Docket No. 50-305

JUL 9 1971

Mr. E. W. James, Vice President
Power Generation & Engineering
Wisconsin Public Service Corporation
P. O. Box 1200
Green Bay, Wisconsin 54305

Dear Mr. James:

In reviewing your application for the Kewaunee Nuclear Power Plant, we find that we need additional information to complete our evaluation to support issuance of an operating license. The specific information required is listed in the enclosure.

A portion of the requests for information were discussed with your personnel at meetings held on March 16, April 14, May 21, and June 2, 1971. We recognize that some of the information requested may be available in the public record in the context of our regulatory review of similar features of other facilities. If such is the case, you may wish to incorporate the information by reference in your application.

As our review of your application continues, we may request additional information, particularly that corresponding to Sections 2, 3, 12, 13, and 14.

The additional information requested has been categorized into groups which correspond directly to like numbered sections in your application. You may wish to amend your application by submitting revised pages for the appropriate portions of the Final Safety Analysis Report rather than by submitting separate responses to the questions. If so, please provide cross references to those pages.

Please contact us if you desire any discussion or clarification of the material requested.

Encl:
Request for Additional info
Distribution page 2

Sincerely,

Peter A. Morris, Director
Division of Reactor Licensing

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Enclosure: Request for Additional Information

Mr. E. W. James

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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

July 9, 1971

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Power Generation & Engineering
Wisconsin Public Service Corporation
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Please contact us if you desire any discussion or clarification of the material requested.

Sincerely,

Frank Schroeder
for Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Request for Additional Information

REQUEST FOR ADDITIONAL INFORMATION
WISCONSIN PUBLIC SERVICE CORPORATION
KEWAUNEE NUCLEAR POWER PLANT
DOCKET NO. 50-305

1.0 GENERAL

1.1 On the basis of our evaluation of the seismic instrumentation system that you have proposed (see Section 1.6.10 in the FSAR), we have concluded that additional equipment and measures should be provided to assure the integrity and performance of structures, systems, and components important to safety under seismic disturbances. Provide further information and justification for the selection, placement, and use of seismic instrumentation to determine whether the plant can continue to be operated safely during and after an earthquake which affects the plant. The measures we consider that would provide a suitable basis for a seismic instrumentation system and program to determine the seismic response of nuclear power plant features important to safety and to permit comparison of such responses with those used as the design bases are set forth in Safety Guide No. 12,"Instrumentation for Earthquakes," published March 10, 1971.

1.2 Provide the following information with respect to the use of the seismic instrumentation system:

1.2.1 Describe the means that will be used to inform the control room operator, within a few minutes after an earthquake, as to the value of the peak response acceleration level experienced in the basement of the reactor containment structure.

1.2.2 Describe the means that will be used to provide direction to the control room operator as to the actions to be taken within a few minutes of the earthquake based upon the peak acceleration level experienced in the basement of the reactor containment structure.

1.2.3 Describe the plans and the analytical criteria that will be used for the timely utilization of the data to be obtained from the installed seismic instrumentation, including the basis that will be used to account for the cyclic characteristics of the earthquake, in order to obtain comparisons of the earthquake response spectra with the design response spectra specified for the site.

2.0 SITE AND ENVIRONMENT

2.1 Provide the following information to permit an independent hydrologic engineering review to be made of the Lake Michigan probable maximum water level and the effects of that level on the nuclear plant:

2.1.1 Present an analysis of the seiche (surge hydrograph) of the probable maximum water level resulting from probable maximum meteorological events coincident with a maximum monthly high lake level. The method of determining the surge produced by a moving squall presented by Platzman[1] is one satisfactory technique for computing the probable maximum water level by applying the combined wind and pressure effects associated with maximum meteorological criteria. The probable maximum meteorological parameters that have been used with the Platzman theory in establishing the probable maximum surge (still water level) at other nuclear power plants sited on Lake Michigan, are as follows:

(1) Pressure jump: 0.21 inches Hg

(2) Wind speed: 65 knots

(3) Direction and speed of propagation - critical combination that produces maximum surge.

2.1.2 Describe the shore protection fronting the plant: i.e., riprap, structures, and slopes. Using the analysis techniques described in Reference 4 in the FSAR, or techniques that are similar, determine the probable maximum coincidental wave heights and runup on safety-related plant components.

2.1.3 Describe the location and the waterproofing or protective measures that will be taken to prevent flooding of areas containing safety-related equipment and located below the elevation defined by the probable maximum water level plus runup.

2.1.4 Determine and substantiate the capability of the safety-related features of the plant to withstand the dynamic loadings that would be imposed when the probable maximum lake level conditions occur. In particular, the steel splitter in the screenhouse forebay, the concrete and sheet piling discharge structure, the lake side of the screenhouse, and the intake crib should be analyzed for dynamic loadings with the level of the lake at its probable maximum level.

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[1] "Prediction of Surge in Southern Basin Lake Michigan," Platzman, Part I, Irish Part II, Highes, Part III, published in Monthly Weather Review, Volume 93, No. 5, May 1965, and unpublished data from Dr. Platzman.

2.1.5 Figure 1.2-9 in the FSAR illustrates the interior accesses to the screenhouse forebay at elevation 586.0. If the probable maximum water level at the intake crib is approximately elevation 586.0 or above, or if water can reach the open forebay at higher elevations, describe how these accesses are sealed to prevent flooding of the circulating water pump room and other areas of the plant through the access tunnel. Describe any flood protection features that have been provided.

2.2 Provide an estimate of the probable minimum water level for the lake and its effects on the operation of the plant. The probable minimum water level could be determined by review of the original unpublished work by Platzman (See Footnote 1, page 2) that indicates that the depth of the trough of the surge wave can be equal to the height of the surge wave. Include drawings of the intake crib showing pertinent elevations, including those relating to the minimum pump suction heads required for normal and emergency operation.

2.3 Will the plant be free of potential flooding from a local probable maximum flood (PMF) resulting from probable maximum precipitation[2] on the small stream immediately south of the plant?

2.4 Provide a description of the off-shore structures that is sufficiently detailed that an independent analysis of effluent diffusion can be made. Provide information to verify the acceptability of the assumptions made in your diffusion analysis; in particular, the assumed diffusion relationship, lake water velocities, and alongshore restricted diffusion factors should be substantiated for the plant effluent location.

2.5 Provide the location and demand of private and public surface and ground water users within several miles of the site. Locations of user sources relative to the plant should be indicated on a map.

2.6 Analyze the potential consequences of a spill of radioactive material on local ground water users. Include the consideration of each of the following conditions in your analyses:

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[2] Hydrometeorological Report No. 33, "Seasonal Variation of the Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1000 Square Miles and Durations of 6, 12, 24, and 48 Hours," April 1956, by U.S. Weather Bureau

2.6.1 A surface spill reaching the ground water through wells, old bore holes and other surface-ground water interfaces, in addition to normal percolation.

2.6.2 The drawdown effects of local wells on the water table at the plant site.

2.6.3 The possibility of a reversal of local ground water gradients due to high lake levels.

4.0 REACTOR COOLANT SYSTEM

4.1 Recently, it has been determined that sensitized stainless steel components have been used in safety and relief valves installed on the reactor coolant systems of certain plants. In particular, type 304 and 316 stainless steel nozzles and bushings that have been fabricated with a hard faced interior surface by a preheat process that resulted in severe sensitization have been provided for nuclear plants. The use of such components is of concern. Provide a discussion of the safety and relief valves for the Kewaunee plant, indicating if any pressure-containing parts were or could have been sensitized to a significant extent. If such sensitization was or is present, describe the corrective actions that have been or will be taken.

4.2 On page 4.2-1 of the FSAR, it is stated that the plant can be operated at 10% power with one loop out of service.

4.2.1 Provide a description of plant behavior when operated, in this mode, addressing in particular the flow patterns and heat transfer characteristics in the core.

4.2.2 Is it your intention to operate the plant with one loop out of service? If so, under what conditions and for how long?

4.2.3 Describe the changes that must be made to safety settings to enable one loop operation. Are these adjustments made manually or automatically?

4.3 The low head safety injection lines terminate in nozzles injecting into the reactor outlet plenum. The location of these nozzles is not clear from the description and the drawings provided. Provide a sketch or drawing showing the location of these low-head injection nozzles. If these nozzles penetrate the core barrel, provide a summary of the analysis that was performed to assure that the design is adequate to accommodate thermal stress and vibration.

4.4 Provide additional information in Tables 4.3-1 and 4.3-2 in the FSAR to show the stress intensities and cumulative fatigue usage factors for reactor vessel components.

4.5 Recent fracture toughness test data indicate that the current ASME Code rules do not always provide adequate fracture toughness of ferritic materials. The Charpy V-notch tests are adequate to measure the upper shelf fracture energy value; however, they generally do not predict correctly the Nil Ductility Transition (NDT) temperature or the transition temperature region in which fracture toughness increases rapidly with temperature.

The NDT temperature, therefore, must be obtained from other tests, such as the dropweight test (DWT). In addition, the transition temperature region shifts to higher temperatures when the thickness of the specimen tested is increased (size effect). In order to be able to establish appropriate heatup and cooldown limits for this plant, provide the following information:

4.5.1 For all pressure-retaining ferritic components of the reactor coolant pressure boundary whose lowest pressurization temperature* will be below 250° F, provide the material toughness test requirements and data (Charpy V-notch impact test curves and dropweight test NDT temperature, or others) that have been specified and reported for plates, forgings, piping, and weld material. Specifically, for each component provide the following data or your estimates based on the available data.

(1) The maximum NDT temperature as obtained from DWT tests,

(2) The maximum temperature corresponding to the 50 ft-lb value of the C_v fracture energy, and

(3) The minimum upper shelf C_v energy value for the weak direction (WR direction in plates) of the material.

4.5.2 Identify the location and the type of the material (plate, forging, weld, etc.) for which the data provided in response to item 4.5.1 above were obtained. Where these fracture toughness parameters occur in more than one plate, forging, or weld, provide the information requested in 4.5.1 for each of them.

4.5.3 For reactor vessel beltline materials, including welds, specify:

(1) The highest predicted end-of-life transition temperature corresponding to the 50 ft-lb value of the Charpy V-notch fracture energy for the weak direction of the material (WR direction), and

(2) The minimum upper shelf energy value that will be acceptable for continued reactor operation toward the end-of-service life of the vessel.

4.5.4 Furnish the proposed heatup and cooldown curves that will be used to control the pressure and temperatures to which the ferritic material of the reactor coolant pressure boundary will be exposed during the first 2 years of operation and at the end of the service life.

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*lowest pressurization temperature of a component is the lowest temperature at which the pressure within the component exceeds 25% of the system normal operating pressure, or at which the rate of temperature change in the component material exceeds 50°F/hr under normal operation, system hydrostatic tests, or transient conditions.

4.6 State the number of Charpy V-notch specimens oriented with respect to the weak direction (WR orientation in plates) of plates, forgings, and weld materials that will be included in the reactor vessel material surveillance program.

4.7 Describe the plans that were followed to avoid partial or local severe sensitization of austenitic stainless steel during heat treatments and welding operations for core structural load bearing members and component parts of the reactor coolant pressure boundary. Describe welding methods, heat input, and the quality controls that were employed in welding austenitic stainless steel components.

4.8 If nitrogen was added to stainless steel types 304 or 316 to enhance its strength (as permitted by ASME Code Case 1423 and USAS Case 71), provide justification that such material will not be susceptible to stress corrosion cracking under severely sensitized conditions.

4.9 The FSAR (see page 4.2-4) refers to Westinghouse Topical Report WCAP-7477L to justify the use of sensitized stainless steel in components of the reactor coolant pressure boundary. Westinghouse personnel stated in a meeting on August 18, 1970, that this report was being revised. Submit revised copies of this report so that the latest available information on this subject can be used in our evaluation of your application.

4.10 Provide the following information regarding the primary coolant pump flywheel:

4.10.1 Indicate the nil-ductility transition (NDT) temperature of the primary coolant pump flywheel material, as obtained from dropweight tests (DWT), the minimum acceptable Charpy V-notch (C_v) upper shelf energy level in the weak direction (WR orientation in plates), and the fracture toughness of the material at the normal operating temperature of the flywheel.

4.10.2 State if the calculated combined primary stresses in the primary coolant pump flywheel at the normal operating speed include the stresses resulting from the interference fit of the wheel on the shaft, and the stresses due to centrifugal forces.

4.10.3 State the highest anticipated overspeed of the flywheel and the basis for this assumption.

4.10.4 State the estimated maximum rotational speed that the flywheel could attain in the event that the reactor coolant piping ruptures in either the suction of discharge side of the pump. In addition, describe the results of any studies directed towards: (1) determining the top speed the pump or motor can reach due to physical limitations

(e.g., the speed at which the pump impeller seizes in the wear rings due to growth from centrifugal force or the speed at which motor parts come loose and grind or bind to prevent further increase in speed); (2) establishing speed and torque for pipe break sizes; (3) devising means to disengage the motor from the pump in the event of pump overspeed; (4) verifying that pump parts generated as missiles at top speed do not penetrate the pump casing and that any parts leaving in the blowdown jet do not penetrate containment; (5) establishing failure speeds for motor parts and whether they will penetrate the motor frame and if so with what energy; (6) defining a minimum rotor seizure time.

4.10.5 Describe the inservice inspection program proposed for the flywheels, state the access provisions, type and frequency of inspections, and the acceptance criteria.

4.11 If any component within the reactor coolant pressure boundary has been designed or fabricated outside of the United States provide the following information:

4.11.1 Identify the manufacturer and describe his qualifications, experience in the construction of nuclear power plant components, and experience in furnishing components for nuclear power plants in the U.S.

4.11.2 Describe the steps that have been taken to assure that the quality levels achieved in the fabrication of foreign procured components are equivalent to those obtained from U. S. manufacturers.

4.12 Describe the design and arrangement provisions for access to the reactor coolant pressure boundary as required by Sections IS-141 and IS-142 of Section XI of the ASME Boiler and Pressure Vessel Code - Inservice Inspection of Nuclear Reactor Coolant Systems. Indicate the design improvement applied to the reactor vessel, in particular, to facilitate inservice inspection.

4.13 Section XI of the ASME Boiler and Pressure Vessel Code recognizes the problems of examining radioactive areas where access by personnel will be impractical, and provisions are incorporated in the rules for the examination of such areas by remote means. In some cases the equipment to be used to perform such examinations is under development. Provide the following information with respect to your inspection program:

4.13.1 Describe the equipment that will be used or is under development for use in performing the reactor vessel and nozzle inservice inspections for the plant.

4.13.2 Describe the system to be used to record and compare the data from the baseline inspection with the data that will be obtained from subsequent inservice inspections.

4.13.3 Describe the procedures to be followed to coordinate the development of the remote inservice inspection equipment with the access provisions for inservice inspection afforded by the plant design.

4.14 Describe the ability of the proposed leakage detection systems to differentiate between identified and unidentified leaks from components within the primary reactor containment and indicate which of these systems will provide means for locating the general area of a leak.

4.15 Discuss the adequacy of any system that depends on reactor coolant activity for the detection of changes in leakage to perform reliably during the initial period of plant operation when the coolant activity may be low.

4.16 Estimate the normal total leakage rates and major sources of leakage anticipated for your plant on the basis of operational experience from other plants of similar design.

4.17 Specify the proposed maximum allowable leakage rate from unidentified sources in the reactor coolant pressure boundary, and the basis for the proposed limit. In addition, furnish the following information:

4.17.1 The length of a through-wall crack that would leak at the rate of the proposed limit, as a function of wall thickness.

4.17.2 The ratio of that length to the length of a critical through-wall crack, based on the application of the principles of fracture mechanics.

4.17.3 The mathematical model and data used in such analyses.

4.18 Provide a discussion of the tests to be performed to demonstrate the sensitivities and operability of the leakage detection systems.

4.19 Provide additional information concerning the support structure for the reactor vessel (see page 4.2-4 in the FSAR). Specifically, explain the design and function of the support pads. How are these pads cooled? What are the design thermal gradients across these pads and what tests or measurements are planned to assure that the performance is in accord with the design?

4.20 Provide the basis for the sizing of the power operated relief valves and the code safety valves on the pressurizer. What transient or accident condition was used to determine the worst overpressure condition requiring operation of the safety valves?

4.21 The list of transients that have been used in the design of components within the reactor coolant pressure boundary as specified in Section 4.15 and Table 4.1-8 of the FSAR appears to be incomplete. Identify all design transients and their number of cycles, such as transients induced by control system or other system malfunction, component malfunctions, any single operator error, and inservice hydrostatic tests that are specified in the ASME Code-required "Design Specifications" for the components of the reactor coolant pressure boundary. Categorize all transients or combinations of transients with respect to the conditions identified as "normal", "upset", "emergency", or "faulted" as defined in the ASME Section III Nuclear Vessel Code (Summer 1968 Addenda).

4.22 Specify the editions of the component codes and addenda (or the codes "in effect") that have been used for all components within the reactor coolant pressure boundary including the code case interpretations that have been applied.

4.23 Paragraph 101.5.4 of USAS B31.1.0 - 1967 edition of the Code for Pressure Piping requires that piping shall be supported to prevent excessive vibration under startup and operating conditions. Provide a description of the vibration operational test program that will be used to verify that the piping and piping restraints within the reactor coolant pressure boundary have been designed to withstand dynamic effects resulting from valve closures, pump trips, and other anticipated events. Provide a list of the transient conditions and the associated anticipated events (i.e., pump trips, valve actuations) that will be used in the vibration operational test program to verify the integrity of the system. Include those transients introduced in systems other than those within the reactor coolant pressure boundary that could result in significant vibration response of reactor coolant pressure boundary systems and components.

4.24 Did the criteria that were used to examine the effects of pipe ruptures consider the occurrence of ruptures at any location within the reactor coolant pressure boundary, or at limited areas within the system? Provide confirmation that both longitudinal and circumferential type ruptures were evaluated and describe the basis for the design approach.

4.25 Provide a more detailed description of the measures that were used to assure that the containment vessel and all essential equipment within the containment, including components of the primary and secondary coolant systems, engineered safety features, and equipment supports have been adequately protected against blowdown jet forces and pipe whip. The description should include:

4.25.1 Pipe restraint design requirements to prevent plastic hinge formation.

4.25.2 The features provided to shield vital equipment from pipe whip.

4.25.3 The measures taken to physically separate piping and other components of redundant engineered safety features.

4.25.4 A description of the analyses performed to determine that the failure of lines, with diameters of 3/4 inch or less, will not cause failure of the containment vessel under the most adverse design basis accident conditions.

4.25.5 The analytical methods that were used.

4.26 Section 1.6.11 of the FSAR implies that vibration test data obtained for the R. E. Ginna reactor plant will be used to establish the capability of the Kewaunee core support structures and other internals components to sustain flow induced vibration effects. Provide a detailed comparison of the applicable design parameters for the Ginna and Kewaunee units verifying that no significant design or fabrication differences exist between the subject reactors that could materially affect the vibrational response characteristics of the reactor internals. If the data obtained from the Ginna reactor can be shown conclusively to be applicable to the Kewaunee reactor, we would consider the Ginna reactor to be an acceptable prototype reactor for assessment of vibrational response, but we would expect additional measures to be taken to confirm the acceptability of the vibrational response of the Kewaunee reactor internals such as those indicated below. Discuss your intent to include such measures in the preoperational functional testing program for this plant, and explain the bases for any differences in your response.

4.26.1 The reactor internals important to safety should be subjected, during the preoperational functional testing program, to all significant flow modes of normal reactor operation for a sufficient period of time to duplicate the number of vibration cycles imposed on the prototype reactor.

4.26.2 After the reactor internals have been subjected to the significant flow modes of normal reactor operation, visual or surface examinations of the reactor internals should be conducted to detect any evidence of excessive vibrations, and the presence of flaws or wear induced by unanticipated vibrations. These examinations should be conducted for all major load-bearing structural elements whose failure could adversely affect the structural integrity of the reactor internals, and at all areas of lateral, vertical, and torsional restraints provided within the reactor vessel.

4.26.3 In lieu of the visual or surface examinations of the reactor internals specified in 4.26.2, a vibration test program may be implemented during the preoperational functional testing program by using sufficient and appropriate vibration-measuring instrumentation to detect the predominant vibratory responses observed in the prototype reactor.

4.26.4 A summary of the inspection of 4.26.2, or the results from the vibration test program of 4.26.3, should be the subject of a report, submitted to the Commission within 3 months after completion of the inspections or tests.

4.27 Section 14.3.3 of the FSAR states that analyses of two-loop PWR internals for LOCA and seismic loadings are continuing. Indicate when the results of these analyses, applicable to the Kewaunee reactor, will be submitted for our review.

4.28 Describe the design and installation criteria for the mounting of the pressure-relieving devices (safety valves and relief valves) within the reactor coolant pressure boundary and on the main steam lines outside of containment. In particular, specify the design criteria that were used to take into account full discharge loads (i.e., thrust, bending, torsion) imposed on valves and on connected piping in the event the valves discharge concurrently. Indicate the provisions made to accommodate these loads.

4.29 Appendix B of the FSAR delineates the emergency and faulted operating condition categories that were applied to certain reactor coolant system components. Identify any other components or systems that are not a part of the reactor coolant pressure boundary for which the design stress limits associated with faulted conditions were applied. In the event that a faulted operating condition category was applied to any system or component exclusive of those within the reactor coolant pressure boundary, provide the bases for such application including the loading combinations and associated design stress limits.

5.0 CONTAINMENT SYSTEM

5.1 Figures 5.2-5 and 5.2-7 in the FSAR do not clearly indicate the manner in which the multiple-flued heads of the penetration assemblies interface with the shield building. Provide a description indicating how adequate support is provided for the penetration and the process line while still assuring that relative motion between the multiple-flued head and the shield building is allowed.

5.2 Discuss the support provided, if any, for electrical cables in the annulus and indicate how this support interacts with the containment shell and the shield building. If no support is used, provide the basis for this design.

5.3 Provide details regarding the flexible expansion joint separating the shield building from adjacent structures. Indicate the leak tightness of the joint and the provisions that have been made to assure continued joint integrity during plant lifetime.

5.4 Figure 5.4-1 in the FSAR shows a Containment Cleanup System consisting of fans and particulate-absolute-charcoal filters. This system is not described in the FSAR. Provide a description of the system including its design bases, and indicate when and under what conditions it is to be operated.

5.5 For penetrations having the expansion bellows, define the "maximum internal pressure" to which the double bellows seal can be tested (see page 5.6-1 in the FSAR). What is the basis for the selection of this pressure capability?

5.6 During an OBE or DBE the steel containment vessel is subjected to dynamic vertical and horizontal loads and dynamic overturning moments. Explain the mode of transmission of these loads into the concrete foundation and into the subfoundation. Indicate in detail the method of analysis used to determine the corresponding stresses in the steel shell and the concrete foundation and demonstrate that the critical stresses in the steel and the concrete at these locations meet the criteria for allowable design stresses.

5.7 The arrangement of the interior concrete inside of the steel containment vessel is such that the contact surface between the steel and the interior concrete is not axisymmetric. Therefore during a LOCA the pressure and the temperature distribution in the steel shell and in the concrete would not be axisymmetric. Considering that thin shells are very sensitive to loadings that are not axisymmetric, describe the investigations made to analyze this lack of axial symmetry in the interaction of the concrete and the steel structure, and demonstrate that the resulting critical stresses in the shell and the interior concrete meet the criteria for allowable design stresses.

5.8 It is stated on page 5.2-3 of the FSAR that the operating loads on the steel shell include the piping reactions at penetrations resulting from thermal movements. Because of the interactions between exterior structures, interior structure, and the steel shell, it is conceivable that under certain conditions, the reaction forces on the steel shell penetrations for certain pipes could be reversed. This could result in exterior compression loads acting on the penetration and the shell, and cause modified reactions on the shell anchors and the concrete. Describe the appropriate analyses that have been performed to investigate this problem and the provisions that have been made to prevent the development of a condition that could damage the steel shell or its anchorage features.

5.9 The lower part of the steel shell is sandwiched between two layers of concrete and therefore not accessible for inspection during the lifetime of the plant. Describe the measures taken during construction to detect leakage through the bottom plates and the welded seams. Describe the measures to be taken to detect any possible corrosion of the plates or the welded seams during the lifetime of the plant.

5.10 It is stated on page 5.2-11 of the FSAR that low viscosity chemical grout has been used at the ellipsoidal bottom of the steel shell. Provide the composition of this grout and discuss its potential corrosive effect on the steel plates and seam welds.

5.11 The lower part of the steel shell includes a transition zone between the top of the internal concrete and the top of concrete outside of the shell. Describe the provisions made in the joints between the steel and concrete to prevent corrosion of the steel shell in this zone. Provide the surveillance program that is to be in effect to monitor for corrosion in this zone during the lifetime of the plant.

5.12 The description of the method of analysis of the concrete-steel shell interaction in the embedded zone (see pages 5.2-12 and 5.2-13 of the FSAR) is not clear. Explain the assumptions made on loads, elastic properties of the structures, and the mechanics of the interaction, and demonstrate that the deformations of the structures are compatible and that each structure, considered as a free body, is in equilibrium for axisymmetric and non-axisymmetric loads.

5.13 Provide the quality control procedures used and the tolerances specified for the removal of the temporary columns and stiffeners in the knuckle area.

5.14 Describe the method of analysis used to evaluate the spring constants that were assumed in the investigation of the seismic effects on the personnel and equipment locks taking into account the interactions with concrete. Provide results of analyses to demonstrate that the critical stresses due to an OBE and DBE in the containment shell and in the locks, and especially in their connecting welds, (see page 5.2-18 in the FSAR), meet the criteria for allowable design stresses.

5.15 We understand that soil-structure interactions have been investigated for the shield building using a finite element method of analysis and by including springs in the model used to study the dynamic behavior of the structure. Provide the following additional information:

5.15.1 What finite element method was used to represent the non-axisymmetric seismic loads?

5.15.2 Describe the method used for the transition from finite elements to the springs, and from the springs to the resulting continuous dynamic load distribution under the base slab and in the lateral walls of the structure.

5.16 Describe the effect of the seismic and tornado design loadings on the dimensioning of the reinforcing in the dome, in the wall, and in the discontinuity zones at the base of the wall and at the dome. Explain the method used to evaluate the effects of "lobar" motion. Indicate the critical stresses and their locations. (Reference is suggested to the paper by Isao Toriumi "Model Analysis of Aseismic Design of a Nuclear Reactor Building," Nuclear Structural Engineering 2 (1965) pp. 301-305.)

5.17 Provide typical sketches showing the reinforcing pattern in the base slab, in the wall, in the dome, and at special points such as the discontinuity zones in the wall, the base and the dome, and at large openings. These sketches should show the shear reinforcing and the provisions for anchorage of bars in the tension zone.

5.18 Describe the interaction between the penetration assemblies and the concrete shield building under seismic, pressure, temperature and jet loads.

5.19 For all bellows assemblies that are part of the containment boundary, provide a discussion of the methods of analysis used to establish the critical stresses and deformations. Include in this discussion the fuel transfer tube. For bellows assemblies not accessible for visual inspection indicate the special provisions or tests that will be made for surveillance during the lifetime of the plant.

5.20 Expand Table B.6-6 in the FSAR to indicate for each internal and external missile the mass to cross sectional area ratio, the impact velocity, the impact point, and the impact angle and the source and type of missile.

5.21 Describe the method of analysis used to evaluate the impact stresses of missiles on the Class I (seismic) structures. Also describe the missile protection for the Class I (seismic) structures.

5.22 Describe the provisions made to support and anchor Class I (seismic) equipment on Class I (seismic) or Class II (seismic) structures. Include provisions for anchoring the equipment against seismic, forces, tornado forces, and jet forces. Present a discussion, with typical sketches, of the supports and the anchors for the reactor vessel, steam generators, pressurizer, main coolant pumps, control room equipment, batteries, and switch gears.

5.23 For the containment building interior structures, describe the loadings used in the design, the methods of analysis for thermal, jet, seismic, pressure and mechanical loads, and the design criteria and methods. Provide the results of analyses to demonstrate that the critical stresses meet the criteria for allowable design stresses.

5.24 Discuss the thermal effects in the concrete structures adjacent to and at the level of the reactor core that result from neutron absorption. Describe methods and the criteria used for the design of the concrete, the reinforcing steel, and the embedded structural steel members in these structures.

5.25 Provide sketches of the containment building interior structures showing the reinforcing steel patterns.

5.26 Describe the methods of analyses that were used for structures consisting in part of Class I (seismic), Class II (seismic), and/or Class III (seismic) elements, and the procedures used in the design for seismic loads generated by the OBE and the DBE.

5.27 For cases where a structure of a lower seismic design classification is adjacent to a structure of a higher classification, for instance a Class II (seismic) structure adjacent to a Class I (seismic) structure describe the provisions that were made in the design of the lower classification structure to prevent damage to the higher classification structure under conditions associated with design basis seismic or tornado events.

5.28 Describe the provisions made to tie down removable slabs, blocks, and partitions, in order to prevent them from becoming missiles.

5.29 For all Class I (seismic) structures, including the containment structure, describe the provisions made to prevent ground water infiltration into the part of the structures located below the highest ground water level.

5.30 Has a cathodic protection system been provided for the Class I (seismic) structures? If so, describe the system and the basis for its design.

5.31 In regard to the spent fuel pool, provide the assumptions and results of an analysis to show that the fuel pool can withstand, without leakage which would uncover the fuel, the impact of a dropped fuel cask from the maximum height to which it can be lifted by the crane. If this incident might result in leakage of water from the pool, estimate the rate of water loss from the pool and compare this rate with the available makeup capability.

5.32 The allowable stresses for concrete structures including the shield building (see Tables 5.2-4 and 5.9-1 in the FSAR) are listed, for the worst case, as:

$$f_c = 0.85\ f'_c$$

$$f_s = 0.90\ Y.S$$

The allowable stress for concrete ($0.85\ f'_c$) equals the rupture stress for concrete beams in bending and thus provides no safety margin. In contrast, the allowable stress for steel, (0.90 Y.S.) is well below the utlimate strength of steel and provides a large safety margin. Demonstrate that, despite this lack of consistency in the safety margins for concrete and for reinforcing, a balanced design has been achieved for the steel percentages used in your design.

5.33 Permanent deformations have been permitted in some cases (see page B.6-4 in the FSAR) for safe shutdown requirements. List the structures designed using this criterion and justify its use in each case. Demonstrate that this criterion does not violate the design criteria pertaining to allowable stresses.

5.34 The use of direct stress superposition (see page B.6-5 in the FSAR) is correct only for perfectly elastic materials. Describe and justify the actual procedure used in the design and dimensioning of metals beyond the yield point and of reinforced concrete members.

5.35 Explain the provisions made for bond and anchorage of reinforcing bars in concrete under tension. List the allowable stresses used in the design and demonstrate that the structures have an acceptable safety margin against bond and anchorage failure of main reinforcing.

5.36 Describe the design methods and the design criteria that were used to evaluate the stresses in and the safety factors provided for steel elements cast in concrete structures supporting Class I (seismic) equipment.

5.37 The information in Tables 5.2-4, 5.2-5, and 5.2-1 indicates that for loading condition 3, the stress in reinforced concrete is 1.5 times the ACI 318-63 allowable stress. Explain whether this applies to the working stress or the ultimate strength design.

5.38 Indicate whether the concrete samples for compression and slump tests were taken at the point of concrete placement or at some other point.

5.39 Indicate whether full size bar samples were used for user tests of reinforcing bars and whether the yield strength was checked in accordance with ASTM specifications.

5.40 We understand that the hoop reinforcing bars near the dome in the shield building have been spliced by welding to structural angles. Provide information to demonstrate that this type of splicing fully develops the bar strength and will not be a crack starter.

5.41 Describe the quality control procedures that were used for the Cadweld splices, for the Cadweld powder, and for the welding electrodes used in the welding of the steel plates and the reinforcing bars in the Class I (seismic) structures, including the steel containment shell and the shield building.

5.42 Describe any structures composed of high density concrete for shielding purposes, and specify the mix, and the type of aggregates used in the concrete.

5.43 List the specified construction tolerances for the containment structures and indicate where they were exceeded and by how much. Discuss the consequences of exceeding tolerances on the functional safety of the structural system.

5.44 The shield building is designed for an internal pressure of 3 psig due to tornado loading and a pressure rise in the annulus of 0.5 psig following a LOCA. To assure that the building will meet the design requirements an acceptance strength pressure test should be considered for the shield building and the design should permit tests to be performed during the lifetime of the plant, in the event that such tests are needed. In this regard provide the following information:

5.44.1 Specify the planned test pressure and provide the basis for its selection. This test should be representative, insofar as practicable, of the actual conditions that will exist during design basis events including (1) a tornado, and (2) a combined LOCA and DBE.

5.44.2 Describe the measurements and the inspection procedures that will be used during the tests.

5.44.3 Describe the acceptance criteria for the test.

5.44.4 Describe the permanent installations that will be provided to ensure a sufficient degree of accessibility to all the critical parts of the shield building.

5.45 Describe the proposed test procedures for Type B (local leakage at penetrations), and Type C (isolation valve) tests for both preservice and inservice containment leakage tests.

5.46 Provide test intervals for the Type B and C proposed inservice containment leakage tests.

5.47 Provide the following additional information regarding the design and performance of the shield building ventilation system (see Section 5.5 in the FSAR) and the auxiliary building special ventilation system (see Section 9.6 in the FSAR) (assume a fission product source term within the containment equivalent to that defined in Safety Guide No. 4):

5.47.1 Provide a malfunction analysis for the electric heaters in the filter trains including the consequences of overheating the air following a design basis accident. Indicate whether air temperature sensors and heater-to-fan electrical interlocks (as described in the Preliminary Safety Analysis Report) are included in the final design and discuss their significance to safety.

5.47.2 Provide a malfunction analysis for the charcoal filters following a design basis accident including a conservative calculation of charcoal temperature assuming the maximum heat from absorbed iodine, that the electric heaters remain on, and the loss of forced-flow from the fan. Provide a discussion of the design features that will prevent iodine desorption or charcoal ignition for the assumed malfunctions.

5.47.3 Provide the material specification for the high efficiency particulate (HEPA) filters, including the filter material (e.g., fiberglass), adhesives, and binders. Provide information to show that radiation from the fission products retained on the filter following a design basis accident or heat from the electric heaters will not cause functional failure of the filter materials.

5.47.4 Provide final design information including a drawing showing the arrangement and spacing of the heaters and filters, and a tabulation of the charcoal filter design parameters. For the charcoal filters, list the type of charcoal, the arrangement and number of drawers in each filter unit, the number of filter units in each ventilation train, weight and thickness of charcoal in each drawer, and the minimum and maximum filter face air velocity for the range of system air flows.

5.47.5 Provide the test procedures for measuring the portion of system air flow that may bypass each filter.

5.47.6 Provide a description of the periodic tests to be performed to monitor the HEPA filter efficiency. If DOP tests similar to those for recently-licensed plants will be used, provide a discussion of the effects of residual DOP in the HEPA filters on charcoal filter performance following the design basis accident. Include in your discussion, the amount of DOP that could be trapped in the HEPA filters from periodic tests, the amount of DOP released from the HEPA filters and trapped in the charcoal filters due to air temperature increase following a design basis accident, and the adverse effects of DOP on the charcoal filter functional capabilities, such as decreasing its capability to retain elemental iodine, converting elemental iodine to an organic form and lowering the charcoal ignition temperature due to the presence of DOP.

5.48 Topical reports JAB-PS-01 and JAB-PS-03, prepared by John A. Blume and Associates, provide details of the seismic analyses performed for this plant, and contain recommendations concerning analytical techniques that should be used for piping, equipment, and structures. Indicate the extent to which each of the recommendations presented in the reports has been followed.

5.49 Specify the method of seismic analysis (modal analysis response spectra, modal analysis time history, equivalent static load, or empirical (tests) analysis) that was employed in the design of the listed Class I (seismic) structures, systems and components. Provide the basis that was used to determine the amplified vertical response loadings for the seismic design of structures, systems and components including the effect of the vertical and horizontal response of equipment and components on the input that was used for piping and instrumentation.

5.50 Provide the criteria that were used to compute shears, moments, stresses, deflections and/or accelerations for each seismic-excited mode and for the combined total response, including the criteria for combining closely spaced modal frequencies.

5.51 Provide the various assumptions made regarding structure material properties and soil-structure interactions. Describe the measures that were taken to assure that the calculated responses of Class I (seismic) structures conservatively reflect the expected variations in the periods of vibration of the structures.

5.52 With reference to the seismic analysis of Class I (seismic) items by the response spectrum method using floor response spectra, the shape of these floor response spectra is dependent on the assumptions made for the structural properties, dampings, and soil-structure interactions. Describe the measures that were taken to consider the effects on the floor response spectra of expected variations in structure response.

5.53 The use of both the modal analysis response spectrum and time history methods provides a check on the response at selected points in the station structures. List the responses obtained from both methods at selected points in the Class I (seismic) structures in order to provide a basis for checking the seismic system analysis.

5.54 Provide justification for the use of static factors to account for torsional accelerations (see Table I of JAB-PS-03) in the seismic design of Class I (seismic) structures in lieu of a combined vertical, horizontal, and torsional multi-mass dynamic analysis.

5.55 Describe the evaluation performed to determine seismic induced effects of Class II (seismic) piping systems on Class I (seismic) piping.

5.56 For Class I (seismic) piping buried or otherwise located outside of the containment structure, describe the seismic design criteria employed to assure that allowable piping and structural stresses are not exceeded due to differential movement at support points, at containment penetrations, and at entry points into other structures.

5.57 Provide the seismic design criteria employed to assure the adequacy of Class I (seismic) mechanical components such as pumps, heat exchangers, and electrical equipment such as cable trays, battery racks, instrument racks and control consoles. Describe the measures taken for seismic restraint to meet these criteria, the analytical or testing methods employed to verify the adequacy of these restraints and the methods utilized to determine the seismic input to these components.

5.58 Topical Report WCAP-7397-L, "Seismic Testing of Electrical and Control Equipment," is referenced in the FSAR; however, in this report, vertical and horizontal excitations were considered separately. Discuss the response of this equipment when subjected to combined excitations.

5.59 Provide the criteria employed to determine the field location of seismic supports and restraints for Class I (seismic) piping, piping system components, and equipment, including the placement of snubbers and dampers. Provide the procedures followed to assure that the field location and characteristics of these supports and restraining devices are consistent with the assumptions made in the dynamic analyses of the system.

5.60 In order to assess that the seismic design bases for the structures, systems, and components of this plant have been properly translated into the required specifications, drawings and procedures that assure adequate designs capable of withstanding seismic and other concurrent loads, provide the following information:

5.60.1 Identify the design organizations involved in the seismic design of all safety-related items of the plant, their respective responsibilities, and the documented procedures followed to assure that these responsibilities have been met. Identify the organization that was assigned overall responsibility for the adequacy of seismic design.

5.60.2 In regard to the interchange of design information among the involved design organizations, revisions thereto, and coordination of all aspects of the seismic design, describe the documented procedures employed to assure that the interchanges and coordination among design organizations were followed.

5.60.3 Describe the design control measures instituted to verify the adequacy of the seismic design and identify the responsible design groups or organizations that performed this function.

5.60.4 Describe the requirements that were included in the purchase specifications for safety-related equipment to assure adequate design and functional integrity under the seismic design conditions. Describe the provisions that were included in the purchase specifications to permit the purchaser to verify that these requirements were satisfied.

5.61 Identify those piping systems for which pipe ruptures were postulated and the resulting jet forces considered in the design of the containment vessel and vessel penetrations. Provide the combinations of loadings that included these jet impingement forces, and that were used in the design, and identify any associated stress limits that differed from those allowed by Table 5.2-1 in the FSAR.

5.62 Provide a more detailed description of the containment vacuum relief system (vacuum breaker system) that includes the following information:

5.62.1 A failure analysis of the active components of the system.

5.62.2 Developmental test evidence that the vacuum breaker system will function as stated.

5.62.3 Provisions for testing of the system during normal plant operation.

5.63 Provide the criteria that were employed in the design, fabrication, and nondestructive examination of containment penetration bellows expansion joints. Specify the code (or code case interpretations), the design loading combinations, the associated operating condition categories that apply (i.e., normal, upset, and emergency), and the corresponding design stress limits that were specified.

5.64 The only design criterion presented in the FSAR for the guard pipe portion of the containment hot penetration assemblies indicates that the pipe wall thickness was calculated on the basis of 1.5 times the process line design pressure using the rules of USAS B31.1.0 - 1967, Power Piping Code. However, the guard pipes that traverse the shield building annulus are not specifically within the jurisdiction of any component code currently in effect and consequently are not within the scope of code design rules. Therefore, in order to assess the adequacy of the guard pipes to sustain all normal and postulated accident conditions without failure, provide the design criteria that account for all loading combinations that may be encountered during both "normal" and "accident" conditions (e.g., comparable to the normal, upset, emergency and faulted operating condition categories defined in Appendix B of the FSAR). Compare the stress limits associated with each of the loading combinations with the maximum stresses calculated.

5.65 Describe the requirements that were specified for the fabrication and nondestrictive examination of the containment hot penetration guard pipes. Identify any component code rules that may have been applied.

5.66 Indicate whether an independent review of the design of the containment hot penetration guard pipes is being performed similar to that being performed for the Prairie Island application.

5.67 Indicate the margins provided for the expansion of the containment penetration assemblies, beyond those that are provided for temperature, pressure, and seismic effects, to accommodate relative movements between the shield building and the containment vessel.

5.68 Provide the criteria that were applied to preclude instability of the steel containment shell under loadings other than internal pressure. Specify the combination of loads considered (e.g., internal vacuum, local impact, seismic, annulus heatups), the magnitude of the loads, the stress and deformation limits applied, and the minimum margins provided against unstable deformation.

5.69 For the second overpressure test that will be performed with the internal concrete support system in place, indicate your intent to conduct the test inspection required by paragraph UG-100 of Division 1, Section VIII of the ASME B&PV Code. In what manner was paragraph N1314(d) of Section III of the Code considered in developing the test requirements.

5.70 It appears that no means will be provided for continuously checking the weld seams of the containment vessel for leakage. Describe the method to be used for leak checking during normal operation and the frequency of these checks.

6.0 ENGINEERED SAFETY FEATURES

6.1 On page 6.1-3 of the FSAR, it is stated that the release of fission products from the containment is limited, in part, by the reduction in the fission product concentrations in the containment atmosphere that results from the borated water sprayed into the containment atmosphere. While no credit is claimed for the scrubbing action of the sprays, discuss the reduction in fission product concentrations that you would expect as a result of spray operation.

6.2 On page 6.2-7 of the FSAR, it is stated that one of the two boric acid tanks is the initial source of concentrated boric acid solution for the safety injection pumps. Provide the following additional information with respect to this part of the system:

6.2.1 Which tank is aligned to the safety injection pump suction?

6.2.2 If either tank may be used, how is the selection made?

6.2.3 It appears possible for the suction of the safety injection pumps to be aligned to closed valves from the boric acid tanks, in which case neither the fluid from the boric acid tanks nor from the refueling water storage tank would be available for injection into the reactor vessel by the high head injection system. What design provisions assure that at least one of the valves (8815A or 8815B) is open and ready to discharge when the redundant parallel valves to the suction of the safety injection pumps open? Is the status of valves 8815A and 8815B displayed in the control room?

6.3 In the description of the sequence for change-over from injection to recirculation given on page 6.2-11 of the FSAR, it is stated that upon receipt of the first low level alarm from the refueling water storage tank (RWST), the operator takes action to assure that sufficient NPSH exists for the operating pumps to run until the tank is nearly empty. Explain what is meant by this statement. Available NPSH at a given level in the refueling water storage tank should be independent of any operator action.

6.4 It is stated on page 6.2-13 of the FSAR that all motors, instruments, transmitters, and associated cables that are located inside the containment and required to operate following an accident are designed to function under the appropriate post-accident temperature, pressure, and humidity conditions. Identify all such equipment and describe the qualification tests that were performed on each of these items to demonstrate that the equipment will perform as designed.

6.5 Discuss the design of the control circuits for the motor-operated isolation valves (8800A and 8800B) between the accumulator tanks and the primary coolant system, indicating the assurance provided by the design that the valves will be open when required. The inclusion of the following features would provide an acceptable design:

(1) Valve position visual indication that is actuated by sensors on the valve ("open" and "closed").

(2) An audible alarm, independent of item (1), which is actuated by a sensor on the valve when the valve is not in the fully open position.

(3) A lock-out of power to the valve operator at any time the primary coolant pressure exceeds a preselected valve (specified in the Technical Specifications).

As an alternative, consideration could be given to the incorporation of the following features in lieu of item (3).

(4) Automatic opening of the valves when the primary system pressure exceeds a preselected value (specified in the Technical Specifications).

(5) A safety injection signal used to automatically remove (override) any bypass feature that may be provided to allow a motor-operated valve to be closed, for short periods of time, when the primary system is at pressure (in accordance with the provisions of the Technical Specifications).

6.6 Describe the interlocks that are provided to prevent opening the containment sump recirculation line valves and thereby prevent dumping the RWST water directly to the sump.

6.7 The practice of permitting small diameter piping for essential systems to be "field-run" should be limited insofar as it is practical to do so. When it is permitted (1) stringent quality assurance measures should be taken to assure that the installation has been performed in such a manner that the assumptions made for design and safety assessment purposes remain valid, and (2) tests should be performed on the completed item to provide a final indication of acceptability. In view of these requirements, provide the following information:

6.7.1 A discussion of the extent to which you permitted "field-running" of small diameter piping for essential systems, including all engineered safety features.

6.7.2 A description of the special quality assurance measures and performance tests that were conducted to assure satisfactory installations.

6.8 On page 6.2-14 in the FSAR, reference is made to three boric acid tanks. Only two boric acid tanks are specified on page 6.2-7 and in Table 9.2-3. Examine this discrepancy and correct the FSAR as necessary. Also, on page 6.2-15, the reference to the boric acid tank design parameters should be Table 9.2-3 instead of Table 9.2-2.

6.9 Provide the following data in chart form for both the safety injection pumps and the residual heat removal pumps:

6.9.1 The required NPSH for the range of flow conditions including run-out flow.

6.9.2 The minimum available NPSH during the injection phase of operation.

6.9.3 The minimum available NPSH during the recirculation phase of operation.

6.10 Provide the results of appropriate analyses to demonstrate that the piping and vital components of safety related systems, including the core cooling systems, will be capable of withstanding the potential hydraulic forces (e.g., water hammer and steam compression forces) that could result from system actuation with the pump discharge lines not completely filled with fluid or from rapid valve closure in an operating line. What special design features are provided to prevent the development of such forces?

6.11 Page 6.2-20 in the FSAR contains a statement that the high head safety injection piping is protected by a relief valve in the test line inside containment, relieving to the pressurizer relief tank. Figure 6.2-1 in the FSAR does not show this valve. Confirm that this valve is in fact provided and describe its location.

6.12 Provide a description of the test data that were used for the design of the headers and nozzles of the containment vessel internal spray system to assure uniform coverage of the containment volume.

6.13 Page 6.4-8 of the FSAR contains a statement that the containment spray piping has appropriate connections and valves for chemical addition for pH control. Provide the following additional information:

6.13.1 What is the range of pH values allowed for the injection water? For the recirculation water?

6.13.2 What chemical additives will be used to control the pH? How will the need for pH adjustment be determined? Specifically, what means are provided to make the chemical addition and what procedure will be followed?

6.13.3 What affect does varying the pH have on the hydrogen generation rates as analyzed in Section 14 of the FSAR?

6.13.4 Describe and summarize the results of your analyses of the materials of construction and protective coatings used in the containment and the corrosion of these materials by the borated spray water.

6.14 Describe the routing of the supply piping to the containment internal spray headers. Is this piping inside or outside the containment shell? Is it protected from possible missiles generated within containment?

6.15 Discuss the inservice inspection program for fluid systems other than the reactor coolant pressure boundary, including items to be inspected, accessibility requirements, and the frequency and types of inspection. The fluid systems to be considered are the engineered safety features, reactor shutdown systems, cooling water systems, and the radioactive waste treatment systems.

6.16 For electrical and mechanical equipment of the reactor protection system and engineered safety features located in the primary containment or elsewhere in the plant, describe the design criteria that were established in order to limit to acceptable levels the potential effects of radiation on these components resulting from both normal operation and accident conditions super-imposed on long-term normal operation. List the items or systems for which these criteria were applied and describe the analysis and testing performed to verify compliance with the criteria.

6.17 Identify all safety related equipment and components (e.g., motors, cables, filters, pump seals) located in the primary containment or elsewhere in the plant that are required to be operable during and subsequent to a loss-of-coolant accident or a steamline break accident. Describe the qualification tests that were performed on each of these items to assure their performance in a combined high temperature, pressure, and humidity environment for the time period required. Your response should consider the susceptibility of redundant engineered safety features and instrumentation located outside containment to the environment resulting from a single pipe failure or major leak in the systems that circulate fluids outside the containment during post-accident conditions.

9.0 AUXILIARY AND EMERGENCY SYSTEMS

9.1 Describe in greater detail, the boron analyzer mentioned on page 9.2-15 in the FSAR. What is the accuracy of the instrument and how is it calibrated? What means are provided to periodically check the operation of the analyzer? What would be the effect of maloperation of the analyzer?

9.2 Provide additional data regarding the speed control of the charging pumps. Do the pumps receive signals from more than one pressurizer level controller? If not, what assurance is there that a malfunctioning controller could not result in overpressurization of the system?

9.3 If the floor and sides of the refueling cavity are not lined with stainless steel, specify the lining material and describe its corrosion resistance.

9.4 Safety Guide No. 13, published March 10, 1971, addresses the special considerations involved with the fuel storage pool. Since this Guide was published after the FSAR, discuss the adequacy of your design in light of the desired features as detailed in the Guide, and indicate what changes, if any, you intend to make to your design.

10.0 STEAM AND POWER CONVERSION SYSTEMS

10.1 Provide a drawing showing an elevation view of the circulating water intake and discharge structures, including the screenhouse and intake crib.

10.2 On page 2.6-4 of the FSAR a statement is made that, "The Kewaunee plant design incorporates features to insure a continuous supply of cooling water." Figure 10.2-7 in the FSAR indicates an auxiliary water inlet in the circulating water intake line. Provide a discussion, supported with drawings or sketches, as appropriate, that includes the following information:

10.2.1 A description of the design bases of the circulating water system and the method of operation under both normal and accident conditions.

10.2.2 A description of the auxiliary water inlet and its ability to perform at the probable maximum and probable minimum water levels in the lake, under conditions where the intake crib is blocked to flow.

10.2.3 An overall evaluation of the ability of the system to deliver an assured supply of cooling water under all conditions of plant operation, and in the event of design basis natural phenomena and postulated lake traffic accidents.

10.3 Describe and provide a summary of the results of tests that have been performed to demonstrate that the main steam line isolation valves and the non-return valves will function in accordance with design.

10.4 Provide a discussion of the possible dynamic loads on the system caused by rapid steam line isolation valve action. What design provisions have been incorporated to accomodate these dynamic loads?

11.0 WASTE DISPOSAL AND RADIATION PROTECTION SYSTEM

11.1 Recent operating experiences of plants similar to Kewaunee have revealed a degraded performance of the liquid waste processing systems as compared to performance levels anticipated in the design. Waste evaporator throughput has been on the order of 75% of the design value while the maximum attainable decontamination factors have been about 10^3. In combination with higher than anticipated plant leak rates, this degraded performance of the evaporator train has led to operating problems for the plants and to higher than anticipated levels of radioactivity discharge. Describe the measures you have taken or intend to take to preclude similar problems at Kewaunee.

11.2 Provide a description of the proposed manner of processing laundry and hot shower wastes that have too high a radioactivity content to permit direct discharge to the sewage system. Other plants have experienced considerable difficulty in processing these wastes because of the foaming action of the detergents in the waste liquid.

11.3 A reference is made on page 11.1-8 in the FSAR to six gas decay tanks while page 11.1-11 and Table 11.1-3 specify only four such tanks. Correct as necessary. Also, correct Table 11.2-5, which specifies a "minimum" dose instead of what should be a "maximum" dose adjacent to the spent fuel pool.

11.4 Provide a tabulation of the total anticipated maximum yearly discharge of each isotope (including tritium) expected to be released from all sources (provide separate tabulation for the releases that will result from containment purging) to the environment for both liquid and gaseous effluents and compare them with the limits of 10 CFR Part 20. Assume water leakage from the reactor coolant system and fission product leakage from the fuel rods consistent with your proposed Technical Specifications and with data from operating plants, such as Ginna. In addition, provide the following:

11.4.1 Describe the bases for these estimates, including the assumptions made with regard to decontamination factors used for holdup, filtration, evaporation and demineralization.

11.4.2 Provide any test or operational data from similar radwaste treatment components in operating plants.

11.4.3 Include a description using numerical values to show the anticipated use of the containment vent and purge system to relieve pressure buildup in the containment, and to permit access to the containment by plant personnel for routine inspection and maintenance.

11.5 Provide additional information in the form of process data on process flow diagrams for both liquid and gaseous wastes. On the liquid waste process flow diagram show the microcurie content and flow rate in each flow path. On the gaseous waste process flow diagram show flow rates, temperatures, pressures, specific isotope radioactivity, and minimum holdup time in decay tanks, at appropriate steps of the process.

11.6 Provide a description of your procedures to determine and record the activity of specific isotopes that will be discharged as waste to the lake. Include in the description the frequency of periodic determinations of isotopic composition and changes in conditions that would require these determinations (such as changes in the radwaste process or unexplained changes in gross activity measurements).

11.7 Provide the following information regarding tritium:

11.7.1 A comparison of the anticipated maximum tritium concentration in the circulating water discharge and in Lake Michigan, with the current tritium concentration in the lake as determined by your preoperational monitoring program. Provide also your analysis of the fraction of tritium in the primary coolant that will be released to the circulating water discharge volume.

11.7.2 A discussion of the uncertainties in your estimates of the amounts of tritium that will be generated.

11.7.3 A description of the methods to be used to monitor tritium concentrations prior to discharge from the plant.

11.8 Provide the following information with respect to the instrumentation, sampling techniques, and laboratory analytical procedures to be used for evaluation of gaseous and liquid effluents and other in-plant and environmental radiation and radioactivity levels.

11.8.1 A description of instrument types, sensitivities, ranges, setpoints, and calibration methods and frequencies.

11.8.2 A discussion to demonstrate the capabilities of the instrumentation and procedures to detect, measure, and control effluent releases, by appropriate radionuclide, within design objectives and expected levels for routine operation and for expected operational occurrences.

11.8.3 An identification of each path by which liquid and gaseous effluents can be released from the plant, and of the instrumentation to be used for monitoring each path.

11.8.4 An evaluation of possible annual releases of radioactive materials if effluents having radioactivity levels equivalent to the monitor set point are continuously released.

11.8.5 The bases for the requirements presented in Section 3.9 of the proposed Technical Specifications and a discussion of how the monitoring system will be used to demonstrate conformance to the specifications.

11.9 Provide a description of radiation monitoring instruments, including sensitivities, that can be used to determine the effectiveness of the HEPA and charcoal filters in engineered safety features following a design basis accident. Provide a discussion of the feasibility of including an iodine monitor and features to automatically isolate one of the filter trains if the charcoal filter fails to perform its intended function.

11.10 Page 11.2-13 of the FSAR describes the containment purge vent-air particulate monitor and page 11.2-14 describes the containment purge vent radioactivity gas monitor. Provide a description of the capability of the instruments to detect specific isotopes, including the capability to detect gaseous iodine 131 in the presence of noble gases. Provide a description of the method of instrument calibration for measuring the quantity of radioactive noble gases, iodines and particulate matter. State the filter efficiency of the paper (tape) for radioactive iodine.

11.11 Page 11.1-8 of the FSAR describes the processing of water and steam discharged to a flash tank from the steam generator blowdown system. Provide an analysis of the maximum concentrations of radioactivity that could exist (1) in the discharge canal due to the continuous release of water from the flash tank having a radioactivity level equal to the setpoint of the liquid radiation monitor and (2) in the stack due to the continuous release of steam and gases from the flash tank having a radioactivity level equal to the setpoint of the stack gas monitor. Assume for the analysis that the maximum reactor coolant activity and the maximum reactor coolant leakage to the steam generator are as specified in Section 3.1 of the Technical Specifications.

11.12 On page 11.2-10 of the FSAR, a statement is made that the radiation monitoring detectors inside containment would not be expected to operate following a LOCA. Provide the basis for this design and describe the means that are provided to monitor the post-LOCA radiation levels in the containment.