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Mr. R. L. Ferguson, Managing Director
Washington Public Power Supply System
3000 George Washington Way
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Richland, Washington 99352

Dear Mr. Ferguson:

SUBJECT: REQUEST FOR INFORMATION FOR THE REACTOR SYSTEMS BRANCH

In the course of our review of WNP-2 by the Reactor Systems Branch, we have identified a need for additional information. Our request for this additional information is contained in the enclosure to this letter. These questions, with minor changes, are essentially the same questions discussed at the meeting held with WNP-2 on October 16 and 17, 1980.

Sincerely,

Original signed by
Robert L. Tedesco

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosure:
Request for Additional
Information

cc w/encl:
See next page

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ENCLOSURE

REQUEST FOR INFORMATION FOR REACTOR SYSTEMS BRANCH

WNP-2

Q211.107 (3.5.1.2) Regulatory Guide 1.70, Revision 3, Section 3.5.1.2, requires that the structures, systems, and components protected by physical barriers should be identified. The discussion and the figures in the FSAR do not indicate where, if at all, physical missile barriers are used.

Identify all structures, systems, and components that are protected by physical barriers. Provide a description of the types of physical barriers that are employed at your plant.

Q211.108 (3.5.1.2) Section 3.5.1.1.2 of the FSAR states that missile trajectories are selected to encompass the most adverse conditions. It is not clear from the information provided in the FSAR what the trajectories of the credible primary missiles would be and what systems might be disabled by the missiles.

Provide the bases for selection of the probable missile trajectories and show the trajectories on the appropriate FSAR figure. Include a discussion on the system, component, or structure that could be damaged or disabled by a missile. The extent of damage from each missile should be discussed.

Q211.109 (3.5.1.2) Section 3.5.1.1.3.2 states that thermowells and sample probes do not present potential hazards as postulated missiles affecting safe shutdown.

Provide justification to support this position on the thermowells and sample probes.

Q211.110 (5.2.2) The notations, "251 BWR/5-MSIV, 14¢/% Void Coefficient" on Figures 5.2-4 and 5.2-5 indicate that these figures may be generic and not specifically for WNP-2. Confirm that these figures are applicable to WNP-2. If these curves are not applicable to WNP-2, complete the necessary analyses to provide data similar to that now presented on Figures 5.2-4 and 5.2-5.

Q211.111 (5.2.2) Article NB-7200, Overpressure Protection, of the ASME Boiler and Pressure Vessel Code, Section III, requires that an overpressure protection report be provided. No overpressure report could be found in the FSAR. Provide this report.

Q211.112 (5.2.2) Section 5.2.2.4.2.1 of the FSAR states that cyclic testing has demonstrated that the safety/relief valves are capable of at least 60 actuation cycles between required maintenance. Will the actuations of the safety/relief valves recorded? If so, how will these data be recorded and reported to the NRC?

Q211.113 (5.2.2) It would appear that improper setpoints would be a credible common mode failure which could result in degradation of the pressure relief systems. Show that adequate safety margin has been included in the overpressurization analysis to protect against a common mode failure of the safety/relief valves to open at the prescribed values.

Q211.114 (5.2.2) Subsection 5.2.2.4.1 of the FSAR states that each safety/relief valve is provided with a device to counteract the effects of backpressure which results in the discharge line when the valve is open and discharging steam. What type of device is provided? Describe the device and what effects would be anticipated if the device were to fail.

Q211.115 (5.2.2) Subsection 5.2.2.4.1 of the FSAR states that setpoints for the power actuated mode for each safety/relief valve are specified in Table 5.2-2. Table 5.2-2 provides a listing of the setpoints and valve capacities of the valves in the five safety mode groups (spring-operated mode), but no data are presented for the relief mode of operation. Provide the relief setpoint for each safety/relief valve in Table 5.2-2 and in Figure 5.2-6.

Q211.116 (5.2.2) Provide the results of hydraulic calculations that show the Mach number, pressure, and temperature at various locations from upstream of the safety/relief valves to the suppression pool at maximum flow conditions. The concern is related to the potential for the development of damaging shock waves to the discharge piping. Include the effects of suppression pool swell variations on the operation of the safety/relief valves.

Q211.117 (5.2.2) Resolve the following inconsistencies:

a) Figure 3.2-2 of the FSAR indicates in details B and C that the instrument air supply lines to the safety/relief valve air accumulators are safety class G (non-safety grade). Figure 9.3-2 shows these lines as safety class 2 or 3 (safety grade).
b) Figure 5.2-6 shows the safety/relief valves assigned to the automatic depressurization function are F013-M, -N, -P, -R, -S, -U, and -V. Figure 9.3-2 shows the dual accumulators used for the ADS valves assigned to safety/relief valves F013-D, -E, -H, -J, -M, -P, and -S.

Q211.118 (5.2.2) Subsection 5.2.2.4.1 of the FSAR states that the pneumatic accumulator provided for each safety/relief valve has sufficient capacity to provide one safety/relief valve actuation. Figure 3.2-2 indicates that the air supply line upstream of the ball check valve is safety class G (non-safety grade). If the air line were to break upstream of the ball check valve, would there be an indication in the control room of this break and an indication of the accumulator status? If an indication is given, what operator action would be required? Also, show that accumulator capacity for one actuation is sufficient.

Q211.119 (5.2.5) Subsection 5.2.5.2 of the FSAR indicates that temperature and pressure monitoring devices are used as primary detection devices for unidentified leakage. Regulatory Guide 1.45 states that humidity, temperature, or pressure monitoring should be considered as alarms or indirect indications of leakage. Justify this exception to the criteria of Regulatory Guide 1.45. Demonstrate that the unidentified leak detection systems can detect leakage on the order of one gallon per minute in a one-hour period.

Q211.120 (5.2.5) Subsection 7.6.1.13.7 of the FSAR states that the same leak detection monitor (a three-channel unit) will detect both airborne particulate and gaseous activities in the drywell atmosphere using scintillation detectors.

Explain how these two different types of airborne activities are separated by the monitor. Justify taking credit for both monitoring techniques in subsection 7.6.2.4.2.1.2 while using the same device. State the sensitivity and response time of the radioactivity monitor.

Q211.121 (5.2.5) Subsection 5.2.5.5.5 of the FSAR states that the leak detection system will satisfactorily detect unidentified leakage of 5 gpm. Subsection 7.6.2.4.2.1.2 states that the sensitivity and response time for each portion of the leak detection system for detection of unidentified leakage is one gallon per minute in less than one hour (excluding airborne systems). Resolve this inconsistency.

Q211.122 (5.2.5) The response to Q211.007 requires additional information. It is unclear how the comparison will be made between the radioactivity monitoring and the sump level monitoring.

Describe briefly the mechanics of making these data comparisons. What calibration and operability verification tests will be performed for each independent leakage detection system? Which leakage detection system is to be used as the reference for comparison with the other systems? Do the radiation monitoring systems have radioactive sources (check sources) built into the systems?

Q211.123 (5.2.5) Identified leakage is determined during pre-operational testing or is measurable during reactor operation. Will the identified leakage be measured regularly and recorded? If so, provide the frequency that these data will be recorded and indicate what procedural guidelines are to be used to change the magnitude of the base identified leakage rate?

Q211.124 (5.2.5) It is unclear in subsection 5.2.5.2f of the FSAR whether comparative "grab" samples of the continuously monitored containment atmosphere can and will be taken on a periodic basis. Resolve this ambiguity. If "grab" samples are not to be taken, justify the omission of these comparative data.

Q211.125 (5.2.5) Standard Review Plan 5.2.5 specifies that unidentified leakage should be collected separately from the identified leakage so that a small unacceptable unidentified leak is not masked by larger acceptable identified leakage. Section 5.2.5 of the FSAR does not clearly indicate that separate collection of identified and unidentified leakage is provided.

Provide assurances that identified and unidentified leakage will be collected separately. If separate collection is not to be provided, provide justification for use of a common collection reservoir and show that a small unidentified leak of about 1 gpm would be recognized within one hour.

Q211.126 (5.2.5) Provide a list of all indications available to the control room operator for evaluating and detecting unidentified leakage. Show how the operator will determine the amount of leakage by observing the indications that are available to him, including the need for unit conversion (count rate to gpm, etc). If the monitoring is computerized, discuss the backup procedures available should the computer become inoperative.

Q211.127 (5.4.7) Resolve the following inconsistencies in Figure 5.4-14b:

a) In mode B of the RHR system operation, there is an unexplained 500 gpm increase in flow in going from process points 15B to points 21B and 23B.
b) In mode C-1, the sum of the flows past process points 40 and 40.2 should be equal to the flow past point 19. As presented, the sum of flows past points 40 and 40.2 is twice the flow that is tabulated for point 19.
c) In mode E, it is not clear what the total system flow should be (14,900 or 7450 gpm).
d) In the summary of the various modes of RHR system operation, reference is made in mode D to note 13. Note 13 has been deleted from the P&ID. Provide supplemental information to make this reference meaningful or delete this reference altogether.
e) Subsection 5.4.7.1.1.2 and Table 6.3-2 of the FSAR state that the functional design bases for the LPCI mode of RHR operation is to pump 7067 gpm of water per loop into the reactor core region of the reactor vessel. Figure 5.4-14b and the response to Q211.038 state that each loop should supply 7450 gpm to the reactor core region under accident conditions.

Q211.128 (5.4.7) Subsection 5.4.7.1 of the FSAR states that spoolpiece interties are provided to permit the RHR heat exchangers to be used to supplement the fuel pool cooling system.

Describe the administrative controls that will be exercised for the use of these spoolpieces. What would be the effects if the spoolpieces were left in place and the RHR system were operated in any or all of the RHR modes of operation? Similarly, a spoolpiece is shown on drawing M521 that connects the low pressure core spray (LPCS) system to the RHR loop A suction pipe. Describe the purpose of this intertie and, also, describe the effects on both the LPCS and RHR systems if the spoolpiece were inadvertently left in place. Are the same administrative controls used for the fuel pool cooling system spoolpiece used for the LPCS spoolpiece?

Q211.129 (4.6) The standby liquid control system and the recirculation flow control system are reactivity control systems. Address or reference these systems in Section 4.6 and address all requirements of Standard Review Plan 4.6.

Q211.130 (4.6.1.1.2.4) Table 1.3-8 indicates specific design changes from the PSAR to the FSAR for the CRD system. The design changes for the CRD return line modification addressed in Question 211.19, have not been included in the text description of the FSAR and Figures 4.6-5a, 4.6-5b, and 4.6-6a have not been revised. Revise the text description in the FSAR to reflect the specific design changes in Table 1.3-8 for the CRD system and modify the above figures accordingly.

Q211.131 (4.6.1.1.2.4.1) The scram discharge volume-header piping is sized to receive and contain all water discharged by the control rod drives during a scram, independent of the instrument volume. Show quantitatively how a minimum volume of 3.34 gallons per drive is required since approximately 4 gpm is required to insert the rods with up to an additional 0.34 gpm required for cooling.

Q211.132 (4.6.1.1.2.4.2.1) Resolve the following items relating to filtration of condensate water for the CRD hydraulic system.

a) The text description indicates that normal filtration of condensate water on the suction side of the CRD water pump is accomplished by a 25 micron filter and that a 250 micron strainer is provided in the bypass line for the 25 micron filter when it is being serviced. Figure 4.6-6a indicates that double filtration of condensate water on the suction side of the CRD water pump normally occurs via a 250 micron strainer and a 25 micron filter in series. Explain this discrepancy.

b) Describe provisions in the WNP-2 design to protect the hydraulic control units (HCUs) and control rod drives (CRDs) from damage due to inadvertent failure of either the pump suction filter or the drive water filter. If none exist, provide justification that inadvertent failure of either filter will not cause damage to the HCUs and CRDs.

Q211.133 (4.6.1.1.2.4.2.2) In Figure 4.6-5b and Drawing M528, pressure transmitter (N005) transmits a signal to a pressure switch (N600) in the process instrumentation panel in the control room, which energizes an annunciator in the control room at any time pressure in the charging header falls below the setpoint. Explain why an alarm on high is indicated for the pressure switch (N600) instead of an alarm on low which would provide protection against charging header pressure falling below the setpoint.

Q211.134 (4.6.1.1.2.4.2.4) In the text for the CRD cooling water header, there is no discussion of valves F129, F130, F131, and F132 which are shown on Figure 4.6-5B. These valves are not included on Drawing M528. Explain this discrepancy and update the FSAR accordingly.

Q211.135 (4.6.1.1.2.4.3.9) The text description of the scram accumulator indicates that a check valve in the accumulator charging line prevents loss of water pressure in the event supply pressure is lost. The symbol for valve 111 in Figure 4.6-5b and Drawing M528 appears to be that of a normally open globe valve instead of a stop-check globe valve. Explain this apparent discrepancy.

Q211.136 (4.6.2) Identify the specific common mode failure analysis and protection from common mode failures referenced in Section 15A by Sections 4.6.2.1 and 4.6.2.2, respectively.

Q211.137 (4.6.2.3.1.2) Identify the layout studies done to assure that no interference exists which will restrict the passage of control rods and the preoperational test(s) that are used to show acceptable performance.

Q211.138 (4.6.4.1) Provide the common mode failure probability value for the control rod drive and the standby liquid control systems.

Q211.139 (5.4.6.2.2) Provide the following information concerning RCIC equipment and component descriptions.

a) Section 5.4.6.2 of Regulatory Guide 1.70 states that significant design parameters for all components of the RCIC system be identified and that all components be shown on appropriate P&I diagrams. Design parameters for only a portion of the RCIC components are included in Section 5.4.6.2.2.2. Some of the more important components omitted are the:

1) Water leg pump
2) Barometric condenser
3) Vacuum tank
4) Condensate pump
5) Turbine and steam supply drain pots
6) Turbine governing and trip throttle valves
7) Pump suction strainers in the suppression pool

Provide the significant design parameters for all RCIC components not included already in Section 5.4.6.2.2.2 and verify that each component can be identified on Figures 5.4-9a and 5.4-9b.

b) The RCIC turbine is identified as component C001 in Section 5.4.6.2.2.2 and as component C002 in Figure 5.4-9b. Correct this discrepancy.

Q211.140 (5.4.6.2.1.3) Four keylocked valves (F063, F064, F068, and F069) are indicated in step "a" as electrical interlocks. However, one of these valves, valve F064, is not indicated as keylocked in Figure 5.4-9a, while valve F008 is indicated as keylocked. Resolve this discrepancy.

Q211.141 (5.4.6) Is the RCIC electro-hydraulic system integrated with the turbine governing valve of a safety grade design (i.e., Seismic Category I)?

Q211.142 (5.4.6) Describe the design features and operating procedures that preclude water hammer effects at the pump discharge of the RCIC system.

Q211.143 (5.4.6.4) Show how the pre-operational initial startup test programs for the RCIC system in Section 14.2.12.1.8 meet the intent of applicable sections in Regulatory Guide 1.68.

Q211.144 (5.4.6) The ASME Boiler and Pressure Vessel Codes, Section III, Article NB-7000 requires that individual pressure relief devices be installed to protect lines and components that can be isolated from normal system overpressurization protection. With reference to appropriate P&ID, identify those portions of the RCIC system that can be isolated from normal system overpressure protection. Discuss the relief devices provided or provide the basis for deciding that relief devices are not required.

Q211.145 (5.4.6) At some BWR installations, the check valves in the turbine exhaust line of the RCIC system which serve a containment isolation function have been damaged as the result of intermittent closure. The intermittent closures arise from flow oscillations in the exhaust line associated with formation and collapse of steam bubbles in the suppression pool. One type of corrective action involves use of a sparger on the exhaust piping in the suppression pool to reduce the flow oscillations.

a) Is the 10" exhaust pipe shown in Figure 5.4-9a installed as a sparger for this purpose?

b) Are there other design features used at WNP-2 to prevent this type of damage?

Q211.146 (5.4.6) In the responses to Questions 211.046 and 031.015, it is stated that an automatic safety-grade switchover from the condensate storage tank to a Seismic Category I supply (i.e., the suppression pool) has been provided as a convenience to the operator. Provide a description of the automatic switchover feature and its initiating signal and confirm that both electrical and mechanical features are safety grade.

Q211.147 (5.4.6.1.2.1) (5.4.6.2.4e) The text indicates all components of the RCIC system are capable of individual functional testing during normal plant operation. Table 1.3-8 indicates each component, except the flow controller, is capable of functional testing. Resolve the discrepancy with respect to functional testing of the RCIC flow controller.

Q211.148 (15.0) Resolve the following items in Table 15.0-2:

a) Modify the values of vessel level trip to agree with the values specified in Figures 5.2-6 and 5.3-2 (item 29).

b) Specify the maximum percent relieving capacity assumed in Chapter 15 for each mode of SRV actuation (items 25 and 26).

c) Provide the following information concerning the high flux trip setpoint used as input to the REDY model (item 29):

   1) Explain why the high flux trip setpoint should not be increased to 122% NBR prior to multiplication by the thermal-power correction factor of 1.043 to account for the setpoint plus calibration error, instrument accuracy, and transient overshoot specified in Table 7.2-4.

   2) Explain why the thermal-power correction factor is applied to the high flux trip setpoint used in the REDY model.

d) Provide the following information concerning the APRM thermal trip setpoint used as input to the REDY model (item 30):

   1) Specify the highest flow-related trip setpoint to be given in the Technical Specifications and how this value is obtained.

   2) Is the 122.03% NBR setpoint equal to the setpoint to be specified in step d)1) times the thermal-power factor of 1.043 specified in step c)1)?

e) Table 15.0-2 does not contain all of the input parameters used in the REDY computer code. For each transient and accident analyzed in Chapter 15, provide the following:

   1) A list of all input parameters.

   2) Justification that the input parameters are conservative.

Q211.149 (15.0) Provide a realistic range and permitted operating band for the exposure dependent parameters in Tables 4.4-1 and 15.0-2. In Table 15.0-2, provide assurance that values of parameters selected yield the most conservative results.

Q211.150 (15.0) Provide a listing of the transients and accidents in Chapter 15 for which operator action is required in order to mitigate the consequences. For corrective actions required prior to 20 minutes, provide justification.

Q211.151 (15.0) The analysis of transients and accidents in Chapter 15.0 does not state which of the RPS time response delays in Table 7.2-5 is used in the REDY computer model (NEDO-10802). For each transient and accident in Chapter 15.0, specify which delay time in Table 7.2-5 is used in the analysis and why the specified delay time is conservative.

Q211.152 (15.0) In relation to Figure 15.0-2, confirm the following items for all transients in Chapter 15.0 which require control rod insertion to prevent or lessen plant damage:

a) The scram curve used in Chapter 15.0 analyses (Figure 15.0-2) has a total reactivity worth of $37.05 and is the nominal scram curve multiplied by the standard transient safety conservatism factor of 0.80.
b) The slowest allowable scram insertion speed was used for the scram curve applied to Chapter 15.0 analyses.
c) The end of cycle 1 scram curve has a total reactivity worth of $40.21 and is identified incorrectly in Figure 15.0-2.

Q211.153 (15.0) For transient analysis, credit has been taken for safety/relief valve (SRV) actuation only in the relief mode. A more conservative approach would be to take credit for SRV actuation in the safety mode, resulting in higher peak vessel pressures.

a) What quantitative effect on MCPR and peak vessel pressure does credit for SRV actuation only in the safety mode have on each transient analyzed in Chapter 15?
b) In Section 5.2.2, the relief mode appears to be nonsafety grade because credit for 50% relieving capacity associated with power-actuated pressure-relief valves in ASME B&PV Code Section III, NB-7000, was not assumed for overpressure protection. Are all equipment and components required for SRV actuation in the relief mode nonsafety grade? If not, identify specific equipment and components that are safety grade and those which are nonsafety grade.
c) If the relief mode is nonsafety grade, explain why credit was taken for this mode of SRV actuation in Chapter 15.0. If the relief mode is safety grade, explain why credit for SRV actuation with up to 50% relieving capacity in the relief mode and additional relieving capacity up to 50% as required in the safety mode was not applied to analyses in Section 5.2.2 and Chapter 15.

Q211.154 (15.0)

Modify the sequence of events tables in Section 15.0 to specify the opening and closing times of referenced valves and the time at which each reactor vessel alarm or trip water level is attained throughout the duration of each transient in places where this information is not already included. Include appropriate delay times from the initiating signals and confirm the delay times are applied consistently between the event tables.

Q211.155 (15.0)

Modification of NSOA drawings to include use of nonsafety-grade systems or components which mitigate transients and accidents was requested in Question 211.85. In conjunction with this request:

a) Provide a table of the nonsafety-grade equipment and components assumed to mitigate consequences for each.
b) Provide the $\Delta(\Delta CPR)$ and $\Delta(\Delta$peak vessel pressure) that would result if only safety-grade systems or components were assumed in the analysis for each event in Section 15.0 that takes credit for specific nonsafety-grade systems or components.

Q211.156 (15.0)

Discuss how the pre-operational and startup tests will be used to confirm flow parameters used in Chapter 15 analyses. Provide details of any previous test of components in test facilities conducted to show satisfactory performance of the recirculation and feedwater flow control systems and respective pumps.

Q211.157 (15.0)

Analyze the turbine trip and generator load rejection transient from a safe shutdown earthquake event. Credit should not be taken for non-seismically qualified equipment or any equipment contained in a non-seismic structure.

Q211.158 (15.0)

On page 4-7 of NEDO-10802, it is stated that the difference in trend of flow coastdown versus initial power between the analytical and experimental coastdown curves for Dresden Unit No. 2 (a BWR/3) in Figure 4-11 was due in part to differences between actual and computed jet pump efficiencies.

a) How has this effect been treated in analysis of WNP-2 transients involving flow coastdown with two recirculation pump trips?
b) Is this treatment applicable to WNP-2 which is a BWR/5?

Q211.159 (15.0)

GE calculations performed for decrease in reactor coolant temperature (Section 15.1) and for reactor pressure increase (Section 15.2) events using the proposed ODYN licensing basis model (NEDO-24154) have shown that in some cases a more limiting CPR is predicted than by the current REDY licensing bases model (NEDO-10802). Since Question 211.49 was submitted, the ODYN model has been approved. Based on a letter to Glen G. Sherwood dated 1/23/80 from Richard P. Denise, the staff's ODYN licensing position is that GE can proceed with ODYN analysis of certain events described in Section 15 of licensing application Safety Analysis Reports. Provide the following additional information in conjunction with Question 211.49:

a) An ODYN analysis of the applicable events (One-D) listed in Tables 2-1 and 2-2 of NEDE-25154-P.
b) A list of all input parameters for each event.
c) Justification that input parameters for each event are conservative.

Q211.160 (15.0)

For each transient and accident analyzed in Section 15, identify each normally operating system for which credit has been taken.

Q211.161 (15.0)

Provide assurance that the limiting pump trip is assumed in analyzing decrease in reactor coolant system flow rate transients. Different trip signals may cause different coastdown characteristics. Identify the trip signal that can be expected to produce the most severe pump coastdown characteristics.

Q211.162 (15.0)

In the analyses for the generator load rejection and turbine trip transients, credit is taken for immediate reactor scram and recirculation pump trip obtained from a valve closure signal (turbine control valve for load rejection and turbine stop valve for turbine trip). Analyze these transients without taking credit for immediate reactor scram and recirculation pump trip. Take credit only for safety-grade, seismic Category I equipment and assume loss of offsite power. What is the effect of the failure of a single safety-grade component? Provide the effect on analytical results that WNP-2 operation with the new 8 x 8 fuel design with two water rods will have.

Present curves similar to those of Figures 15.2-2 and 15.2-4 and give values of maximum vessel pressure and minimum MCPR with the times at which these values occur. Evaluate the percent of fuel rods which would reach boiling transition. Since this event is not an anticipated transient, limited fuel failure can be allowed if dose consequences are acceptable.

Q211.163 (15.0)

For the majority of events analyzed in Chapter 15, the recirculation flow control mode (automatic or manual) assumed in the analysis is not specified. Our concern is that the mode selected may not result in the most severe margins on MCPR and peak vessel pressure.

a) Specify the recirculation flow control mode assumed for each event analyzed in Chapter 15.
b) Specify the change in MCPR and peak vessel pressure that results in these parameters for each event if the opposite recirculation flow control mode had been assumed in the analysis.

Q211.164 (15.1.1.2.2)

On page 15.1-2, it is stated that the thermal power monitor (TPM) is the primary protection system for mitigating the consequences of the transient resulting from loss of feedater heating. A description of this monitor, which typically involves the flow-weighted APRM scram in conjunction with a 6-second time constant circuit, was not found in the WNP-2 FSAR. Provide this this description in sufficient detail to permit evaluation of the TPM for WNP-2.

If the time constant, which affects scram initiation by the TPM, is less than the effective time constant for the WNP-2 fuel for this type of transient, the TPM should provide a conservative measure of the time variation in surface heat flux. However, if the time constant is appreciably larger than that for the fuel, the fixed APRM trip without a time constant would provide the scram protection. The resulting MCPR would then be less than that predicted for the TPM scram which has a lower setpoint.

There is no current provision in the Technical Specifications for surveillance of this time constant circuit. It is the staff's position that credit be taken only for the fixed APRM scram in Chapter 15 unless the TPM is approved by the staff and appropriate limiting conditions for operation and surveillance requirements are incorporated in the Technical Specifications for WNP-2.

a) Provide an analysis of the "loss of feedwater heating" transient assuming credit only for the fixed APRM scram. This is a more conservative approach because it will result in a more severe transient due to the higher fixed APRM scram setpoint.
b) Revise NSOA Figure 15A.6-21 to indicate the high flux scram signal occurs from the fixed APRM scram instead of the TPM.
c) Re-evaluate single failure criteria in Section 15.1.1.2.3 without taking credit for the TPM.

Q211.165 (15.1.2.3.2) Provide the following information relating to the "feedwater controller failure at maximum demand" transient:

a) Explain the discrepancy between the assumed feedwater controller failure valves at maximum demand specified in the text (135% flow) and in Table 15.1-3 (146% flow). Provide the basis for selecting the magnitude of FW flow increase assumed in the analysis.

b) In conjunction with the magnitude of feedwater (FW) increase assumed in the analysis, explain why the full FW increase is attained at essentially zero seconds in Figure 15.1-3. In GESSAR 238-732, the FW increase is initiated at zero seconds and attains the full value (maximum demand) at approximately 5 seconds.

c) If the FW temperature at the reactor vessel has been assumed constant, provide a quantitative analysis that includes the effect of FW temperature variation on MCPR and the basis for determining this variation. Incorporate any changes from step a) above concerning the appropriate value of FW flow rate assumed in the analysis of this transient.

Q211.166 (15.1.3.3.2) The pressure regulator failure at 115% NBR steam flow is simulated in Figure 15.1-4 in a manner consistent with GESSAR 238-732. However, the assumed pressure regulator failure value of 115% NBR steam flow for WNP-2 appears low compared to a failure value of 130% steam flow used in other FSARs with approximately 15% greater than the normal maximum flow permitted by the steam flow limiter.

a) Explain the difference between the 110% NBR steam flow indicated as the normal maximum flow limit in this section and the 115% value specified in Section 15.1.3.1.1.

b) Explain the basis for selecting the assumed pressure regulator failure value of 115% NBR steam flow used in the FSAR. If a new steam flow value in excess of that permitted by the steam flow limiter is chosen, provide the basis for selecting the amount of steam flow in excess of that permitted by the steam flow limiter.

Q211.167 (15.1.3.3.3) The depressurization rate has a proportional effect on the voiding action of the core. For the "pressure regulator failure-open" transient, the assumed depressurization rate results in a L8 trip. The results are not consistent with GESSAR 238-732 where a lower depressurization rate results in a trip from low turbine inlet pressure. Explain this discrepancy and provide justification that the assumed trip provides the most restrictive margins on MCPR and peak vessel pressure.

Q211.168 (15.1.4.2.1) For the "inadvertent opening of a safety/relief valve" transient, include the time at which suppression pool temperature alarms and Technical Specification limit are attained in event Table 15.1-5.

Q211.169 (15.0) Modify Table 15.0-1 as follows:

a) Provide a calculated MCPR value for all events in Table 15.0-1 where a MCPR value is not specified.

b) Correct the following discrepancies between values of parameters in Table 15.0-1 and corresponding text values and confirm other discrepancies do not exist.

Event	Maximum Core Average Surface Heat Flux, % of Initial: Table	Maximum Core Average Surface Heat Flux, % of Initial: Text	SRV Actuation: Table	SRV Actuation: Text
15.2.6 (Case 1)	-	-	No	Yes
15.4.4	146.6	80.6	-	-
15.4.5 (Case 1)	141.0	79.0	-	-
15.4.5 (Case 2)	134.6	75.0	-	-

Q211.170 (15.0) For event category 15.3 in Table 15.0-1, identify the most limiting anticipated transient for MCPR and maximum vessel pressure.

Q211.171 (15.0) Provide an analysis of the "loss of instrument air" transient.

Q211.172 (15.6.5.2.1) In the description of event sequences for LOCA inside containment, several items need additional clarification.

a) The initiating times for MSIV closure and ECCS actuation in the text description appear inconsistent with the corresponding event occurrence times in Table 6.3-1. Explain these apparent discrepancies.

b) Confirm that the zero reference time for Tables 6.3-1 and 6.2-8 are the same.

Q211.173 (15.1.3.2.1) Add the "initial core cooling" safety action indicated in NSOA Figure 15A.6-23 for the "pressure regulator failure-open" transient to event Table 15.1-4 for consistency.

Q211.174 (15.2) The treatment of uncertainties associated with SRV setpoints appears to be handled in three different ways for the events associated with the sections shown below:

Section	Treatment of SRV Setpoint Uncertainties
15.2.3.3.4	Setpoints include errors (high) for all valves
15.2.4.3.4	Setpoints are assumed 15 psi higher than the valves nominal setpoint
15.2.5.3.4	Setpoints are assumed at upper limit of Technical Specifications for all valves.

Explain this apparent discrepancy. If no discrepancy exists, standardize the wording between these sections for consistency.

Q211.175 (15.2.1.2.3) It is indicated that the "pressure regulator-closed" transient with failure of the backup pressure regulator is less severe than the "turbine trip with bypass"transient in Section 15.2.3. This agrees with GESSAR 238. As a result, only a qualitative evaluation of the transient was provided. However, quantitative results from the Grand Gulf FSAR indicate the opposite. The staff's concern is that quantitative results for this transient may be similar to those for Grand Gulf. Provide a quantitative analysis of the "pressure regulator-closed" transient assuming failure of the backup pressure regulator.

Q211.176 (15.6.5.3.3) In Table 6.3-3, it is indicated that the corewide metal-water reaction for WNP-2 has been calculated at 102% of licensed core power. Explain why the above calculation was not based on the thermal power of 3462 MWt specified in Table 6.3-2 (104.18% of licensed core power) to be consistent with the thermal power value used for LOCA calculations inside containment.

Q211.177 (15.2.7.2.1) Review of the "loss of all feedwater flow" transient indicates that the feedwater flow decreases to zero in 5 seconds. For the analyses presented in the FSARs indicated below, the reactor vessel water level decreases to the L3 scram trip setpoint as follows:

FSAR	Time at which L3 trip occurs, sec	Vessel ID, in./no. of fuel assemblies	Rated Power, MWT
Susquehanna	4.6	251/764	3293
Fermi-2	6.8	251/764	3293
Grand Gulf	4.1	251/800	3833
WNP-2	7.36	251/764	3323

For WNP-2 analysis, it would appear that the L3 setpoint would be reached at a time slightly less than that for either Susquehanna or Fermi-2 because the power level is slightly higher and all three have the same size vessel. Provide an explanation as to why the L3 setpoint for WNP-2 should not be attained before that for Susquehanna or Fermi-2. Include appropriate design considerations (differences in piping, setpoints, etc) in the response.

Q211.178 (15.2.3.3.2) A turbine stop valve full-stroke closure time of 0.10 seconds is used in the analysis of the "turbine trip" transients. Demonstrate quantitatively or provide references that show that turbine stop valve full-stroke closure times smaller than 0.10 second do not result in unacceptable increases in ΔCPR and reactor peak vessel pressure for transients analyzed in Section 15, or provide either (1) justification that a smaller full-stroke closure time cannot occur or (2) a minimum full-stroke closure time that will be incorporated in the Technical Specifications.

Q211.179 (15.2.4.3.2) The "closure of all MSIVs" transient (closure time 3 sec) results in a position switch scram at 0.3 second and indirectly causes a scram trip of the main turbine and generator due to the decrease in pressure sensed by the main turbine. From Figure 15.2-5, it cannot be determined whether or not a turbine stop valve and turbine control valve scram occurs during the time interval that the MSIVs are closing from the full open position to the 90% scram position. Indicate in Table 15.2-5 the time at which the above indirect scram trips occur and the times at which the TSVs and TCVs become fully closed.

Q211.180 Question deleted.

Q211.181 (15.2.6.3.4) For the "loss of AC power" transients, it is indicated that the trip of the feedwater turbines may occur earlier than simulated if the inertia of the condensate and booster pumps is not sufficient to maintain feedwater pump suction pressure above the low suction pressure trip setpoint. The simulation of this transient assumes sufficient inertia and, thus, the feedwater pumps are not tripped until the time that level reaches the high water level trip setpoint (L8). What quantitative effect on MCPR and peak vessel pressure would an earlier trip (insufficient inertia) of the feedwater turbines have?

Q211.182 (15.2.9.2.1) Revise Table 15.2-12 to indicate the time that suppression pool alarms are received, the Technical Specification limit is exceeded, and the maximum value of the suppression pool temperature is attained.

Q211.183 (15.3.1.3.2) In the analysis of the one and two recirculation pump trip events in Section 15.3.1, a minimum design rotating inertia was used to obtain a predicted rate of decrease in core flow greater than expected. Specify the inertia value used for each applicable transient in Section 15 and the basis for selection. Discuss the sensitivity of MCPR and peak reactor vessel pressure to changes in the inertia value.

Q211.184 (15.3.2.3.3) From the text description in the Grand Gulf FSAR, it is indicated that the design of the hydraulic limit on maximum valve stroking rate is intended to make the fast closure of one and two recirculation valve transients less severe than the corresponding trip of one and two recirculation pump transients in Section 15.3.1. However, the results for events 15.3.1 and 15.3.2 in Table 15.0-1 indicate that for the one valve case this does not occur for WNP-2.

a) Explain why the transient result for the one valve closure event in Section 15.3.2 is more severe than the result in Section 15.3.1.
b) Explain why a scram occurs for the analysis of the "fast closure of one recirculation valve" transient in the WNP-2 FSAR in view of the fact that for the same analysis presented in the Grand Gulf FSAR, no scram occurs.

Q211.185 (15.3.3) For the recirculation pump seizure accident we note in Table 15.3-5 that credit is taken for nonsafety-grade equipment (L8-trip) to terminate this design basis accident (DBA). Section 15.3.3 of the Standard Review Plan requires use of only safety-grade equipment to mitigate the consequences of this DBA and that the safety functions be accomplished assuming the worst single failure of an active component. Re-evaluate this DBA with the above specific criteria and provide the resulting ΔCPR, peak vessel pressure, and percentage of fuel rods in boiling transition. Assume coincident loss of offsite power as required by the Standard Review Plan.

Q211.186 (15.4.4.2.3) You state that, for the incident involving an abnormal startup of an idle recirculation pump, "Attempts to start the pump at higher power levels will result in a reactor scram on flux." Since such a transient may be more severe than the one presented, supply the analysis beginning at a higher power level.

Q211.187 (15.4.4.3.3) The narrative on page 15.4-19 for the "startup of an idle recirculation pump" transient indicates the core inlet flow rises sharply shortly after the pump starts. However, Figure 15.4-6 does not show this sharp change. Explain.

Q211.188 (15.4.5.3.2) A maximum stroking rate of 30%/second and 11%/second was used for the fast closure of one and two recirculation control valves, respectively, in this section and for the events in Section 15.3.2. In the description of the recirculation control valve stroke rate in Appendix H.3.3.3.7.3.1, the bases for the above stroking rates are not provided. Provide supporting data to justify how the above stroking rates for the analysis of events in Sections 15.4.5 and 15.3.2 were obtained.

Q211.189 (15.2.9.3) For the "failure of RHR-shutdown cooling" event, specific input parameters for the models used to evaluate blowdown rate and suppression pool temperature are shown in Table 15.2-13 along with the analytical results in Figures 15.2-16, -17, -18, and -19. In connection with this, provide the following information:

a) Identify the analytical models used to evaluate blowdown rate and suppression pool temperature.
b) Revise Table 15.2-13 to include all the input parameters for the models to be identified in step a) and provide justification that the input parameters are conservative.

In addition, it is indicated that only a qualitative evaluation of the "failure of RHR shutdown cooling" transient is provided because the core behavior has been analyzed in Section 15.2.6. Update the FSAR to indicate a quantitative analysis has been provided.

Q211.190 (15.4.5.3.3.1) Explain the following from Figure 15.4-7, "Fast Opening of Main Recirculation Loop Valve at 30% Per Second":

a) What causes the drive flow to exceed 100% of rated and level out?
b) Why doesn't the core inlet flow exceed 100% of rated as a result of the drive flow exceeding 100% of rated?

Q211.191 (15.4.5.3.3.2) What causes the core inlet flow and drive flow to exceed 100% of rated in Figure 15.4-8, "Fast Opening of Both Recirculation Loop Valves at 11% Per Second"?

Q211.192 (15.5.1.3.2) Provide justification for use of a HPCS injection temperature of 40°F in analysis of the "inadvertent HPCS startup" transient. Referenced studies should be specified.

Q211.193 (15.5.1.2.3) From the discussion of single failures for the "inadvertent HPCS startup" transient, it is indicated that a single failure of the pressure regulator or level control will aggravate the transient, resulting in reduced thermal margins. Provide the MCPR and peak vessel pressure values that result for this event with the most limiting of the above single failures considered in the analysis.

Q211.194 (15.0) The response to Question 221.02 indicates that 8 x 8 fuel bundles with two water rods will be used at WNP-2 instead of the 8 x 8 fuel bundles with one water rod.

a) Have the transients and accidents in Chapter 15 been evaluated with 8 x 8 fuel bundles using one or two water rods?

b) If the transients and accidents in Chapter 15 were analyzed with the one water rod fuel bundles, what changes in MCPR, peak vessel pressure, percent of rods experiencing boiling transition, and the radiological consequences will result if the two-water-rods design is used in the analyses?

Q211.195 (15.6.5.3.2) In connection with parameters and assumptions used for LOCA calculations inside containment, provide the following items to aid the staff in evaluating their conservatism.

a) An explanation as to why a MSIV closure time of 3.5 seconds in Table 6.2-3 was chosen. Elsewhere in the FSAR, either 3 seconds or 5 seconds were used in analyses.

b) Explain why the core heatup calculation in Table 6.3-2 assumes a bundle power consistent with operation of the highest powered rod at 102% of the maximum (technical specification) linear heat generation rate (LHGR) instead of operation at 104.18% of the maximum LHGR which is equivalent to a core thermal power of 3462 MWt.

c) Explain why the core thermal power value of 3462 MWt in Table 6.2-4 is indicated as 102% of licensed core thermal power (3323 MWt) instead of 104.18%.

d) A tabulation of all permitted axial power shapes addressed by LOCA calculations inside containment. Identify the least favorable axial shape (most conservative) associated with each break size and provide justification of its conservatism.

Q211.196 (6.3) Operating experience has shown that where thermocouples are used to verify ADS valve operation a "false" temperature increase may be indicated even though the valve has not operated. A direct indication of valve position or flow must be used. Specify how you will meet this requirement.

Q211.197 (6.3)

Section 6.3.2.2.1 of the FSAR states that the HPCS system will automatically switch over from the condensate storage tank (CST) to the suppression pool if the CST water supply becomes exhausted or is not available. Review of Figure 7.3-10b indicates that automatic switchover will only occur if the CST water level drops to the minimum level and activates any one of the four level switches (two per tank). However, in the event that CST water cannot be supplied to the pump while the CST water level is above the minimum water level, automatic switchover is precluded. Resolve this apparent discrepancy between the P&IDs and Section 6.3.2.2.1.

Q211.198 (6.3)

Expand the discussion in Section 6.3 to describe the design provisions that are incorporated to facilitate maintenance (including draining and flushing) and continous operation of the ECCS pumps, seals, valves, heat exchangers and piping runs in the long term LOCA mode of operation considering that the water being recirculated is potentially very radioactive.

Q211.199 (6.3)

Discuss the design provisions that permit manual override on the ECCS subsystems once they have received an ECCS initiation signal. Also, include a discussion of any lockout devices or timers that prevent the operator from prematurely terminating ECCS functions. If there are plant procedures to cover this situation, indicate briefly what instructions are provided.

Q211.200 (6.3)

Provide isometric drawings of the major piping for each ECCS subsystem (i.e., LPCI, LPCS, etc) to aid in the evaluation of NPSH and possible equipment flooding. These drawings should show relative elevations and physical locations of the valves, suppression pool, primary containment, pumps, heat exchangers, and the lengths of ECCS piping. The location and number of each of the major valves should be shown on the isometric drawing.

Q211.201 (6.3)

Several plants have used sandbags or sand-filled tanks as biological shielding inside containment. In the event of a LOCA, these tanks or bags could be damaged and sand could be released. Release of sand inside containment could result in damage to the ECCS pumps. Identify any areas where sandbags or sand-filled tanks are used for biological shielding. What precautions would be taken to prevent ECCS damage if sand or similar material were released within containment?

Q211.202 (6.3) A timer is used in each ADS logic. The time delay setting before actuation of the ADS is long enough that the HPCS system has time to operate, yet not so long that the LPCI and core spray systems are unable to adequately cool the fuel if the HPCS system fails to start. Manual reset circuits are provided for the ADS initiation signal and primary containment high pressure signals. By resetting these signals manually, the delay timers are recycled. The operator can use the reset pushbuttons to delay or prevent automatic opening of the relief valves if such delay or prevention is necessary. The operator may also interrupt the depressurization at any time by the same action. The operator would make this decision based on an assessment of other plant conditions.

Discuss in detail any criteria to be given to the operator (e.g., in emergency procedures, or operator training) that would form the bases for the operator's decision. Discuss the consequences of interrupting ADS depressurization prior to reaching the injection pressure for low pressure systems.

Q211.203 (6.3) Restricting orifices are commonly installed downstream of a pump to limit the maximum flow rate that could occur and prevent pump damage if the pump discharge line were to fail (i.e., pump runout protection). It is not clear whether or not restricting orifice plates will be used for the LPCI system at WNP-2. Figures 5.4-13a and 5.4-13b show a restricting orifice in the injection piping of each LPCI loop. However, note 9 on Figure 5.4-13a states that these orifices are recommended but not required.

Describe precautionary measures taken to reduce the potential for LPCI pump damage due to runout conditions.

Q211.204 (6.3) Figures 6.3-53a, -53b, -54a, and -54b show the results of a break in a core spray line from the "lead plant" analyses. The assumed single failure shown on the figures does not appear to be the most limiting. It would appear that the LPCI diesel-generator failure (division 2) would be more restrictive than the LPCS diesel-generator failure (division 1), i.e., only LPCI loop A would be available to reflood the core. Explain why failure of the LPCI diesel-generator (division 2) does not result in a higher peak cladding temperature than that shown on Figure 6.3-54b.

Q211.205 (6.3)

Resolve the following discrepancies or inconsistencies:

a) Table 1.3-3 and Figure 6.3-2 state that the HPCS system will deliver 6350 gpm at a differential pressure (vessel to pump suction) of 200 psid. Table 6.3-2 indicates that HPCS will deliver 6250 gpm at the same differential pressure.

b) Table 1.3-3 and Figure 6.3-6 state that the LPCS system will deliver 6350 gpm at a differential pressure (vessel to drywell) of 128 psid. Table 6.3-2 indicates that the LPCS system will deliver 6250 gpm at a differential pressure of 122 psid.

c) Table VI of Figure 5.206 indicates that the top of the active core is 360.3 inches above vessel zero. Figure 5.3-2 indicates that the top of the active core is 366.31 inches above vessel zero.

d) Subsection 6.3.2.2.4 of the FSAR does not mention the relief valves (F088A and F088B) that are installed on the suppression pool suction pipes for loops A and B. These valves are the same size as the loop C valve (F088C). See the response to Q211.027 and Figures 5.4-13a and 5.4-13b.

Q211.206 (6.3)

The ECCS discharge line fill systems require additional clarification. Provide the jockey pump characteristics (head, capacity, etc) and the maximum expected leakage rates for each system discharge piping.

Q211.207 (6.3, 5.2.2)

Subsection 5.2.2.10 of the FSAR states that the manual and automatic actuation of the relief mode for each safety/relief valve is to be verified in preoperational testing. Subsection 6.3.4.2.2 of the FSAR states that each individual ADS valve is manually actuated prior to or following a refueling outage. The spring setpoint (safety mode) of each valve is to be checked during bench tests during refueling outages. On what schedule will safety/relief valves, other than the ADS valves, be manually operated in the relief mode to verify that the valve is operational?

How many of the safety/relief valves will be removed during each refueling outage to receive preventive maintenance and be tested?

Q211.208 (6.3)

Appendix A to Regulatory Guide 1.68, Rev 2, summarizes the systems to be tested and the performance capabilities that should be demonstrated by each BWR applicant during the preoperational and initial test programs.

It is unclear if the ECCS subsystems are tested using normal and emergency power supplies. Provide assurances that both the normal and emergency power supplies are used to verify ECCS operability.

If emergency power is not to be used in the operability tests, justify the exception to the criteria of Regulatory Guide 1.68, Rev 2.

Q211.209 (5.2.2) (6.3) Provide assurance that your relief valve design is qualified (including testing after being subjected to an environment representative of an extended time period at normal operating conditions) to support your assumption that six of the seven ADS valves will operate. A quantitative history of safety/relief valve operation, including similar valves in other plants, should be included in this evaluation.

Q211.210 (15.2.2) The response to Q211.088 is unacceptable. It is indicated that because the generator load rejection transient is not the most limiting transient, the small increase in surface heat flux that occurs for TCV closure times of less than 0.15 seconds will not affect the MCPR operating limit. Because reclassification of the generator load rejection transient to a moderate frequency event may result in it being the most limiting transient, even with reanalysis by ODYN, the effect of TCV closure times of less than 0.15 seconds should be reconsidered in the derivation of the MCPR operating limit.

Q211.211 (15.3.3) The response to Q211.092 is unacceptable. Explain why the DBA-LOCA event is indicated as conservatively bounding the pump seizure event when different acceptance criteria are used for each. The pump seizure event is evaluated based on exceeding 10 CFR 100 guidelines whereas the main criterion for evaluating the DBA-LOCA event is a peak cladding temperature of 2200°F. Coordinate this request with Q211.185.