit is taken for watertight doors. These doors are verified in the penetration closure program.

1.h.(8) Tanks and other sources of water used for ECCS

RESPONSE

The operability of the control circuits associated with the refueling water storage tank and condensate storage tank are verified in the Residual Heat Removal (RHR) System Cold Preoperational Test Procedure S-03EJ01, and the Condensate System Pre-operational Test Procedure S-04AD01, respectively. See Sections 14.2.12.1.34 and 14.2.12.2.1.

The instrumentation associated with the containment sumps is tested in S-03EJ01. See Section 14.2.12.1.34.

1.i.(5) Containment airlock leak rate test

RESPONSE

The containment air lock is leak tested in the Local Containment Leak Rate Test Procedure S-030002. See Section 14.2.12.1.78.

1.i.(12) Containment air purification and cleanup system

RESPONSE

See Section 14.2.12.1.51.

1.i.(15) Containment penetration pressurization system tests

RESPONSE

WCGS does not have a containment penetration pressurization system.

1.j.(6) Loose parts monitoring system

RESPONSE

See Section 14.2.12.2.36.

1.j.(7) Leak detection system for ECCS and containment spray system outside of containment

RESPONSE

See Section 14.2.12.2.32.

1.j.(8) Reactor control system

RESPONSE

Instrument alignment and calibration is performed during the component test program. Sections 14.2.12.3.25, 14.2.12.3.26, and 14.2.12.3.29 demonstrate the capability of the reactor control system during power ascension testing.

1.j.(9) Pressure control systems designed to prevent leakage across boundaries

RESPONSE

WCGS does not have a pressure control system to prevent leakage across boundaries.

1.j.(11) Traversing incore probe system

RESPONSE

This test cannot adequately be performed prior to core loading. The system is tested prior to operation as described in Section 14.2.12.3.39.

1.j.(13) Incore nuclear instrumentation

RESPONSE

This test cannot adequately be performed prior to core loading. The system is tested prior to operation as described in Section 14.2.12.3.39.

1.j.(14) Instrumentation and controls that affect transfers of water supplies to auxiliary feedwater pumps, ECCS pumps, and containment spray pumps

RESPONSE

See Sections 14.2.12.1.7., 14.2.12.1.28, 14.2.12.1.34, and 14.2.12.1.41.

1.j.(16) Hotwell level control system

RESPONSE

Procedure S-04AD01, Condensate System Preoperational Test, verifies the operability of the hotwell level control system. See Section 14.2.12.2.1.

1.j.(17) Feedwater heater temperature, level, and bypass control systems

RESPONSE

See Section 14.2.12.2.33.

1.j.(18) Auxiliary startup instrument test

RESPONSE

See Section 14.2.12.3.21.

1.j.(20) Instrumentation used to detect internal and external flooding

RESPONSE

See Sections 14.2.12.2.31 and 14.2.12.2.32 for the instrumentation used to detect internal flooding. The WCGS design does not provide instrumentation for the detection of external flooding as all sites are "dry sites."

1.j.(22) Instrumentation that can be used to track the course of postulated accidents such as containment sump level monitors and humidity monitors

RESPONSE

The operability of instrumentation utilized to track the course of postulated accidents is verified in the test procedures associated with the system in which the instrument belongs.

1.j.(24) Annunciators for reactor control and engineered safety features

RESPONSE

See Sections 14.2.12.1.71, 14.2.12.1.72, and 14.2.12.1.73, respectively.

In addition to the above integrated annunciator testing, the annunciator points associated with various reactor functions and ESF components are also tested in the individual system preoperational test procedures.

1.j.(25) Process computers

RESPONSE

The computer was tested and software verified prior to startup testing. During the startup program, verification of these calculations performed by the computer to ensure the plant is operating within technical specification limits were performed and results compared to hand calculations, installed instrumentation, or other analytical programs.

1.l.(4) Isolation features for steam generator blowdown

RESPONSE

See Section 14.2.12.1.72.

1.l.(7) Isolation features for liquid radwaste effluent systems

RESPONSE

See Section 14.2.12.2.6.

1.m.(4) Dynamic and static load testing of cranes, hoists, and associated lifting and rigging equipment, including the fuel cask handling crane. Static testing at 125% of rated load and full operational testing at 100% of rated load

RESPONSE

Static testing at 125% of rated loads and crane bridge, trolley, and hoist speeds at rated loads is addressed in revised Sections 14.2.12.1.54, 14.2.12.1.56, and 14.2.12.1.58. Operability of the fuel handling system, using a dummy fuel assembly, is addressed in Section 14.2.12.1.56.

1.n.(2) Closed loop cooling water systems

RESPONSE

See Section 14.2.12.2.34.

1.n.(6) Chemistry control systems for the reactor coolant and secondary coolant systems

RESPONSE

See Sections 14.2.12.1.27, 14.2.12.1.28, 14.2.12.1.29, and 14.2.12.2.30.

1.n.(9) Vent and drain systems for contaminated or potentially contaminated systems

RESPONSE

See Section 14.2.12.2.32.

1.n.(10) Purification and cleanup systems for the reactor coolant system

RESPONSE

See Sections 14.2.12.1.27, 14.2.12.1.28, and 14.2.12.1.29.

1.n.(12) Boron recovery system

RESPONSE

See Section 14.2.12.1.27 and Section 14.2.12.1.29.

1.n.(14)(c) Battery room ventilation

RESPONSE

Proper ventilation to battery rooms 1 through 4 is supplied by the control building HVAC system, and is verified in Procedure S-03GK01. See Section 14.2.12.1.45.

1.n.(16) Cooling and heating systems for the refueling water storage tank

RESPONSE

There is no cooling system associated with the refueling water storage tank. A source of heat, which is non-safety related, is supplied from the auxiliary steam system and is controlled by a temperature control valve, which is operationally tested in Procedure S-03EC01.

1.o Reactor components handling systems

RESPONSE

See Sections 14.2.12.1.54 through 14.2.12.1.59. The non-permanently installed fuel handling equipment is periodically inspected and verified operational prior to fuel handling evolutions.

2) Initial Fuel Load and Precritical Testing

2.a Shutdown margin verification for the fully loaded core

RESPONSE

The verification of shutdown margin for a fully loaded core is provided by controlling the boron concentration. See revised Section 14.2.12.3.1.

2.b Control rod withdrawal and insertion speeds, sequencers and protective interlocks

RESPONSE

See Sections 14.2.12.1.73, 14.2.12.3.26 and 14.2.12.3.29.

2.d Final reactor coolant system leak rate test

RESPONSE

Determination of the reactor coolant system leak rate is not conducted as a startup test, but is verified on a frequent and routine basis in accordance with the technical specifications, and will be verified prior to initial criticality.

4) Low Power Testing

4.b Confirm by analysis that rod insertion limits will be adequate to ensure a shutdown margin consistent with accident analysis assumptions, with the greatest worth control rod stuck out of the core.

RESPONSE

Verification of rod worth is accomplished by procedure S-07SF08, RCCA or Bank Worth Measurement at Zero Power, Section 14.2.12.3.32. When the results of this test meet the acceptance criteria, shutdown margin is assured by operation within the insertion limits.

4.c Pseudo-rod-ejection test

RESPONSE

See Sections 14.2.12.3.32, 14.2.12.3.33, and 14.2.12.3.38.

4.e Flux distribution determination

RESPONSE

See Section 14.2.12.3.38.

4.f Neutron and gamma radiation surveys

RESPONSE

See Section 14.2.12.3.40.

4.g Determination of proper response of process and effluent radiation monitors

RESPONSE

The operability of the radiation monitors is demonstrated during the Preoperational Test Program. See Sections 14.2.12.1.86, 14.2.12.2.23, and 14.2.12.2.26.

4.h Chemical and radiochemistry tests

RESPONSE

The operability of the primary and secondary sampling systems is verified during the Preoperational Test Program. See Sections

14.2.12.2.22 and 14.2.12.2.26. In addition, chemistry is maintained within technical specification limits during the startup program, using plant procedures.

4.i Demonstration of the operability of control rod withdrawal inhibit or block functions over the reactor power level range during which such features must be operable

<u>RESPONSE</u>

See the response to question 640.11(2).2.b.

4.j Demonstration of the capability of the primary containment ventilation system.

<u>RESPONSE</u>

See Section 14.2.12.2.27.

4.n Demonstration of the operability of the control room computer system

<u>RESPONSE</u>

See the response to question 640.11(1)1.j.(25).

4.r Demonstration of the operability of reactor coolant system purification and cleanup systems

<u>RESPONSE</u>

Preoperational testing of the chemical and volume control system (CVCS) is addressed in Sections 14.2.12.1.24 through 14.2.12.1.29. The ability of the CVCS to control boron concentration is demonstrated throughout the startup program. In addition, the chemistry limits for continued operation during the startup program are maintained within those limits provided in the technical specifications. No additional testing is required.

4.t Performance of natural circulation tests of the reactor coolant system to determine that adequate heat removal capability exists. NUREG-0694 "TMI Related Requirements for New Operating Licenses," Item I.G.1, requires applicants to perform "a special low power testing program approved by NRC to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training." To comply with this requirement new PWR applicants have committed to a series of natural circulation tests. To date such tests have been performed at the Sequoyah 1, North Anna 2, and Salem 2 facilities. Based on the success of the programs at these plants, the staff has concluded that augmented natural circulation training should be performed for all future PWR operating licenses. Includes descriptions of natural circulation tests that, in addition to validating the operating procedures, fulfill the following objectives:

Testing

The tests should demonstrate the following plant characteristics: length of time required to stabilize natural circulation, core flow distribution, ability to establish and maintain natural circulation with or without onsite and offsite power, the ability to uniformly borate and cool down to hot shutdown conditions using natural circulation, and subcooling monitor performance.

Training

Each licensed reactor operator (RO or SRO who performs RO or SRO duties, respectively) should participate in the initiation, maintenance and recovery from natural circulation mode. Operators should be able to recognize when natural circulation has stabilized, and should be able to control saturation margin, RCS pressure, and heat removal rate without exceeding specified operating limits.

If these tests have been performed at a comparable prototype plant, they need to be repeated only to the extent necessary to accomplish the above training objectives.

RESPONSE

See Chapter 18, item I.G.1. A test description for the natural circulation test is provided in Section 14.2.12.3.41.

5) Power-Ascension Tests

5.b Determine that steady-state core performance is in accordance with design

RESPONSE

See Section 14.2.12.3.38.

5.d Demonstrate the capabilities of plant features and procedures for controlling core xenon transients

RESPONSE

Xenon oscillation tests have been performed on other Westinghouse four-loop plants, and results have been documented and approved by the NRC. The procedures associated with the control of xenon transients utilize similar methods as those utilized for the reference plant.

5.e Pseudo-rod-ejection test

RESPONSE

See the response to question 640.11(4).4.c.

5.f Single rod insertion and withdrawal

RESPONSE

This test is scheduled at 50-percent power. See Section 14.2.12.3.33 and 14.2.12.3.38.

5.g Demonstrate operation of the control rod sequencers, and rod withdrawal block functions

RESPONSE

See the response to question 640.11.(2).2.b.

5.h Check rod scram times from data recorded during the startup test phase

RESPONSE

Not applicable (BWR only).

5.i Demonstrate the capability of incore and excore neutron flux instrumentation to detect a control rod misalignment equal to or less than the technical specification limits

RESPONSE

It is not a design requirement of the excore neutron detectors to be capable of detecting a control rod misalignment equal to or less than technical specifications limits. The WCGS design relies on the rod position indication system to provide indication of rod misalignment, with the incore neutron flux instrumentation being available to further investigate the misalignment.

The design analysis allows a rod misalignment of 15 inches. The technical specifications require that the rods be within 7-1/2 inches of the demanded position. This requirement, along with the accuracy of the rod position indication system, which is less than 7-1/2 inches, ensures that the maximum misalignment could be no greater than 15 inches. The rod position indication system will detect this misalignment and is tested in Procedure S-07SF04. See Section 14.2.12.3.28.

In addition, during the RCCA or Bank Worth Measurement at Power Test, Procedure S-07SF09, measurements are made with incore detectors at incremental rod insertion levels to acquaint operating personnel with methods of detection of misaligned rods, but the misalignment is generally greater than that allowed by the technical specifications.

5.l Demonstrate design capability of all systems and components provided to remove residual or decay heat from the reactor coolant system

RESPONSE

See Sections 14.2.12.1.1, 14.2.12.1.7, 14.2.12.1.8, 14.2.12.1.13, 14.2.12.1.35, and 14.2.12.3.14.

5.m Demonstrate that reverse flows through idle loops and differential pressures across the core are in agreement with design values

RESPONSE

Not applicable. The WCGS design requires the operation of all four reactor coolant pumps at power.

5.n Obtain baseline data for reactor coolant system loose parts monitoring system

RESPONSE

See the response to question 640.11.(1).1.j.(6).

5.r Verification of input to, and output from control room process computer

RESPONSE

See the response to question 640.11.(1).1.j.(25).

5.s Verify the performance of the auxiliary feedwater control system, the hotwell level control system, steam pressure control system, and the reactor coolant makeup and letdown control systems

RESPONSE

See Sections 14.2.12.1.7, 14.2.12.1.29, 14.2.12.2.1, 14.2.12.3.11, 14.2.12.3.14, 14.2.12.1.8, 14.2.12.1.27, and 14.2.12.2.27.

5.t Verify the response times, relieving capacities, and reset pressures for the pressurizer relief valves; main steam line safety valves; atmospheric relief valves; and the turbine bypass valves

RESPONSE

See the response to question 640.13 and Sections 14.2.12.1.1, 14.2.12.1.2, 14.2.12.1.4, 14.2.12.1.12, and 14.2.12.1.21.

5.u Verify operability and response times of main steam line isolation and branch steam line isolation valves

RESPONSE

See Section 14.2.12.1.4.

5.v Verification of main steam system and feedwater system performance

RESPONSE

See Sections 14.2.12.1.7, 14.2.12.1.8, 14.2.12.2.27, 14.2.12.3.11, 14.2.12.3.13, 14.2.12.3.14, 14.2.12.3.37, 14.2.12.1.87 and 14.2.12.1.88.

5.w Demonstrate that concrete temperatures surrounding hot penetrations do not exceed design limits.

RESPONSE

Concrete temperatures surrounding hot penetrations are monitored during the Plant Performance Test S-090007.

5.y Verify the proper operation of the incore nuclear instrumentation and instruments and systems used to perform a heat balance

RESPONSE

See Sections 14.2.12.3.22, 14.2.12.3.24, and 14.2.12.3.39.

5.z Demonstrate that process and effluent radiation monitoring systems are responding correctly

RESPONSE

See the response to question 640.11.(4).4.g.

5.aa Demonstrate the operation of the chemical and radiochemical control systems

RESPONSE

See the response to question 640.11.(4).4.h.

5.bb Conduct neutron and gamma radiation surveys to establish the adequacy of shielding

RESPONSE

See the response to question 640.11.(4).4.f.

5.cc Demonstrate the operation of the gaseous and liquid radioactive waste processing, storage, and release systems

RESPONSE

Preoperational testing of the gaseous and liquid radwaste systems is addressed in Sections 14.2.12.1.52, 14.2.12.2.6, and 14.2.12.2.7. These systems are in operation during power ascension to support plant operation.

5.ff Demonstrate that ventilation systems maintain design temperatures

RESPONSE

See Section 14.2.12.2.27.

5.ii Demonstrate that the dynamic response of the plant is in accordance with design for limiting reactor coolant pump trips

RESPONSE

See the response to Regulatory Guide 1.68, Revision 2, in Appendix 3A.

5.kk Demonstrate that the dynamic response of the plant is in accordance with design for the loss of or bypassing of the feedwater heaters

RESPONSE

See the response to Regulatory Guide 1.68, Revision 2, in Appendix 3A and Section 14.2.12.3.42.

5.mm Demonstrate that the dynamic response of the plant is in accordance with design for the case of automatic closure of all main steam line isolation valves at 100 percent reactor power

RESPONSE

See the response to Regulatory Guide 1.68, Revision 2, in Appendix 3A.

5.nn Demonstrate that the dynamic response of the plant is in accordance with design for the case of full load rejection (tripping of the main generator breakers)

RESPONSE

The plant trip from 100-percent power will be initiated by opening the main generator output breakers. See Section 14.2.12.3.11.

Q640.12 (14.2.12) We could not conclude from our review of your individual test descriptions that comprehensive testing is scheduled for several systems and components. Therefore, clarify or expand the appropriate test descriptions to address the following items:

1) 14.2.12.1.1 - Clarify, or reference the FSAR section which clarifies, the purpose of a decreasing condenser pressure signal.

RESPONSE

See Section 14.2.12.1.1.

2) 14.2.12.1.5 - Provide acceptance criteria for steam generator feedwater pump operation.

RESPONSE

The main feedwater system preoperational test, S-03AE01, performs the initial operation of the steam generator feedwater pumps, using auxiliary steam. The final acceptance of the steam generator feedwater pumps is demonstrated during the Automatic Steam Generator Level Control Test Procedure S-07AB01. See revised Section 14.2.12.3.13.

3) 14.2.12.1.7 - Subsection 10.4.9.2.3 indicates four separate actuation signals can cause an automatic start of the motor-driven auxiliary feed pump. Ensure these four are included in your test description acceptance criteria.

RESPONSE

The Auxiliary Feedwater Motor-Driven Pump and Valve Preoperational Test Procedure S-03AL01 verifies the automatic start of the motor-driven pumps on receipt of an ESFAS signal. The Engineered Safeguards (BOP) Preoperational Test Procedure S-03SA02 verifies the input signals identified above. See Section 14.2.12.1.72.

4) 14.2.12.1.8 - Our review of licensee event reports has disclosed several instances of turbine-driven auxiliary feedwater pump failure to start on demand. It appears that many of these failures could have been avoided if more thorough testing had been conducted during the plant's initial test programs. In order to discover any problems affecting pump startup and to demonstrate the reliability of your emergency cooling system, state your plans to demonstrate at least five consecutive, successful, cold quick pump starts during your initial test program.

RESPONSE

The ability of the turbine-driven auxiliary feedwater pumps to successfully undergo five consecutive cold starts was demonstrated in the Auxiliary Feedwater Turbine-Driven Pump and Valve Preoperational Test Procedure S-03AL02. See Section 14.2.12.1.8.

5) 14.2.12.1.9 - Commit to verifying operation of any pump permissive interlocks which serve to prevent cold water addition accidents or serve to protect RCS components from excessive differential pressures at low temperatures.

RESPONSE

There are no reactor coolant pump permissive interlocks that serve to prevent cold water addition accidents or protect RCS components from excessive differential pressures at low temperatures. The WCGS design does not allow operating at power with less than four reactor coolant pumps in operation.

6) 14.2.12.1.17 and 14.2.12.1.18 - State that flow and coastdown testing will be performed for all permissible combinations of pump operation.

RESPONSE

The Reactor Coolant System Flow Measurement Procedure, S-03BB09, confirms that the reactor coolant flow rate in each loop, without the core installed, is greater than design. See Section 14.2.12.1.17. The Reactor Coolant System Flow Coastdown Test, S-03BB10, determines the rate of change of reactor coolant flow, for the configurations identified in the accident analysis, for a decrease in reactor coolant system flow, Section 15.3. See Section 14.2.12.1.18. It is not the intent of the above procedures to verify all permissible combinations of pump operation.

7) 14.2.12.1.29 - Verify that the maximum obtainable boron dilution rate is less than or equal to that assumed in your accident analysis (Subsection 15.4.6).

RESPONSE

Preoperational tests S-03BG01, S-03BG03, S-03BG04, S-03BG05, S-03BG06, and S-04BL01 demonstrated the performance characteristics of the charging and reactor makeup water pumps in various system configurations. Procedure S-03BG06 also verified that the letdown flowrates from the reactor coolant system are within design specifications. Due to the conservatism provided in the accident analysis, subsection 15.4.6, as related to the given dilution flow for the postulated conditions, and considering the data obtained in the above procedures, no additional testing should be necessary to verify the protection margin to dilution. See Sections 14.2.12.1.24, 14.2.12.1.26, 14.2.12.1.27, 14.2.12.1.28, 14.2.12.1.29, and 14.2.12.2.2.

8) 14.2.12.1.34 - Ensure that the interlocks and isolation valves for overpressure protection of the RHR system are tested (Subsection 5.4.7.2.5).

RESPONSE

The interlocks and isolation valves for over pressure protection of the RHR system were tested in the RHR System Cold Preoperational Test Procedure S-03EJ01.

9) 14.2.12.1.39 - State which safety signals are used to test boron recirculation pump and valve response.

RESPONSE

See Section 14.2.12.1.39.

10) 14.2.12.1.40 - Verify that paths for the air-flow test of containment spray nozzles overlap the water-flow test paths of the pumps to demonstrate that there is no blockage in the flow path.

RESPONSE

The supply path for the air-flow test of the containment spray nozzles, verified in Procedure S-03EN01, and the water discharge path of the containment spray pumps, verified in Procedure S-03EN02, utilize the same test connection, therefore ensuring that no blockage exists in the system flow path.

11) 14.2.12.1.41 - State which safety signals are used to test containment spray pump and valve response.

RESPONSE

See Section 14.2.12.1.41.

12) 14.2.12.1.48 - Verify that the cooling fans can operate in accordance with design requirements at the containment design peak accident pressure.

RESPONSE

The ability of the containment cooling fans to operate at the containment design peak accident pressure was verified during performance of the Integrated Containment Leak Rate Test Procedure S-030001. See Section 14.2.12.1.77.

13) 14.2.12.1.64 - a) Verify that the transfer pump flow capacity (Subsection 14.2.12.1.53) is sufficient to satisfy the fuel oil consumption rates. b) Ensure that the 2 hr. and 22 hr. load tests are accomplished within a 24 hr. period.

RESPONSE

a) The fuel oil transfer pump capacity, determined in Procedure S-03JE01 (Section 14.2.12.1.53), was compared with the fuel consumption rate determined in Procedure S-03NF02 (Section 14.2.12.1.64) to verify that the pump capacity exceeds the consumption rate.

b) The 2-hour and 22-hour load tests were performed within a 24-hour period.

14) 14.2.12.1.73 - a) Account for process-to-sensor hardware (e.g., instrument lines, hydraulic snubbers) delay times; b) Provide assurance that the response time of each primary sensor is acceptable; and c) Provide assurance that the total reactor protection system response time is consistent with your accident analysis assumptions.

Note: Item 2 can be accomplished by measuring the response time of each sensor during the preoperational test, ensuring that the response time of each sensor will be measured by the manufacturer within two years prior to fuel loading, or describing the manufacturer's certification process in sufficient detail for us to conclude that the sensor response times are in accordance with design.

RESPONSE

a) See the response to Regulatory Guide 1.118, Revision 2, in Appendix 3A.

b) See the response to item (a).

c) The response times identified as acceptance criteria in Procedure S-03SB01 (14.2.12.1.73) are consistent with the technical specifications and other design documents.

15) 14.2.12.2.6 - Verify that the operability of your liquid radwaste system will be demonstrated by actually processing representative chemical waste streams.

RESPONSE

The Liquid Radwaste System Preoperational Test Procedure S-04HB01, utilizes various chemicals to verify the operability of the reverse osmosis unit. Chemical waste streams were not injected in other portions of the system, since it was not the intent of the preoperational test program to unnecessarily contaminate the system. Adequate data was recorded during the Preoperational Test Program to evaluate the system properly. The system has design provisions (i.e., heat tracing, pipe routing) to ensure proper functioning during operation with actual chemical waste streams. The ability of the liquid radwaste system to process wastes is accomplished during plant operations when wastes are generated.

16) 14.2.12.3.7 - Ensure that the moderator temperature coefficient will be derived, and that it meets the applicable criteria.

RESPONSE

See Section 14.2.12.3.7.

17) 14.2.12.3.9 - Include testing at approximately 50% power. Commit to performing step and ramp changes of full design value, or explain how changes of a lower value can be used to determine the proper response to design load swings.

RESPONSE

See the response to Regulatory Guide 1.68, Revision 2, in Appendix 3A.

18) 14.2.12.3.27 - Commit to retesting rods, whose scram times fall outside the two-sigma limit, at least three additional times.

RESPONSE

The Rod Drop Time Measurement Test Procedure S-07SF03 retests any rods, whose scram times fall outside the two-sigma limit, at least three additional times.

Q640.13 (14.2.12) We have noted on other plant startups that the capacities of pressurizer or main steam power-operated relief valves are sometimes in excess of the values assumed in the accident analyses for inadvertent opening or failure of these valves. Provide a description of the initial plant test or manufacturer's test that demonstrates that the capacity of these valves is consistent with your accident analysis assumptions.

RESPONSE

See Section 18.2.5 for performance testing of the pressurizer power-operated relief valves.

The specification for the main steam atmospheric relief valves required that no single valve capacity be greater than the value specified in the accident analysis (970,000 lbm/hr).

The valve manufacturer has indicated that the maximum flow through the valve based on design inlet pressure conditions with the valve full open is 670,000 lbm/hr. This value was determined using flow coefficients and calculational methods in accordance with ANSI/ISA approved standards.

Due to the significant margin between the actual valve capacity and the value provided in the safety analysis, no capacity testing is required.

Q640.14 (14.2.12.1) Commit to the demonstration of the operability of the temperature sensors downstream of the primary power operated relief valves and safety valves (Figure 5.1-1, Sheet 2).

RESPONSE

The pressurizer relief valve and PRT Hot Preoperational Test Procedure, S-03BB13 (Section 14.2.12.1.21), verified the operability of the temperature sensors downstream of the power-operated relief valves and safety valves.

Q640.15 (14.2.12) Failure of pressurizer overpressure protection valves to reseat, coupled with false position indication has occurred recently. One possible failure cause which has been identified was galling of the valve body due to dry stroking the valves when setting release limits. Explain what procedures will be used to protect valves during limit setting.

RESPONSE

After the pressurizer power operated relief valves have been installed in the system, the valves can be stroked since the valves are shipped in a closed position precluding any foreign material from lodging on the valve seat. Prior to preoperational testing, the valve calibration is performed which checks the closed, mid, and open positions as a minimum. This range is compared to the stroke distance of the valve to check proper travel. The limit switches and/or position indication is then set.

During operation, a periodic calibration schedule is maintained for use in checking the pressurizer power operated relief valves. At the time of calibration, the proper clearance is obtained which will isolate and/or provide the proper alignment. The valves are inspected for any damage or leaks. The open, mid, and closed positions, as a minimum, are recorded. These values are compared with the requirements in written and approved procedures to verify the travel and range.

Q640.16 (14.2.12.1) Verify that functional testing performed on valves with two actuation trains, such as the Main Steam (Subsection 10.3.2.2) and Main Feedwater (Subsection 10.4.7.2.2) Isolation Valves, includes verification of the operability of each actuation train.

RESPONSE

For those valves having two actuation trains, the operability of each actuation train is verified. See Sections 14.2.12.1.71 and 14.2.12.1.72.

Q640.17 (14.2.12.1) Correct the following deficiencies that were noted in your Containment Isolation Valve test description:

1) Subsection 14.2.12.1.10 states that Pressurizer Relief Tank Nitrogen Isolation Valves shut upon receiving a CIS, but these valves do not appear in Table 6.2.4-1.

2) The following valves should close upon receiving a CIS (Table 6.2.4-1) but are not specifically addressed in your test procedure descriptions:

HV-7,8 - Containment Spray Recirculation
FV-29 - Instrument Air to Reactor Building
FV-95,96 - Reactor Sump Pump to Floor Drain Tank
HV-8843 - Boron Injection Tank to CIS Test Line

3) Containment isolation valves should be tested in an integrated manner in as much as practicable. Note that a commitment satisfying this intent could be made in Subsection 14.2.12.1.71.4.C.

RESPONSE

1) See penetration P-62 on Sheet 2 of Table 6.2.4-1. Figure 6.2.4-1, Page 44 indicates that the valves close on receipt of a CIS.

2) The operability of containment isolation valves on receipt of a containment isolation signal was verified in the preoperational tests associated with the system to which the valve belongs. In addition, the response of the valves to a containment isolation signal was verified in the Engineered Safeguards (NSSS) Preoperational Test Procedure S-03SA01. See Section 14.2.12.1.71.

3) The intent of Procedure S-03SA01, Engineered Safeguards (NSSS) Preoperational Test, is to provide an integrated test inasmuch as practicable.

Q640.18 (14.2.12.1) Provide test descriptions 1) that will verify that the plant's ventilation systems are adequate to maintain all ESF equipment within its design temperature range during normal operations; and 2) that will verify that the emergency ventilation systems are capable of maintaining all ESF equipment within their design temperature range with the equipment operating in a manner that will produce the maximum heat load in the compartment. If it is not practical to produce maximum heat loads in a compartment, describe the methods that will be used to verify design heat removal capability of the emergency ventilation systems.

> Note that it is not apparent that post-accident design heat loads will be produced in ESF equipment rooms during the power ascension test phase; therefore, simply assuring that area temperatures remain within design limits during this period will probably not demonstrate the design heat removal capability of these systems. It will be necessary to include measurement of air and cooling water temperature and flows and the extrapolations used to verify that the ventilation systems can remove the postulated post-accident heat loads.

RESPONSE

The Plant Performance Test Procedure S-090007 (Section 14.2.12.2.27), records ambient room temperatures throughout the plant and cooling water system conditions during hot functional testing and power ascension. The recorded temperatures are evaluated to determine potential problems.

The ability of the ESF pump room coolers to maintain the ESF pump rooms within their design limits, for the conditions specified in Section 9.4.3.3, is verified throughout the test program. Each room is monitored during the period when the largest heat load is present. See Sections 14.2.12.1.29, 14.2.12.1.35, 14.2.12.1.37, 14.2.12.1.41, and 14.2.12.1.87. For rooms that do not have coolers (e.g., diesel generator rooms) the WCGS program of verifying the fan capacity provides adequate system verification.

Maintaining the containment air temperature within design limits is verified during the highest attainable heat load. See Section 14.2.12.2.27. Containment cooler fan capacity and proper cooling water flow are verified. See Sections 14.2.12.1.48 and 14.2.12.1.32, respectively. Containment cooler operation at design peak accident pressure is also verified. See Section 14.2.12.1.77.

Since the containment air cooler post-accident heat removal mechanism is mainly steam condensation, and the normal operation heat removal mechanism is the cooling of the air stream with little or no condensation, it is not possible to accurately extrapolate preoperational test data to verify the post-accident heat removal capability. On WCGS, the heat removal capability of the containment air coolers is accurately determined by sophisticated mathematical and computer modeling developed by the air cooler supplier. The accuracy of the model was verified during the prototype testing of three different coils at three different post-accident pressures. Topical Report AAF-TR-7101 (Reference 1 to USAR Section 6.2.2.3) provides a comparison of the measured heat removal during the tests to the computer analysis predictions. The comparisons show very close agreement between the predicted and actual heat removal abilities. The NRC has approved the topical report for reference in Construction Permit and Operating License applications.

Q640.19 (14.2.12.1) Modify the appropriate test description of the Engineered Safety Features System to ensure that the following items are addressed:

1) The starting of the ESF pumps should be verified for both emergency and normal power sources.

2) The SI and RHR pumps should be run under full flow conditions to verify an adequate margin to electrical trip.

3) ESF pumps should be verified able to start under maximum startup loading conditions.

4) Present or reference the full flow analysis done to satisfy the intent of Regulatory Guide 1.79, C.la(2), as committed to in Appendix 3A.

5) Ensure that the recirculation portion of the ECCS Sump Test (Subsection 14.2.12.1.83) verifies a value of NPSH greater than that required under accident temperature conditions.

RESPONSE

1) The ESF pumps were started off normal and emergency power sources in the LOCA Sequencer Preoperational Test Procedure S-03NF02. See Section 14.2.12.1.64.

2) The SI and RHR pumps were run at full flow in accordance with the tests described in Sections 14.2.12.1.34, 14.2.12.1.37, and 14.2.12.1.64.

3) See Sections 14.2.12.1.64 and 14.2.12.1.65.

4) See the response to Regulatory Guide 1.79, Revision 1, in Appendix 3A.

5) Hydraulic model testing has been performed in lieu of the initially planned in-plant test. Data obtained during model testing together with known pressure drops across suction lines and valves (determined using standard engineering calculations) verified that the available NPSH is equal to or greater than that required at accident temperatures.

Q640.20 (14.2.12.1) Recently, questions have arisen concerning the operability and dependability of certain ESF pumps. Upon investigation, the staff found that some completed preoperational test procedures did not describe the test conditions in sufficient detail. Provide assurance that the preoperational test procedures for ECCS and containment spray pumps will require recording the status of the pumped fluid (e.g., pressure, temperature, chemistry, amount of debris) and the duration of testing for each pump. In addition, provide preoperational test descriptions to verify that each engineered safety feature pump operates in accordance with the manufacturer's head-flow curve. Include in the description the bases for the acceptance criteria. (The bases provided should consider both flow requirements for ESF functions and pump NPSH requirements).

RESPONSE

The preoperational test descriptions requested are presently included. See Sections 14.2.12.1.34, 14.2.12.1.37, and 14.2.12.1.41.

Q640.21 (14.2.12.1) Our review of licensee event reports has disclosed that many events have occurred because of dirt, condensed moisture, or other foreign objects inside instruments and electrical components (e.g., relays, switches, breakers). Describe administrative controls that will be implemented to prevent component failures such as these at your facility including precautions that will be taken during initial testing program.

RESPONSE

Components such as relays, instruments, etc., are inspected prior to initial operation. At this time, a visual and/or functional check is performed. After installation, but prior to preoperational testing, the item is checked and calibrated if applicable. These measures should prevent component failure due to dirt, moisture, or other foreign objects.

During operation, a periodic calibration or preventive maintenance schedule is maintained for use in checking equipment. At this time, a check is made for damage and obstructions. These activities should prevent component failure, since they are performed on a regular basis during operation.

Q640.22 (14.2.12) For your DC Power System tests (Subsections 14.2.12.1.67, 14.2.12.2.17 and 18), verify that individual cell limits are not exceeded during the design discharge test and demonstrate that the DC loads will function as necessary to assure plant safety at a battery terminal voltage equal to the acceptance criterion that has been established for minimum battery terminal voltage for the discharge load test. Assure that each battery charger is capable of floating the battery on the bus or recharging the completely discharged battery within 24 hours while supplying the largest combined demands of the various steady-state loads under all plant operating conditions.

RESPONSE

The 125-V (Class 1E) DC System Preoperational Test, S-03NK01; 250-V DC System Preoperational Test, S-04PJ01; and 125-V (Non-Class 1E) DC System Preoperational Test, S-04PK01 verify that individual cell limits are not exceeded during the performance of their design discharge test. Section 14.2.12.1.64 addresses the verification of the safety-related 125-V DC system at minimum voltage. The ability of the battery chargers to recharge their associated battery to normal conditions, after the battery has undergone a design duty cycle, while simultaneously supplying power at a rate equivalent to the design emergency loading, largest motor current load, and the design load, within 12 hours is verified in procedures S-03NK01, S-04PJ01, and S-04PK01, respectively.

Q640.23 (14.2.12) Your test descriptions are not sufficiently detailed to ascertain if the voltage levels at the safety-related buses are optimized for the full load and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power source by appropriate adjustment of the voltage tap settings of the intervening transformers. We require that the adequacy of the design in this regard be verified by actual measurement and by correlation of measured values with analysis results. Provide a description of the method for making this verification.

RESPONSE

The Electrical Distribution System Voltage Verification Test Procedure S-090023 collects the data to be utilized to verify electrical system voltage analysis.

See Section 14.2.12.2.35.

Q640.24 (14.2.12.1) Make a commitment in your test procedure descriptions to perform the pre- and post- hot functional examination for integrity as described in Subsection 3.9(N).2.4.

RESPONSE

See Section 14.2.12.1.13.

Q640.25 (14.2) There are a number of discrepancies between Tables 14.2-1 and Table 14.2-4. Make the appropriate corrections to address the following problems:

1) S-03BB11 Reactor Coolant System Hydrostatic Test is included in Table 14.2-1 (Sheet 1) but missing from Table 14.2-4.

2) S-X3NG01 480-V Class IE System Preoperational Test is included in Table 14.2-1 (Sheet 4) but missing from Table 14.2-4.

RESPONSE

1) See Table 14.2-4.

2) See Table 14.2-4.

Q640.26 (14.2) Table 14.2-5 (Sheet 3) lists S-090007 Plant Performance Test as one of the startup tests. This test is not included in Table 14.2-3. Provide a footnote indicating that the test is a continuation of a nonsafety-related preoperational test.

RESPONSE

See Table 14.2-5.

Q640.27 (14.2)	Table 14.2-5 does not in many cases clearly indicate the power levels specified by the test method portion of the individual startup test descriptions. Modify Table 14.2-5 to indicate the power level or plateau at which each of the individual startup tests will be conducted.

RESPONSE

Table 14.2-5 has been revised to indicate the power levels specified in the test descriptions. It is not the intent of Table 14.2-5 to indicate the plateaus at which the tests are performed. Table 14.2-5 indicates the power level at which the tests begin and end. The test descriptions and test procedures indicate the plateaus at which testing is performed.

See Table 14.2-5 and the individual test descriptions.

Q640.28 (14.2.12)	The response to Item 640.18 on the Plant Performance Test (FSAR Subsection 14.2.12.2.27) should restate that the heat removal capability of the containment air coolers will be verified by extrapolation of data taken from the actual test conditions to the postulated post-accident heat load condition.

RESPONSE

Post-accident heat removal is predominantly by steam condensation (~97 percent) while the plant performance test verifies the convective cooling capability of the containment air coolers. Extrapolation of test data to postulated post-accident conditions, as requested, is thus not appropriate. Verification of post-accident heat removal capability is provided via the vendor's Topical Report which has been reviewed and approved for this purpose by the NRC (American Air Filter Topical Report, TR-7101). The response to Question 640.18 in the WCGS USAR has been revised to document this response and to reference the Topical Report via USAR Section 6.2.2.3.

Q640.29 (14.2.12)	Recent FSAR revisions have made modification to various test abstracts. Provide technical justification for each of the following test abstract modifications, or modify the test abstracts accordingly.

1) The Spent Fuel Pool Crane Preoperational Test (FSAR Subsection 14.2.12.1.54) should reinstate acceptance criteria regarding proper operation of the control circuits and associated interlocks.

2) The LOCA Sequencer Preoperational Test (FSAR Subsection 14.2.12.1.64) should reinstate acceptance criteria for load group 2 and diesel generator operation (Acceptance Criteria items j through p have been deleted).

3) The Reactor Protection System Logic Test (FSAR Subsection 14.2.12.1.73) should reinstate the acceptance criteria for all loop response times measured in the test method.

4) The Plant Performance Test (FSAR Subsection 14.2.12.2.27) should provide objectives and test method regarding evacuation alarm audibility. Alternatively, the Public Address System Preoperational Test (FSAR subsection 14.2.12.2.21) should provide acceptance criteria regarding evacuation alarm audibility in high noise areas.

RESPONSE

1) The test procedure as written for Wolf Creek include an acceptance criterion as requested.

2) These acceptance criteria were deleted inadvertently and have been reinstated as requested.

3) All loop response times are measured and recorded in this test. Response times for five of the trips are compared to typical Westinghouse values but are not subject to WCGS specific acceptance criteria since neither the WCGS USAR nor the Technical Specifications establish quantitative limits for these trips. NRC Office of Inspection and Enforcement, Region III, raised this issue at Callaway in inspection report 50-483/84-01(DE), February 22, 1984. After discussion and further review, Region III concluded that the test approach, as described previously, was acceptable. Disposition of this item is documented in inspection report 50-483/84-09, May 9, 1984.

4) The Plant Performance Test Abstract (S-090007) has been modified to include evacuation alarm audibility in the objective as requested. The test method statement previously in the abstract, together with the note under acceptance criteria, provides a reasonable description of the means by which audibility is verified. Operators are dispatched throughout the plant to verify audibility and log location and acceptability on appropriate data sheets. Problem areas are reported for corrective action.

   Much of the alarm audibility testing at WCGS was performed in conjunction with test S-04QF01, the public address system preoperational test. The test procedure included an acceptance criterion requiring alarm audibility in high noise areas. This criterion implements a portion of the more general one in the abstract, "The evacuation alarm system operates in accordance with system design specifications." This test was performed during hot functional testing high noise conditions at WCGS. Testing performed under S-04QF01 to the requirements of S-090007 was not repeated for the plant performance test.

Q730.1 The Atomic Safety and Licensing Appeal Board in ALAB-444 determined that the Safety Evaluation Report for each plant should contain an assessment of each significant unresolved generic safety question. It is the staff's view that the generic issues identified as "Unresolved Safety Issues" (NUREG-0606) are the substantive safety issues referred to by the Appeal Board. Accordingly, we are requesting that you provide us with a summary description of your relevant investigative programs and the interim measures you have devised for dealing with these issues pending the completion of the investigation, and what alternative courses of action might be available should the program not produce the envisaged result.

There are currently a total of 26 Unresolved Safety Issues discussed in NUREG-0606. We do not require information from you at this time for a number of the issues since a number of the issues do not apply to your type of reactor, or because a generic resolution has been issued. Issues which have been resolved have been or are being incorporated in the NRC licensing guidance and are addressed as a part of the normal review process. However, we do request the information noted above for each of the issues listed below:

1. Waterhammer (A-1)
2. Steam Generator Tube Integrity (A-3)
3. ATWS (A-9)
4. Reactor Vessel Materials Toughness (A-11)
5. Steam Generator and Reactor Coolant Pump Support (A-12)
6. Systems Interaction (A-17)
7. Seismic Design Criteria (A-40)
8. Containment Emergency Sump Performance (A-43)
9. Station Blackout (A-44)
10. Shutdown Decay Heat Removal Requirements (A-45)
11. Seismic Qualification of Equipment in Operating Plants (A-46)
12. Safety Implications of Control Systems (A-47)
13. Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (A-48)

RESPONSE

In the Safety Evaluation Report for Virgil C. Summer and Comanche Peak (NUREG-0717 and -0797), the NRC Staff concluded that those plants could be operated pending resolution of the unresolved safety issues. The reasoning that lead to these conclusions is applicable to WCGS. In general, WCGS agrees with the previous NRC Staff assessments of these issues and also have concluded that the WCGS can be operated without risk to the health and safety of the public. Programs and measurements taken for dealing with these generic issues are discussed below.

A-1 Waterhammer

The WCGS steam generator design incorporates a sealed thermal sleeve and J tubes on the feedring to prevent draining of water from the feedring in the event the feedwater is lost and the steam generator water level drops below the level of the feedring. The design also incorporates a short horizontal length of feedwater piping to the feedring. A waterhammer test of the feedwater system using normal plant procedures was conducted at the WCGS plant. The feedwater connection on each of the steam generators is the highest point of each feedwater line downstream of the main feedwater isolation valve. The feedwater lines contain no high pockets which, if present, could trap steam and lead to waterhammer. The feedwater inlet arrangement for a model F steam generator is of such a design as to minimize the potential for flow-induced tube vibration. A preoperational test for piping vibration and dynamic effects was conducted. For further details refer to Sections 5.4.2.2, 10.4.7.2.1, 3.9(B).2.1, and 3.9(N).2.1.

A-3 Steam Generator Tube Integrity

The WCGS design includes the Westinghouse Model F steam generator which was developed to minimize steam generator tube problems. In addition, WCGS plant use full flow condensate demineralizers and all volatile treatment (AVT) chemistry control. For further details refer to the following Sections: 5.4.2.2, 5.4.2.3.1, 5.4.2.3.3, 5.4.2.4.2, 5.4.2.5.4, 9.3.2, 10.4.6, 10.4.8, and the response to Regulatory Guide 1.121 in Appendix 3A.

A-9 Anticipated Transients Without Scram

Refer to Section 15.8.

A-11 Reactor Vessel Materials Toughness

Refer to Section 5.3 and responses to NRC questions (123.3, .4, .6, .7, .8, and .9).

A-12 Steam Generator and Reactor Coolant Pump Support

The WCGS steam generator and reactor coolant pump supports were designed to meet the fracture toughness requirements of ASME Section III, subsection NF. Westinghouse has concluded that compliance with subsection NF is sufficient to resolve the concerns expressed in NUREG-0577. Refer to Sections 3.8.3.1.2, 3.8.3.1.3, and 5.4.14.

A-17 Systems Interaction

The WCGS design is founded on principles of physical separation, independence of redundant safety systems, and protection against hazards such as high energy line breaks, missiles, flooding, seismic events, fires, and sabotage. The design has been subjected to multiple, interdisciplinary reviews. Examples of such reviews include:

a. USAR Appendix 3B describes the WCGS hazards analysis review program which was conducted on a room-by-room basis for each room in the power block. All components within the rooms were reviewed for the effects of earthquake-induced failures, effects of high and moderate energy piping breaks (flooding, sprays, and jet impingement), and the effects of missiles.

b. A separate review was also conducted on a room-by-room basis to evaluate the fire protection design and the effects of fires in each fire area as discussed in USAR Section 9.5.1.

c. The responses to NRC questions 420.3 and 420.4 describe the reviews conducted to analyze control systems failures and how such failures impact interfacing safety grade systems.

d. Heavy loads analyses as requested in NRC generic letter 81-07.

e. Review of environmental impacts on systems to ensure that they are designed to provide acceptable performance during normal and design basis accident conditions as described in WCGS USAR Sections 3.11(B) and 3.11(N).

A-40 Seismic Design Criteria

As discussed in Sections 3.7(B) and 3.7(N), the WCGS plant has been designed to current seismic design criteria.

A-43 Containment Emergency Sump Performance

The WCGS containment sumps are described in Section 6.2.2.1.2.2 and Figure 6.2.2-3 (10 sheets). Thermal insulation used inside the containment in the WCGS design will not be a significant source of debris. A detailed comparison of the WCGS sumps with the design recommendations of Regulatory Guide 1.82 is provided in Table 6.2.2-1. Sump testing is discussed in Appendix 3A response to Regulatory Guide 1.79.

A-44 Station Blackout

The offsite and onsite power systems are described in Sections 8.2 and 8.3. Several responses to NRC questions in the 430-series are related to NUREG/CR 0660. The independence of the turbine-driven auxiliary feedwater pump train from ac power is discussed in Section 10.4.9.2.2. Plans for emergency procedures and training were provided in SNUPPS letter, SLNRC 81-35 dated May 27, 1981. Specific information regarding station blackout is given in Appendix 8.3A.

A-45 Shutdown Decay Heat Removal Requirements

The WCGS design includes provisions so that cold shutdown conditions can be obtained using safety-grade equipment with only onsite or only offsite ac power. Refer to Appendix 5.4.A. As noted in that appendix, the WCGS design includes redundant, qualified, Class IE pressurizer power-operated relief and block valves.

A-46 Seismic Qualification of Equipment in Operating Plants

Current seismic criteria were used in the WCGS design. Refer to Sections 3.10(B) and 3.10(N).

A-47 Safety Implications of Control Systems

The WCGS control and safety systems have been designed with the goal of ensuring that control system failures will not prevent automatic or manual initiation and operation of any safety system equipment. This has been accomplished by providing independence or isolation between safety and non-safety systems. An analysis is documented in the response to NRC question 420.4.

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Section 6.2.5 describes hydrogen control provisions in the WCGS design. Principal containment design parameters are given in Table 6.2.1-2.

A-49 Pressurized Thermal Shock

Section 5.3 and the responses to NRC questions (123.3, .4, .6, .7, .8, and .9) provide information concerning reactor vessel material properties, material susceptibility to neutron irradiation induced embrittlement, and the increase of nil ductility transition temperature with operating life.