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Docket No.: 50-397

Mr. Neil O. Strand
Washington Public Power Supply System
300 George Washington Way
P.O. Box 968
Richland, Washington 99352

Dear Mr. Strand:

SUBJECT: FIRST ROUND QUESTIONS ON THE WNP-2 OL APPLICATION - PLANT SYSTEMS, PSB, AB, SEB, OLB

In our review of your application for an operating license for the WNP-2 facility, we have identified a need for additional information which we require to complete our review. The specific requests are contained in the enclosure to this letter and are the sixth set of our round one questions; additional requests related to other portions of the WNP-2 facility will be sent during this month. In order to maintain our present schedule, we need a completely adequate response to all questions in the enclosure by June 8, 1979. Please note that the round one questions from the Analysis Branch are substantially similar to those on the Susquehanna docket.

The attached set of round one questions represent the review effort of Plant Systems, the Power Systems Branch, the Analysis Branch, the Structural Engineering Branch and the Operator Licensing Branch. Where we have been able to formulate statements of staff positions (RSP), we have included these in our questions.

Please contact us if you require any discussion or clarification of the enclosed requests.

Sincerely,

Steven A. Varga, Chief
Light Water Reactors Branch No. 4
Division of Project Management

7904190383

MR 4
GD

Enclosure:
As stated

cc: See attached Sheet

OFFICE	LWR#4:DPM	LWR#4:DPM				
SURNAME	DLynch/bm	SAVarga				
DATE	3/16/79	3/16/79				

[illegible]

Washington Public Power Supply System

ccs:
Joseph B. Knotts, Jr., Esq.
Debevoise & Liberman
700 Shoreham Building
806 Fifteenth Street, N. W.
Washington, D. C. 20005

Richard Q. Quigley, Esq.
Washington Public Power Supply System
3000 George Washington Way
P. O. Box 968
Richland, Washington 99352

Nepom & Rose
Suite 101 Kellogg Building
1935 S. E. Washington
Milwaukie, Oregon 97222

Ms. Susan M. Garrett
7325 S. E. Steele Street
Portland, Oregon 94206

Mr. Creg Darby
807 So. Fourth Avenue
Pasco, Washington 99301

Nicholas Lewis, Chairman
Energy Facility Site Evaluation Council
820 East Fifth Avenue
Olympia, Washington 98504

Mr. O. K. Earle
Licensing Engineer
P. O. Box 968
Richland, Washington 99352

ENCLOSURE

STATEMENT OF REGULATORY STAFF POSITIONS

AND

REQUEST FOR ADDITIONAL INFORMATION

WPPSS NUCLEAR PLANT NO. 2

DOCKET NO. 50-397

005.0 PLANT SYSTEMS

005.1 In order that we might evaluate your compliance with the Codes and Standards Rule, Section 50.55a of 10 CFR Part 50, identify the edition of Section III of the ASME Boiler and Pressure Vessel Code and the applicable code addenda for the following Quality Group A components within the reactor coolant pressure boundary identified in Table 3.2-1 of the FSAR. These components are: (1) the reactor pressure vessel; (2) the main steam and feedwater piping inboard of the outermost isolation valves; (3) the piping of other interconnecting systems inboard of the outermost isolation valves; (4) the main steam isolation valves; (5) the explosive valves of the standby liquid control systems; and (6) the system or isolation valves of other interconnecting systems.

040.0 POWER SYSTEMS BRANCH

040.34 Provide a list of the following items, by voltage class, for the electrical penetrations in the containment: (1) the I^2t ratings; (2) the maximum predicted fault currents; (3) an identification of the maximizing faults; (4) the protective equipment setpoints; and (5) the expected clearing times.

040.35 Your description in Appendix C of the FSAR regarding the conformance of the electrical penetrations in the containment of the WNP-2 facility with the staff's positions in Regulatory Guide 1.63, Revision 2, "Electrical Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants," July 1978, does not provide sufficient information to allow an independent evaluation of your design. Demonstrate in detail how your design of these electrical penetrations is in compliance with the requirements of IEEE Standard 279-71.

040.36 RSP In addition to the undervoltage scheme currently provided to detect a loss of offsite power at the safety busses, we require the WNP-2 facility to have a second level of voltage protection, including a time delay, to protect the onsite power system from any adverse effects that could result from a sustained degraded voltage condition in the offsite power system. The design criteria for this second level of voltage protection are:

a. The selection of the voltage and time set points shall be determined by an analysis of the voltage requirements of the safety-related loads at all onsite system distribution levels.

b. The voltage protection shall incorporate coincidence logic to preclude spurious trips of the offsite power source.

c. The time delay which is selected shall be based on the following considerations: (1) the allowable time delay, including a conservative margin, shall not exceed the maximum time delay that is assumed in the appropriate accident analyses in Section 15 of the FSAR; (2) the time delay shall minimize the effect of short duration disturbances which might reduce the availability of the offsite power source(s); and (3) the allowable time duration of a degraded voltage condition at all distribution system levels shall not result in failure of safety-related systems or components.

d. The voltage sensors shall automatically initiate the disconnection of offsite power sources whenever the voltage set point and time delay limits have been exceeded.

e. The voltage sensors shall be designed to satisfy the following requirements: (1) the equipment will be Class IE and will be physically located at, and electrically connected to, the emergency switchgear; (2) independent undervoltage protection will be provided for each division of emergency power; (3) the equipment will have the capability to be tested and calibrated during power operation; and (4) annunciation must be provided in the control room for any bypasses incorporated into the design.

f. The Technical Specifications for the WNP-2 facility, will include: (1) the limiting conditions for operation; (2) the surveillance requirements; (3) the trip set points, including their minimum and maximum limits; and (4) the allowable values of voltage and time delay for the second level voltage protection sensors and its associated time delay devices (i.e., the delayed trip).

040.37 RSP (040.02) We require that the load-shedding function of the diesel-generator bus automatically prevent load shedding of the emergency bus due to a degraded voltage once the diesel-generator is supplying power to the emergency bus. Moreover, the design shall also include the capability of the load shedding function described above to be automatically reinstated if the diesel-generator supply breaker is tripped. Your proposed design of the load-shedding function of the diesel-generator bus is not clear from your response to Item 040.02 since your response merely references Sections 8.3.1.1.1, 8.3.1.1.8.1.7 and 8.3.1.1.8.2.7. Our review of these sections indicates that your answer is not responsive to the question. Accordingly, provide the details of your design and state your intent to comply with our position on this matter. Alternatively, provide justification for any exceptions taken to our position.

We further require that the Technical Specifications for the WNP-2 facility include a test requirement to demonstrate, during shutdown, the full functional operability of the bypass and reinstatement feature at least once in an 18 month period. Proper operation shall be determined by verifying that after an interruption of the onsite power supply, the loads are shed from the emergency busses in accordance with the design requirements and that subsequent loading of the onsite sources is through the load sequencer.

040.38 RSP It is our position that the voltage levels at the safety-related busses be optimized for both the full bus load and the minimum bus load conditions that are expected throughout the anticipated range of voltage variations of the offsite power source. This optimization is to be achieved by an appropriate adjustment of the voltage tap settings of the intervening transformers. Accordingly, we require that you verify the adequacy of this design feature by actual measurements. Provide a description of your test procedure for making this verification. Indicate how the results of this verification program will be compared with the voltages discussed in Section 8.3.1.2.4 of the FSAR. (This Section was added in response to Item 040.01 of our request for additional information.)

040.39 Provide a description of the physical arrangement which connects the field cables inside containment to the containment penetrations (e.g., connectors, splices, or terminal blocks). Indicate how you will provide assurance that these physical interfaces are qualified to withstand a postulated loss-of-coolant accident (LOCA) or the environment resulting from a postulated steam line break.

040.40 Provide a listing of all motor-operated valves in the WNP-2 facility which require a power lock-out to meet the single failure criterion. Indicate in detail how you satisfy this requirement.

040.41 Discuss the capability of the battery chargers of the emergency power system to properly function and remain stable upon the disconnection of the batteries, including a discussion of any anticipated modes of operation which would require a disconnection of the batteries (e.g., when applying an equalizing charge).

040.42 Indicate in detail how the design of the d-c power system provides assurance that safety-related equipment will be protected from damaging overvoltages caused by the battery chargers either as a result of faulty voltage regulation or operator error.

040.43 Review the electrical control circuits for all safety-related equipment to provide assurance that disabling of one component does not, through incorporation in other inter-locking or sequencing controls, render other components inoperable. All modes of test, operation, and failure should be considered. Describe and state the results of your review.

040.44 During our review of the Hatch 2 application for an operating license, we identified certain potential problems that could be caused by the motor-generator sets of the reactor protection system (RPS). These problems were related to the operating characteristics of these motor-generator sets which might exceed the envelopes of acceptable values of voltage and frequency, thereby adversely affecting the connected loads. Indicate whether the motor-generator sets in the WNP-2 RPS are similar to those in the Hatch 2 facility. If they are, provide: (1) a commitment to the generic resolution of this item;* or (2) justification for the use of these motor-generator sets in the WNP-2 facility.

040.45 (9.5.2) The information in Section 9.5.2 of the FSAR regarding the on-site communications system does not adequately discuss the capabilities of this system during anticipated transients and postulated accidents. Accordingly, provide the following information:

a. Identify all working stations on the plant site where it may be necessary for plant personnel to communicate with either the control room or the emergency shutdown panel during and/or following transients and/or accidents, including fires, to mitigate the consequences of the event and to achieve a safe cold shutdown of the WNP-2 facility.

b. Indicate the types of communication systems available at each of the working stations identified in Item (a) above.

c. Indicate the maximum sound levels that could exist at each of the working stations identified in Item (a) for all transients and accident conditions.

d. Indicate the maximum background noise level which would not adversely affect communication with the control room using: (1) the page party communications systems; and (2) any other additional communication system provided for that working station.

*Refer to the attached letter dated February 23, 1979, RS Boyd to GG Sherwood, "Reactor Protection System Power Supply Protective Circuitry."

e. Describe the performance requirements and acceptance tests for the communication systems at the onsite working stations identified in Item (a) to provide assurance that effective communication with either the control room or the 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, including the considerations given to: (1) component failures; (2) a loss of power; and (3) the severing of a communication line or trunk as a result of an accident or fire.

040.46 (9.5.3) Identify the vital areas and the hazardous (e.g., high radiation) areas where emergency lighting is needed for safe shutdown of the reactor and the evacuation of personnel in the event of an accident, including fires. Provide a tabulation of these emergency lighting systems in the WNP-2 facility.

040.47 (9.5.4) In Section 9.5.4.1 of the FSAR, you do not specifically reference ANSI Standard N195, "Fuel Oil Systems for Standby Diesel Generators," for the emergency diesel engine fuel oil storage and transfer system. Indicate whether the design of this system complies with the cited standard. If not, provide justification for non-compliance. (Refer to paragraph II.12 of Section 9.5.4, Revision 1, of the Standard Review Plan (SRP), NUREG-75/087.)

040.48 RSP (9.5.4) In Figure 9.5-4 of the FSAR, you identify the ½ inch minimum flow line, DO(6)-1, as non-safety class G piping. This is unacceptable. Accordingly, we require this line to be designed to seismic Category I criteria and satisfy the requirements for safety Class 3. Revise your design accordingly.

040.49 (9.5.4) In Section 9.5.4.3 of the FSAR, you state that diesel fuel oil is available from local sources. Identify the sources where diesel fuel oil of acceptable quality will be available and the distances from these sources to the WNP-2 facility. Discuss how fuel oil will be delivered onsite under extremely unfavorable environmental conditions. (Refer to Paragraph III.5.b of Section 9.5.4, Revision 1, of the SRP.)

040.50 (9.5.4) In Section 9.5.4.3 of the FSAR, you state that the materials selected for the diesel fuel oil system assure adequate corrosion protection, thereby minimizing fuel oil contamination.

We find this statement to be too general in nature to be meaningful. Accordingly, revise the FSAR to provide a more explicit description of the protection provided for underground piping. If coatings which provide protection against corrosion are being considered for piping and tanks, identify the standards which will govern their application. Discuss the provisions to provide an impressed current type of cathodic protection system, in addition to water-proof protective coatings, for the WNP-2 fuel oil storage and transfer system. The purpose of this cathodic protection is to minimize corrosion of buried piping or equipment. If cathodic protection is not incorporated into the WNP-2 facility, provide justification for this ommission. (Refer to Paragraph III.4 of Section 9.5.4, Revision 1, of the SRP.)

040.51 (9.5.4) In Section 9.5.4.4 of the FSAR, you describe the testing and inspection procedures for the diesel fuel oil system. Provide the specific standards which will be followed to assure a reliable supply of fuel oil to the emergency diesel-generators. (Refer to Paragraphs III.3 and III.4 of Section 9.5.4, Revision 1, of the SRP.)

040.52 (9.5.4) In Section 9.5.4.2 of the FSAR, you state that the diesel-generator fuel oil storage tank is provided with an individual fill and vent line. Indicate whether these lines are located indoors or outdoors and state the height above finished grade at which these lines are terminated. If these lines are located outdoors, discuss the provisions made in your design to prevent the entrance of water or dust into the storage tank during adverse environmental conditions.

040.53 (9.5.4) Discuss how you will detect or prevent the growth of algae in the diesel fuel storage tanks. If algae were detected in these storage tanks, describe the method(s) you propose to clean the affected storage tanks. (Refer to Paragraph III.4 of Section 9.5.4, Revision 1, of the SRP.)

040.54 (9.5.5) Describe how the design of the WNP-2 diesel engine cooling water system provides assurance that all components and piping are filled with water. (Refer to Paragraph III.2 of Section 9.5.5, Revision 1, of the SRP.)

040.55 (9.5.5) In Section 8.3.1.1.8.1.11 of the FSAR, you state that the WNP-2 diesel-generators are automatically started and then run without any electrical loads following a postulated LOCA if offsite power is available at the 4.16 kV Class 1E busses. Discuss how long the Division 1 and 2 diesel generators can run unloaded without a degradation of the diesel engine perfor-

mance or without a loss of reliability. Provide a similar discussion for the Division 3 diesel-generator which supplies power to the high pressure core spray system.

040.56 (9.5.5) In Section 9.5.5.2 of the FSAR, you state that each diesel engine cooling water system is provided with an expansion tank to provide for system expansion and for venting air from the system. In addition, the expansion tank is intended to: (1) provide for minor system leaks at pump shaft seals, valve stems and other components; and (2) maintain the required net positive suction head (NPSH) at the cooling water system circulating pump. Indicate the size of the expansion tank and state its location. Demonstrate by analysis that the size of the expansion tank is adequate to: (1) maintain the required NPSH at the pump; and (2) provide make-up water for seven days of continuous operation of the diesel engine at its full rated load without the addition of water into the expansion tank. Alternatively, provide a seismic Category I, safety class 3 make-up water supply for the expansion tank.

040.57 (9.5.6) Describe the instrumentation, controls, sensors and alarms in the starting air system which alert the operator when the design parameters of this system are exceeded. Discuss the actions of the operator if this system annunciates an alarm in the control room. Our concern is whether there is sufficient time for the operator to take appropriate action.

040.58 (9.5.8) In Figure 9.5-5 of the FSAR, you show the location of the diesel engine air intake louvers. Discuss the effect of adverse meteorological conditions (i.e., heavy or freezing rain, ice, snow or a severe dust storm) on the availability of the emergency diesel-generators. (Refer to Paragraph III.5 of Section 9.5.8, Revision 1, of the SRP.)

040.59 (9.5.8) Describe the instrumentation, controls, sensors and alarms of the diesel engine combustion air intake and exhaust system which alert the reactor operator when the design parameters of this system are exceeded. Discuss the actions of the operator if this system annunciates an alarm in the control room. As before, our concern is with the time available for an operator to take appropriate action. (Refer to Paragraphs II.1 and II.4 of Section 9.5.8, Revision 1, of the SRP.)

040.60 (10.2) Expand your discussion of the turbine speed control and overspeed protection system. Provide additional explanation of the turbine and generator electrical load following capability for the turbine speed control system with the aid of system schematics (including turbine control and extraction steam valves to the heaters).

Tabulate the individual speed control protection devices (normal emergency and backup), the design speed (or range of speed) at which each device begins operation to performs its protective function (in terms of percent of normal turbine operating speed). In order to evaluate the adequacy of the control and overspeed protection system provide schematics and include identifying numbers to valves and mechanisms (mechanical and electrical) on the schematics. Describe in detail, with references to the identifying numbers, the sequence of events in a turbine trip including response times, and show that the turbine stabilizes. Provide the results of a failure mode and effects analysis for the overspeed protection systems. Show that a single steam valve failure cannot disable the turbine overspeed trip from functioning. (SRP 10.2, Part III, items 1, 2, 3 and 4).

040.61 (10.2) Provide a discussion of the inservice inspection program for the throttle-stop valve, the control valve, the reheat stop valve and interceptor steam valve. Discuss the capability for testing of essential components during operation of the turbine-generator system. (Refer to Paragraph III.5 and III.6 of Section 10.2, Revision 1, of the SRP.)

040.62 (10.2) Provide a complete list of turbine-generator protective trips. Separate these trips into two categories: (1) those that will trip the turbine due to mechanical faults; and (2) those that will trip the turbine due to electric faults in the generator.

040.63 (10.2) Provide in Section 10.2.5.6 of the FSAR, information regarding the turbine-generator automatic load-following function.

040.64 (10.4.1) Discuss the means taken to prevent galvanic corrosion of the condenser tubes and other components. (Refer to Paragraph III.1 of Section 10.4.1, Revision 1, of the SRP.)

040.65 (10.4.1) Discuss the effect of any degradation of the main condenser (i.e., leakage or a partial loss of vacuum) on reactor operation. (Refer to Paragraph III.1 of Section 10.4.1, Revision 1, of the SRP.)

040.66 (10.4.1) Discuss the possible mechanisms for either hydrogen production in the exhaust steam side of the condenser or hydrogen carry-over from the reactor. Indicate the estimated rate of hydrogen accumulation in standard cubic feet per minute. Discuss how a buildup of hydrogen is prevented in the WNP-2 main condenser. (Refer to Paragraph III.1 of Section 10.4.1, Revision 1, of the SRP.)

040.67 (10.4.1) Discuss the means for detecting, controlling and correcting leakage of the cooling water in the condenser tubes into the condensate. (Refer to Paragraph III.2.a at Section 10.4.1, Revision 1, of the SRP.)

040.68 (10.4.1) Provide the permissible cooling water inleakage and time of operation with inleakage to assure that the condensate/feedwater quality can be maintained within safe limits. (Refer to Paragraph III.2.b of Section 10.4.1, Revision 1, of the SRP.)

040.69 (10.4.1) Indicate and describe the means of detecting radioactive leakage into and out of the main condenser. Indicate what provisions have been incorporated into the WNP-2 facility to preclude unacceptable accidental release of radioactivity to the environment. (Refer to Paragraph III.2.c of Section 10.4.1, Revision 1, of the SRP.)

040.70 (10.4.1) Discuss the operation of the main steam line isolation valves if there is a loss of condenser vacuum. (Refer to Paragraph III.3b of Section 10.4.1, Revision 1, of the SRP.)

040.71 (10.4.1) Indicate what design provisions have been incorporated into the WNP-2 facility to preclude failures of either the condenser tubes or other components resulting from: (1) a turbine by-pass blowdown; or (2) other high temperature drains into the condenser shell. (Refer to Paragraph III.3.c of Section 10.4.1, Revision 1, of the SRP.)

040.72 (10.4.1) In Section 10.4.1.4 of the FSAR, you discuss the tests and the initial field inspection of the main condenser. However, you do not indicate the frequency and extent of inservice inspection of this component. Accordingly, provide this information. (Refer to the first paragraph of the Acceptance Criteria in Section 10.4.1, Revision 1, of the SRP.)

040.73 (10.4.4) Provide a discussion regarding the measures you have taken to provide assurance that a failure of a high energy line in the turbine by-pass system will not have an adverse effect on, or preclude operation of, the turbine speed controls or any safety-related components or systems located close to the turbine by-pass system. (Refer to Paragraph III.4 of Section 10.4.4, Revision 1, of the SRP.)

040.74 (10.4.4) In Section 10.4.4.4 of the FSAR, you discuss the tests and the initial field inspection of the turbine bypass system. However, you do not indicate the frequency and extent of inservice testing and inspection of this system. Accordingly, provide this information. (Refer to Paragraph II.3 of Section 10.4.4, Revision 1, of the SRP.)

ATTACHMENT 1 to SECTION 040

February 23, 1979

Dr. Glenn G. Sherwood, Manager
Safety and Licensing Operation
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Dear Dr. Sherwood:

SUBJECT: REACTOR PROTECTION SYSTEM POWER SUPPLY PROTECTIVE CIRCUITRY

Your letter dated October 31, 1978, subject as above, requested our approval of the conceptual design of the proposed modifications to the reactor protection system power supply protective circuitry described therein. These modifications were proposed in response to our concern about the capability of the Class IE reactor protection system and other Class IE systems and components powered by the reactor protection system power supplies to accommodate the effects of possible sustained abnormal voltage or frequency conditions from the non-Class IE reactor protection system power supplies. These abnormal conditions could be caused by possible though unlikely combinations of undetectable single failures or by the effects of earthquakes, and could result in damage to the Class IE systems and components powered by the reactor protection system power supplies with the attendant potential loss of capability to perform their intended safety functions.

The proposed modifications to the reactor protection system power supply protective circuitry would consist of the addition of two Class IE "protective packages" in series between each motor-generator set and its respective reactor protection system bus, and between the alternate power source and the reactor protection system buses. Each protective package would include a breaker and associated overvoltage, undervoltage and underfrequency relaying, and would meet the testibility requirements for Class IE equipment.

With the protective packages installed, any random undetectable or seismically-induced abnormal voltage or frequency conditions in the outputs of the two motor-generator sets or the alternate power supply would trip either one or both of the two protective packages installed between each power supply and its respective reactor protection system bus thereby producing a half scram on that channel and retaining full scram capability on the other channel. The proposed modifications would

provide fully-redundant Class IE protection for the Class IE systems and components powered by the reactor protection system power supplies and would thereby bring the overall reactor protection system design into full conformance with Criteria 2 and 21 of the General Design Criteria, Institute of Electrical and Electronic Engineers Standards IEEE-279 and IEEE-379, and the applicable provisions of our Standard Review Plan.

On the bases of its conformance to the aforementioned criteria, we conclude that the conceptual design of the proposed modifications to the reactor protection system power supply protective circuitry is acceptable contingent upon implementation in conformance with the applicable criteria for Class IE systems. We note that the installation of the two protective packages in series with the alternate power source will obviate the need for technical specification time constraints on plant operation while using the alternate power source to supply power to one of the reactor protection system buses.

Sincerely,
Original Signed by
Roger S. Boyd

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

cc: Mr. Larry Gifford
General Electric Company
4720 Montgomery Lane
Bethesda, Maryland 20014

130.0 STRUCTURAL ENGINEERING BRANCH

130.10 (3.3.2) Your response to Item 130.1 is not satisfactory. The tornado load listed in Tables 3.8-15 and 3.8-16 of the FSAR is the combined effect of three separate loads generated by the Design Basis Tornado. Accordingly, describe your procedures to combine the tornado wind load, the tornado differential pressure load and the tornado missile load. Provide your procedures to evaluate the effective loads resulting from tornado generated missiles.

130.11 (3.3.2) (3.8.2) (372.8) Airborne dust has been identified as a possible design basis meteorological parameter in Item 372.8 of our request for additional information. Describe your procedures to evaluate airborne dust loadings. Indicate whether an airborne dust load was included in any of the loads and load combinations defined in Section 3.8.2.3 of the FSAR.

130.12 (3.5.3) Indicate the formula you used to determine the penetration depth of missiles into steel barriers.

130.13 RSP (3.7.1) You state in Section 3.7.1.1 of the FSAR that a response spectrum dynamic modal analysis was performed on the reactor building structure for a Safe Shutdown Earthquake (SSE) using: (1) the design response spectra defined in Figure 3.7-1 and the damping values of Table 3.7-1 of the FSAR; and (2) the design response spectra scaled to a maximum horizontal ground acceleration of 0.25g using the damping values defined in Regulatory Guides 1.60, Revision 1, and 1.61, Revision 0, respectively. Compare the horizontal and vertical floor response spectra in the reactor building at the locations specified below using your two design criteria cited above. Additionally, provide the same comparisons for structural responses (i.e., the seismic shear, the moments, the deflections, and the floor response spectra) in the horizontal and vertical directions for all other seismic Category I structures. This comparison of the floor response spectra should be performed assuming damping values of two percent and five percent of the critical damping value, at the operating floor, the reactor stabilizer level, the reactor vessel support, the divider barrier, the base mat and the refueling hatch level for the reactor building and at the basemat, an intermediate elevation, and an upper elevation for all other seismic Category I structures.

130.14 (3.7.1) Indicate whether and how a base line correction was made for the design time history shown in Figure 3.7-5 of the FSAR.

130.15 (3.7.1) You state in the first footnote to Table 3.7-1 of the FSAR that: "Damping values tabulated in Regulatory Guide 1.61, Revision 0, are utilized for designs which include the ground response spectra in accordance with Regulatory Guide 1.60, Revision 1, either independently, or in conjunction with LOCA and S/R valve discharge related hydrodynamic loads." However, this statement contradicts your statements in Sections 3.7.1.1 and 3.7.1.3 which indicate that Regulatory Guide 1.61 was not used. Remove this ambiguity. Indicate the damping values you assumed for soil.

130.16 (3.7.2) Indicate in Section 3.7.2.1.7 of the FSAR how the stresses due to relative displacements of supports are combined with other stresses. Indicate in Section 3.7.2.1.8.3, those cases where the relative displacements were insignificant and thus neglected in your analysis. Revise any other sections of the FSAR affected by your response to this item.

130.17 RSP (3.7.2) You state in Sections 3.7.2.1.8.1 and 3.7.2.1.8.2 of the FSAR that only one horizontal component and the vertical component of seismic responses are combined in determining the maximum stresses. (Other sections of the FSAR also discuss combining only two components of a seismic response; e.g., Section 3.7.2.6.) It is our position that the maximum stresses be determined by combining three orthogonal seismic components using the square-root-of-the-sum-of-the-squares (SRSS) methodology. Accordingly, provide justification for your approach. Alternatively, revise your analysis and the appropriate sections of the FSAR to reflect our position on this matter. (Refer to Section C.1.1 of Regulatory Guide 1.92, Revision 1, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," February 1976.)

130.18 (3.7.2) You state in Section 3.7.2.8.2 of the FSAR that when the piping system is anchored and supported at points with different excitations, the response spectrum analysis is performed using the response spectra at or above the center of mass of the piping system for the nuclear steam supply system (NSSS) and components. However, it is our position that for systems and components supported at different locations, an envelope response spectrum be used for the analysis. Accordingly, provide justification for your approach. Alternatively, revise your analysis to reflect our position on this matter.

130.19 (3.7.2) In Section 3.7.2.1.8.2 of the FSAR, you state that the load factors are derived by utilizing the response spectra for a conservatively chosen fundamental frequency based on the maximum span length of an assumed simply supported beam. Provide the details of this method.

130.20 (3.7.2) In Section 3.7.2.1.8.3.1 of the FSAR, you state that the stresses obtained for each natural mode are superimposed for all modal displacements of the structure using the SRSS methodology. Indicate whether there are any closely-spaced modes as defined by Equation 3.7.2.1-13. If so, indicate how the responses of the closely-spaced modes were combined with other modal responses in your calculations.

130.21 (3.7.2.3) Your response to Item 130.3 regarding the criteria used for decoupling subsystems is unclear. However, your criteria appear different from the acceptance criteria delineated in Paragraph II.3b of Section 3.7.2 of the Standard Review Plan (SRP), NUREG-75/087. Provide clarification and/or justification for your decoupling criteria.

130.22 (3.7.2) It is unclear how the values of the equivalent dynamic shear modulus, G, presented in Section 3.7.2.4 of the FSAR were determined. Accordingly, provide the pertinent information contained in Reference 3.7-6 for all seismic Category I structures.

130.23 (3.7.2) Identify in Section 3.7.2.5 of the FSAR, those cases where the peaks of floor response spectra were widened by less than 15 percent of the structural frequencies. (Refer to Paragraph II.9 of Section 3.7.2 of the Standard Review Plan.)

130.24 (3.7.2) For non-symmetric structures, there may be dynamic coupling between translational and torsional motions. As a result, calculating the torsional moment as the product of the inertial force and the distance between the centers of mass and rigidity may or may not be adequate. Accordingly, provide in Section 3.7.2.11 of the FSAR, the bases for your approach. Indicate how the torsional effects were included in the generation of the floor response spectra.

130.25 (3.7.2) In Section 3.7.2.12 of the FSAR, you state that Table 3.7-17 contains comparisons of the seismic responses obtained from the response spectrum approach and the time history approach. However, this comparison is not provided. Accordingly, provide this information.

130.26 (3.7.2) (130.6) Your response to Item 130.6 is not satisfactory. Accordingly, provide in Section 3.7.2.14 of the FSAR, the details of your procedures to calculate the capability of safety-related structures to resist overturning and sliding. Additionally, provide the factors of safety against overturning and sliding for each safety-related structure due to horizontal loads.

120.27 (3.7.2) In Section 3.7.2.15 of the FSAR, you state that the design values of the damping coefficients in Table 3.7-2 are significantly smaller than the calculated values. However, this information is not in this Table. Accordingly, provide this information and your bases for Equation 3.7.2.15-1.

130.28 (3.7.3) It is not apparent to us which of your references is the Reference 10 that you cite in Section 3.7.3.2.1 of the FSAR. Accordingly, provide justification for the number of earthquake cycles you estimate in this section.

130.29 (3.7.3) You present two conditions in Section 3.7.3.3 of the FSAR which must be satisfied in selecting the field locations of seismic supports and restraints. Elaborate on how these two objectives were met.

130.30 (3.7.3) The methods of analysis described in Sections 3.7.3.12.1.3 and 3.7.3.12.1.5 of the FSAR are unacceptable. The procedure for predicting the seismic stresses in buried pipes suggested by Newmark (Reference 3.7-12) is based on the assumption that pipes will move with the soil during an earthquake; i.e., there will be no slippage at the soil-pipe interface. You have not provided a basis for your proposed extension of these procedures to include consideration of friction and slippage (e.g., Equations 3.7.3.12.1.3-2 through 3.7.3.12.1.3-5 and Equation 3.7.3.12.1.5-3). Accordingly, it is our position that you use the procedures in Reference 3.7-12. Alternatively, provide either theoretical or experimental justifications for the equations cited above. Identify the values of "Vm" used in your analysis and explain how they were determined.

130.31 Indicate the clearance between the pipes and sleeves shown in Figures 3.8-48 and 3.8-49 of the FSAR and indicate the differential displacements of these buried pipes relative to the buildings. Provide the details of your procedures for estimating the stresses in those portions of the buried pipes connected to the buildings.

130.32 (3.8.2) Provide in Section 3.8.2.1 of the FSAR, the mechanical properties of the two inch thick, compressible, isolation material between the steel containment vessel and the reinforced concrete biological shield. Indicate if the properties of this compressible material could be affected by radiation.

130.33 (3.8.2) You state in Section 3.8.2.1 of the FSAR that the jet impingement force of 534 kips discussed in Section 3.8.2.3.5.2 might cause

local yielding of the drywell shell but that a plastic analysis in accordance with Section III of the ASME Code demonstrates that rupture of the drywell shell will not occur. You conclude on this basis that the jet impingement force will not adversely affect the leakage rate from the primary containment. Provide the details of the cited analysis and indicate the value of the calculated ductility ratio. Provide the basis for your conclusion that the leak tightness of the containment vessel will not be adversely affected.

130.34 (3.8.2) In Section 3.8.2.2 of the FSAR, you state that the structural steel attachments beyond the boundaries established for the primary containment steel vessel are designed and constructed according to AISC specifications for the design of structural steel buildings where applicable. You further state that you use allowable stress limits in accordance with Sub-Article NE-3131 (3) of Section III of the ASME Code. Identify these structural steel attachments and provide your basis for applying two different structural design codes for the design of the same structural elements.

130.35 (3.8.2) The load combinations considered in your analysis appear to be different than those delineated in Section 3.8.2 of the SRP. Provide justification for these apparent deviations and/or provide an assessment of the significance of the deviations with respect to the adequacy of the structural design. Indicate how a particular type of structural load (e.g., moments, shear, or stresses) resulting from a number of different loads, were combined (i.e., either by using the absolute sum or by the SRSS methodology).

130.36 (3.8.2) The computer programs AX1, AX2, and AX3 which you cite in Section 3.8.2.4.2 of the FSAR, are for analysis of axisymmetric structures. Explain how these programs incorporate the effect of the vertical and horizontal stiffeners which reinforce the primary containment vessel.

130.37 (3.8.2) Provide in Section 3.8.2.5 of the FSAR, a detailed presentation of your buckling criteria, including your procedures to establish these criteria, for the primary containment vessel.

130.38 (3.8.3) References 3.8-5, 3.8-6 and 3.8-7 of the FSAR have not been submitted for our review and acceptance. It is our position that the use of reports or references for definitions, criteria, and methods of analysis is unacceptable until the referenced documents are reviewed and accepted by us. Accordingly, submit the cited references for our review. Alternatively, refer to other documents which have been previously accepted by us.

130.39 (3.8.3) Table 3.8-10 of the FSAR presents the load combinations and load factors for internal structures made of reinforced concrete. However, it appears from this table that the abnormal/severe environmental loads were not considered. Similarly, it appears from our review of Table 3.8-11 that the extreme environmental, abnormal, and abnornmal/severe environmental loads were not considered for internal steel structures. Additionally, it appears that live loads were not considered in some of the load combinations in this latter table. Accordingly, provide justification for these ommissions and/or assess the significance of these ommissions with respect to the adequacy of the WNP-2 structural design.

130.40 (3.8.3) Though Section 3.8.3.4 of the FSAR indicates that it contains a description of the methods of analysis for internal structures, we cannot find any such description in this section. Accordingly, provide a discussion of your method of analysis for computing forces and displacements in all internal structures. If computer codes were used for this analysis, identify these codes. Provide or describe, whenever applicable, the models used in performing such analyses. Provide comparisons between the design values of forces and displacements and their allowable values based on: (1) the allowable stress limits; or (2) the capacity determined by the ultimate strength of these structures.

130.41 (3.8.3) (3.8.4) You indicate in Tables 3.8-13 and 3.8-17 of the FSAR that the plastic section modulus might be used in conjunction with the elastic working stress design method for the factored load combinations. Indicate whether you did this in your structural analysis. If so, provide justification for this approach.

130.42 (3.8.4) For both steel and reinforced concrete structures, the load combinations which you used are different from those presented in Section 3.8.4 of the SRP. Provide your bases for these deviations and/or provide an assessment of the significance of these deviations with respect to the adequacy of the WNP-2 structural design.

130.43 (3.8.5) Indicate whether you used Section III, Division 2 of the ASME Boiler and Pressure Vessel Code in your analysis and design of the reactor building foundation mat. If not, provide justification for this deviation and assess the significance of any deviation from this code with respect to the structural adequacy of the foundation mat of the WNP-2 reactor building.

130.44 (3.8.5) Indicate whether there is any uplifting (i.e., tilting from the horizontal) predicted for the foundation mats of all seismic Category I structures. If so, identify these structures and indicate the amount of the estimated uplifting.

220.0 ANALYSIS BRANCH

221.0 Reactor Analysis Section

221.01 Section 4.4 of the FSAR contains no discussion of "crud" buildup (i.e., deposits on the surfaces of the fuel elements) and its effect on both the critical power ratio (CPR) and the core pressure drop. Provide your assumptions regarding the sensitivity of the CPR and the core pressure drop due to variations in the amount of crud present. Provide data supporting your assumptions on crud buildup and discuss how crud buildup in the core would be detected.

221.02 (4.4.2) The data base for the approved correlation of the CPR contained in GEXL is for 7x7 fuel bundles and 8x8 fuel bundles having one water rod. However, you have not provided a substantial data base to support the use of the GEXL code to calculate the CPR of the 8x8, two water rod fuel bundles proposed for the WNP-2 facility. Accordingly, demonstrate that the GEXL correlation cited above is applicable to the fuel bundles proposed for the WNP-2 facility by comparison to applicable data. Our requirement in this matter must be satisfied prior to issuance of an operating license for the WNP-2 facility. Alternatively, you may increase the limiting value of the minimum critical power ratio (MCPR) by 0.05 to account for the uncertainty in using the GEXL code for the analysis of the 8x8, two water rod fuel elements.

221.03 (4.4.2) In Section 4.4.2.5 of the FSAR, you state that: "There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution of an operating reactor." Indicate whether this statement refers specifically to the WNP-2 calculations and identify the operating reactor which was used for the data comparison.

221.04 (4.4.2) Your discussion on the flow distribution in Section 4.4.2.5 of the FSAR does not address: (1) the effect of uncertainties on the flow distribution; or (2) the effect of channel flow uncertainty, coupled with the other uncertainties listed on Table 4.4-6, on the MCPR uncertainty. Additionally, Table 4.4-6 does not address the flow distribution uncertainties. Accordingly, provide this information.

221.05 (4.4.4.5) In Section 4.4.4.5.2 of the FSAR, you state that "...analytical models of the individual flow paths were developed as an independent check of the tests... When

using these models for hydraulic design calculations..." Provide the the equations comprising the model, including your assumptions. Provide a comparison of the results of this model with physical data.

221.06 (4.4.4) Indicate what fraction of the fuel bundle coolant flow is water-rod flow.

221.07 (4.4.4) In Section 4.4.4.5.1 of the FSAR, you state that "...the nominal expected bypass flow fraction is approximately 10%." Indicate the fraction of calculated bypass flow for the WNP-2 core, including the uncertainty of this value.

221.08 (4.4.4) Identify the name of the computer program cited in Section 4.4.4.5. Provide references which document the code.

221.09 (4.4.4) In Section 4.4.4.6.7 of the FSAR, you state that the most limiting condition for thermal-hydraulic stability occurs at the end of core life with power peaking toward the bottom of the core. Indicate whether the typical values of core stability provided in this section are based on the core characteristics at the end of core life. If not, provide values of the decay ratio for this condition. Provide the power profile and the void reactivity coefficients used for this analysis.

221.10 (4.4.4) In Section 4.4.4.6 of the FSAR, you state that as new experimental or reactor operating data are obtained, the analytical model is refined to improve its capability and accuracy." This implies that a comparison of older versions of the model with test data, as shown in Figure 4.4-4, are meaningless for the WNP-2 facility if it has been analyzed with an updated version. Indicate whether the comparisons of the analytical model with test data, as given in Figure 4.4-4, are based on the same version of the model as was used for the WNP-2 facility. If not, provide comparisons using the present WNP-2 model. In addition, provide a description of the code. While you may reference a submittal on another docket, references to KAPL reports on the code, STABLE, are unacceptable.

221.11 (4.4.4) In Section 4.4.4.6 of the FSAR, you reference the GE topical report NEDO-10802 for the model used to perform system stability calculations. You also state that this model is periodically refined as new experimental or reactor operating data are obtained. Indicate whether

the version of the model used for the analysis of the WNP-2 facility is described in NEDO-10802. If not, describe the changes.

221.12 RSP (4.4.4) We require that a loose parts monitoring (LPM) system be installed in the WNP-2 facility and that it be operational prior to startup testing. Accordingly, provide a description of your proposed LPM system so that we may evaluate it prior to issuance of an operating license. Our positions on the design criteria for a LPM system can be found in Section C of draft Regulatory Guide 1.133, "Loose-Part detection Program for the Primary System of Light-Water-Cooled-Reactors," September 1977. Indicate when you will submit a description of your proposed LPM system.

221.13 (4.4.2) In Table 4.4-6 of the FSAR, you list uncertainties used in the statistical analysis performed to establish the limit which ensures the integrity of the fuel cladding. Provide a discussion of the experimental data base used to derive the uncertainty values listed in Table 4.4-6 and provide appropriate references to this data base, where possible. In particular, describe the applicability of these values to the 8x8, two-water rod fuel bundle which you propose for the WNP-2 facility.

440.0 OPERATOR LICENSING BRANCH

441.01 Provide a detailed description of the fire protection training and the subsequent retraining for the initial plant staff and for the replacement personnel in the manner described in paragraphs II.6.A through II.6.E of Section 13.2 (Revision 1) of the Standard Review Plan (SRP), NUREG-75/087.

441.02 (13.2.1) Discrepancies, possibly in terminology, exist between the training programs described in Section 13.2.1.1 of the FSAR and those cited in Figure 13.2-1. Correct the following discrepancies:

a. The Nuclear Energy training cited in Figure 13.2-1 is not described in Section 13.2.1.1.

b. The course entitled, "Basic Nuclear Theory" described in Section 13.2.1.1.1 is not cited in Figure 13.2-1.

c. The course entitled," BWR Fundamentals," cited in Figure 13.2-1 is not described as such in Section 13.2.1.1.

d. The two week turbine-generator training course cited in Figure 13.2-1 is not described in Section 13.2.1.1.

441.03 Provide the contingency plans for additional training for individuals to be licensed prior to criticality, should the fuel loading be delayed from the date presently indicated in the FSAR (i.e., March 1980). Revise the FSAR, as required, to indicate your best estimate of the fuel load date.

441.04 (13.2.2) The reactivity manipulations performed to satisfy the requalification requirements of Section 3.a of Appendix A to 10 CFR Part 55 must be approved by the NRC staff. Manipulations for boiling water reactors which we find acceptable can be found in the American National Standard ANSI/ANS 3.1.1978, paragraph 5.5.1.2.1, "Control Manipulations."

441.05 (13.2.2) A statement should be included in the description of your program which indicates that the individuals who prepare and grade the Annual Retraining examination are exempt from taking the examination. A maximum of three licensed personnel may be exempted in this manner.