MAY 29 1979

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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 - RSB

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 ninth set of our round one questions. The enclosed questions represent the review effort of the Reactor Systems Branch and reflect staff concerns regarding the functioning of the reactor systems of the WNP-2 facility based on the General Design Criteria of 10 CFR Part 50, our regulatory guides, our acceptance criteria in the Standard Review Plan and the review of similar facilities. In order to maintain our present schedule, we need a completely adequate response to all questions in the enclosure by July 11, 1979.

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

Sincerely,

Original signed by:
S. A. Varga

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

Enclosure:
Requests for Additional Information

cc:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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300 George Washington Way
P. O. Box 968
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Information

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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

211.0 REACTORS SYSTEMS BRANCH

211.49 The analyses you present in the FSAR to show compliance with the requirements for protection against overpressurization which are contained in the ASME Boiler and Pressure Vessel Code, refers to the General Electric topical report, NEDO-10802, for the analytical model used to evaluate transients in the WNP-2 facility. However, GE has submitted an updated analytical model, ODYN, to evaluate plant transients. Accordingly, reanalyze the pressure transients in the WNP-2 facility using the ODYN code. Alternatively, provide assurance that the method of analysis described in NEDO-10802 is bounding in regard to predictions of the peak pressure. The analysis must include the effects of the recirculation pump trip (RPT) due to high pressure and the RPT trip resulting from the turbine stop valve/control valve closure, where applicable. If you reanalyze the pressure transients using the ODYN code, provide an analysis which establishes whether the closure of all main steam isolation valves (MSIV's) is the most severe overpressure transient, including consideration of a second safety-grade scram (e.g., a scram resulting from a high neutron flux) and the effects of the RPT.

211.50 You have not provided sensitivity studies in the FSAR which show the effect of the initial operating pressure on the peak transient pressure attained during a limiting overpressure event. Accordingly, submit the following additional information:

a. Provide a sensitivity study which shows that increasing the initial operating pressure, up to the maximum pressure permitted by the high pressure trip setpoint, will have a negligible effect on the peak transient pressure.

b. Alternatively, propose an operating limitation on the reactor pressure which will be incorporated into the WNP-2 Technical Specifications, thereby providing assurance that the actual reactor operating pressure will not exceed the initial pressure assumed in your analysis of pressure transients.

211.51 (5.2.2) The performance of essentially all types of safety/relief valves has been below the expectations for this type of safety-related component. Based on the number of reportable events involving malfunctions of these valves in operating boiling water reactors (BWR's), we believe that significantly improved performance of the safety/relief valves (SRV's) should be required of the SRV's installed in new plants such as the WNP-2 facility. Accordingly, provide a detailed description of the provisions you will incorporate in the SRV's of the WNP-2 facility which represent an improvement over the SRV's of presently operating BWR plants in the six areas listed below. In responding to this item, explain why or how your additional provisions will provide the improvements which we seek in the performance of the SRV's. Finally, identify the SRV manufacturer.

a. Valve and valve operator type and/or design. Provide a discussion of your proposed improvements in the air actuator, especially in the materials used for such components as the diaphragms and the seals. Discuss the safety margins and confidence levels associated with the air accumulator design. Discuss the capability of the reactor operator to detect low pressure in both air accumulators.

b. Specifications. Indicate what new provisions you have employed to ensure that the specifications for the valves and valve actuators include design requirements which reflect the operation of the SRV's over the anticipated range of environmental conditions (i.e., the temperature, humidity, and vibration), to which the valves and valve actuators will be subjected during plant transients and postulated accidents.

c. Testing. It is our position that prior to installation, the SRV's should be proof-tested under the appropriate environmental conditions, for time periods representative of the most severe operating conditions, to which they may be subjected.

d. Quality Assurance. Indicate what new programs you have instituted to assure that valves are manufactured to your design specifications and will operate as required by your specifications. For example, indicate the test you will perform to assure that the blowdown capacity of the SRV's is correct.

e. Valve Operability. Provide a description of your surveillance program to monitor the performance of the SRV's during the plant lifetime. Identify the information that will be obtained in this surveillance program and indicate how these data will be utilized to improve the operability of the valves. For example, indicate how this program will reduce the malfunctions that have occurred in operating BWR facilities.

f. Valve Inspection and Overhaul. You state in the FSAR that half of the SRV's will be bench checked and visually inspected every refueling outage. However, depending on operating cycle length, this may result in several years between inspections. Our concern in this matter arises from operating experience which has shown that failure of the SRV's may be caused by exceeding the manufacturer's recommended service life for the internal components of the SRV's or their air actuators. Accordingly, indicate the frequency at which you intend to visually inspect and overhaul those SRV's which function as part of the automatic depressurization system (ADS). Indicate what provisions will be incorporated into the WNP-2 facility to ensure that inspection and overhaul of all the SRV's is in accordance with the manufacturer's

recommendations for the SRV's installed in the WNP-2 facility and that the design service life for any component of the SRV, is not exceeded.

211.52 (5.2.2) Provide the initial values of all system and core parameters assumed in your analysis of pressure transients, including: (1) their nominal operating range; (2) their uncertainties; and (3) the operating limits on these parameters that will be established in the WNP-2 Technical Specifications.

211.53 (5.2.2) In Section 5.2.2.2.4 of the FSAR, you discuss SRV characteristics which include valve groups and pressure setpoints. However, it is not apparent to us how these two items are factored into your analysis. For example, the setpoint range for the spring actuation safety mode is indicated in Section 5.2.2.2.4 as 1165 to 1205 pounds per square inch gauge (psig) whereas Table 5.2-2 lists 1130 to 1205 psig for this range. Define the phrase "valve groups" and indicate how you include consideration of valve groups in your analysis. Discuss how you use these different setpoint values in your analyses.

211.54 (5.2.2) The peak pressures occurring after closure of the MSIV's due to scrams initiated by high flux and high pressure signals are not consistent between Figures 5.2-4 and 5.2-5 of the FSAR. Further, Section 5.2.2.2.3.1 erroneously states that generator load rejection with bypass failure is shown on Figure 5.2-4. Correct these inconsistencies.

211.55 (5.2.2) Indicate whether the WNP-2 facility will incorporate a fast scram system.

211.56 (5.2.2) Provide calculations to support the values you assume for the discharge coefficients and the flow capacities of the SRV's.

211.57 (15.0) Indicate the power-operated pressure relief setpoints and the flow capacities assumed in your transient analyses in Section 15 of the FSAR.

211.58 (6.3) Confirm that adequate net positive suction head (NPSH) will exist if operator action is not initiated within 20 minutes following a postulated loss-of-coolant accident (LOCA). Provide your detailed NPSH calculation to demonstrate conformance with our positions in Section C of Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," November 1970, for the pumps in the emergency core cooling system (ECCS).

211.59 (6.3) You state in the FSAR that no operator action is required until 10 minutes after an accident. However, it is our position that no operator action should be required for 20 minutes after an

accident. Accordingly, discuss the consequences of the reactor operator not performing his required duties until 20 minutes after a postulated LOCA. Discuss all actions which the operator is required to perform to place the plant in the long-term cooling mode following a postulated LOCA.

211.60 (6.3) On page 6.3-10 of your FSAR, you state that the high pressure core spray (HPCS) is automatically shutdown by a signal indicating a high water level in the reactor pressure vessel (RPV). Indicate what provisions are incorporated in the WNP-2 facility to prevent premature termination of the HPCS flow. State whether any interlocks are provided (e.g., a LOCA signal) which would prevent automatic shutoff.

211.61 (6.3) When the water level in the condensate storage tanks (CST) drops to a predetermined level, the HPCS pump switches automatically to the suppression pool. Provide assurance that the water level in the CST will supply an adequate NPSH at the time this switchover occurs. In addition, show that the minimum submergence of the suction piping in the CST will preclude formation of an undesirable vortex. Describe the preoperational testing you will perform to demonstrate that such vortex formation will not occur.

211.62 (6.3) Provide assurance that adequate NPSH exists in the event of a passive failure of the ECCS in a water-tight pump room. Discuss the possibility of vortex formation at the suction intake of the remaining ECCS pumps with the lowered suppression pool level that would result from this type of postulated accident. Discuss the preoperational tests you will perform to demonstrate that there is no impairment of the functional capability of the ECCS due to a lowered suppression pool level.

211.63 (6.3) Confirm that the low pressure coolant injection (LPCI) system does not perform any other function such as containment cooling during the short-term portion of the recovery phase following a postulated LOCA. If the LPCI system will be used for another function during this time period, this additional function must be considered in your LOCA analyses. (Refer to Item 211.82 of this enclosure.)

211.64 Provide the values of the total break area which you assumed for the following postulated breaks: (1) the recirculation line break; (2) the steam line break inside and outside containment; (3) the feedwater line break; and (4) the core injection spray line break.

211.65 (6.3) Indicate the differences between the assumed values of break areas for postulated steam line breaks inside and outside containment. Your analyses of these postulated breaks indicates that the reactor core could become uncovered if no operator action took place within 20 minutes after this postulated accident. Indicate the effect on the peak clad temperature if the operator takes no action for 20 minutes after an accident. In your response, include a discussion of all your assumptions.

211.66 (6.3) Identify all ECCS valves which may be potentially submerged or subject to spray impingement following a postulated LOCA. Discuss the environmental qualification of these valves for these conditions.

211.67 (6.3) Indicate whether there have been any recent changes or corrections to your ECCS analysis. If so, provide the references for the latest model changes and corrections in the list of references provided for your ECCS analysis.

211.68 (6.3) Justify the designation in Table 6.3-3 of the FSAR, of the Zimmer facility as the lead plant for the WNP-2 facility with respect to the LOCA break spectrum analysis. Our concern is that the Zimmer fuel assembly is an 8x8 fuel array wth one water rod while the WNP-2 fuel assembly will be an 8x8, two water rod array.

211.69 (6.3) Correct the small break model curves shown on Figure 6.3-13 of the FSAR for both the failure of the diesel-generator which disables the LPCI and the diesel-generator failure which disables the low pressure core spray (LPCS). Specifically, correct the apparent inconsistency between the values of peak clad temperature (PCT) in Figures 6.3-32 and 6.3-39 of the FSAR and those in Figure 6.3-13 at a break area equal to 80 percent of the break area for the design basis accident (DBA) and at 60 percent of DBA break area.

211.70 (6.3) Demonstrate that a postulated failure of the HPCS in conjunction with a postulated break whose area ranges from 1.0 square foot to the DBA break area is not more limiting than the postulated failure of the diesel-generator which disables the LPCI system over the same range of break areas.

211.71 (6.3) Indicate why the plots of water level versus time for the 1.0 square foot transition break assuming a failure of the HPCS system are different for the small break method and the large break method.

211.72 (6.3) Provide information on applicable tests which demonstrate that the pumps used for long-term cooling, both for normal operation and following a postulated LOCA, will operate effectively during the time period required to fulfill their function.

211.73 (6.3) Table 6.3-5 is not clear. Discuss the intent of the column headed, "Effect on ECCS," with regard to the particular break location; i.e., indicate the postulated break location.

211.74 (6.3) Check valves in the discharge side of the HPCS, the LPCI/RHR and the LPCS systems perform an isolation function since they protect these low pressure systems from the high pressures in the reactor.

We require that: (1) these check valves be classified as ASME IWV-2000, Category AC; and (2) the leak testing for these valves be performed according to the applicable code specifications. You should recognize that a test which simply draws suction on the low pressure side of the outermost check valves, will not be acceptable. Such a test only verifies that one of the check valves in series is fulfilling its isolation function. The required testing frequency is that specified in the ASME Boiler and Pressure Vessel Code, except in those cases where only one or two check valves separate high and low pressure systems. In these cases, we require that you perform leak testing of these valves at each refueling after the valves have been exercised. Accordingly, identify all ECCS check valves which should be classified as Category AC in accordance our position on this matter. Verify that you will perform the required leak testing in accordance with the required frequency and that you have the necessary test lines to leak test each valve. Provide the leak detection criteria that you propose for the WNP-2 Technical Specifications.

211.75 (6.3) Indicate the provisions incorporated in the WNP-2 facility to protect the water level instrumentation for the CST and the lines from this tank leading to the HPCS systems from the effects of cold weather and dust storms. In responding to this item, cross-reference your responses to Items 010.16 and 211.12.

211.76 (6.3) Some of the ECCS relief valve discharge lines penetrate primary containment and have outlets below the surface of the suppression pool. Since these lines are part of the primary containment boundary, we are concerned that excessive dynamic loads resulting from water hammer during actuation of the relief valves may cause cracking or rupture of these lines. Accordingly, identify these lines which penetrate the primary containment. Provide information concerning the measures you are taking to prevent line damage due to water hammer.

211.77 (6.3) Since the ECCS contains both manually operated and motor-operated valves, there is a possibility that manual valves might be left in the wrong position and that this condition will remain undetected when an accident occurs. Accordingly, provide a list of the locations and types of all manually operated valves in the safety-related systems of the WNP-2 facility. For each of these valves, provide a discussion of your procedures to minimize the possibility of an occurrence as described above. We require that you provide indication in the control room for all critical ECCS valves, either manually or motor-operated.

211.78 (6.3) Recent experience at an operating plant identified a potential for a common mode flooding of ECCS equipment rooms. The problem involved the equipment drain lines. (Refer to IE Circular No. 78-06, May 30, 1978 which is attached to this enclosure). Verify that the specific design of the WNP-2 floor and equipment drains is such that flooding in any one room or location, will not result in flooding of redundant ECCS equipment in other rooms. In responding to this item, cross-reference your response to Item 010.28.

211.79 (6.3) The discussion in Section 6.3.2.2.5 of the FSAR regarding the fill system you propose to prevent water hammer resulting from empty discharge lines in the residual heat removal (RHR) system and in the ECCS, is inadequate. Since there have been about fifteen damaging events due to water hammer that resulted from empty discharge lines of the core spray and RHR systems, we are concerned with the adequacy of fill systems, including their associated instrumentation and alarms, to minimize water hammer. Accordingly, respond to the following matters:

a. Provide a detailed description of the fill system, including the associated instrumentation and alarms, with appropriate references to process and instrumentation drawings (P&ID's).

b. Level transmitters apparently are not used to detect trapped air bubbles upstream of injection valves. A pressure level read downstream of a pump discharge check valve, which is greater than the gravity head corresponding to the highest point in the system, does not necessarily indicate the absence of trapped air pockets. Accordingly, indicate what provisions you have made to avoid trapping of air pockets in the lines. In your response, include a discussion of the effect of leaking valves in bypass test lines.

c. If required maintenance on a particular loop (e.g., the RHR system) necessitates draining of this loop, indicate how the fill system protects the other loop and systems; e.g., the containment spray (CS) system.

d. Indicate the surveillance testing which will be required to demonstrate that the fill system instrumentation is capable of performing its function.

e. Indicate how surveillance tests will be made to determine if the discharge lines for the RHR and CS systems are full as will be required in the WNP-2 Technical Specifications.

f. Assuming the jockey pump system does not maintain full lines, water hammer could occur during surveillance tests of the RHR and CS pumps. If damage occurred due to water hammer, the event would be reported in a licensing event report (LER). However, if special fill and vent procedures were used prior to these tests, water hammer would not occur and any inadequacies of the jockey pump system might not be evident. Accordingly, discuss: (1) your procedures for surveillance tests involving startup of RHR and CS pumps; and (2) your reporting procedures if special filling and venting procedures are used and indicate partially empty lines.

211.80 (6.3) It is our position that the ECCS should be designed to provide sufficient capability to cool the reactor in the event of any single active or passive failure in the ECCS during the long-term cooling phase following a postulated accident. However, you have not presented sufficient information in the FSAR to demonstrate that you satisfy our requirement with regard to passive failures. In particular, our position is that you should provide leakage detection and appropriate alarms which would: (1) alert the reactor operator in the event of passive ECCS failures during the long-term cooling phase; and (2) allow the operator sufficient time to identify and isolate the faulted ECCS line. This design feature should satisfy the requirements of IEEE Std 279-1971, except for the single failure requirements. Accordingly, discuss the following considerations:

a. Indicate the assumed maximum leak rate in the ECCS, including a justification for this value.

b. Indicate the maximum allowable time for corrective operator action, including a justification for this time interval.

c. Demonstrate that your leak detection system will be sensitive enough to: (1) initiate, by alarm, operator action; (2) permit identification of the faulted line; and (3) permit isolation of the line prior to a leak creating undesirable consequences such as flooding of redundant equipment. Our position is that the minimum initiation time for operator action for this task is 30 minutes after the alarm.

d. Demonstrate that your leak detection system can identify the faulted ECCS train and that a leak would be isolable.

You should determine the effects on the ECCS of passive failures of such components as pump seals, valve seals, and measuring devices. Your analysis should address the potential for flooding caused by the ECCS and the potential for ECCS inoperability which could result from a depletion of the water inventory in the suppression pool. Your analysis should include consideration of: (1) the flow paths of the radioactive fluid through floor drains, sump pump discharge piping, and the auxiliary building; (2) the operation of the auxiliary systems that would receive the radioactive fluids; and (3) the ability of the leakage detection system to detect a passive failure. Examine the auxiliary system piping in the vicinity of ECCS equipment and address the potential for flooding from nonsafety-grade piping. (Refer to Attachment 1 to this enclosure.)

211.81 (6.3) During the long-term cooling phase following a small break LOCA, the reactor operator must control the primary system pressure to preclude overpressurization of the RPV after it has been cooled down. Accordingly, provide the following information:

a. Describe the instructions which the operator will follow while performing long-term cooling of the plant.

b. Indicate the time frame in which the operator will perform the required actions, including justification for the timing of the operator's actions.

c. List the instrumentation and components needed to perform this action and confirm that these components meet safety grade standards.

d. Discuss the pertinent safety concerns during this cool-down period and indicate the design margins available for each concern.

e. Provide plots of the temperature, pressure, and the water inventory in the reactor coolant system (RCS), showing the important occurrences during this cool-down period.

In your response, account for the following events: (1) a loss of offsite power; (2) an operator error; or (3) a single failure.

211.82 (6.3) Demonstrate that for all sizes of breaks in a recirculation loop or in ECCS lines which would thereby require actuation of the ECCS, the reactor core is sufficiently covered with water so that diversion of the LPCI system to wetwell spray after 10 minutes is acceptable and that the ECCS systems are in compliance with the requirements of Criterion 35 of the General Design Criterion (GDC) and Section 50.46 of 10 CFR Part 50. In your response, indicate what consideration you have given to the full spectrum of potential single failures and to potential break locations. Confirm that no operator action affecting the performance of the ECCS is required prior to 20 minutes after the initiation of the accident.

In particular, discuss the effects of the following matters on cooling of the reactor core and provide information to show that the requirements of GDC 35 and Section 50.46 of 10 CFR Part 50 are not violated.

a. Provide assurance that the system which diverts the LPCI flow meets the single failure criterion so that diversion of the LPCI system less than 10 minutes after a postulated accident, need not be considered.

b. Provide justification for the conclusion that a break in a ECCS line is the most limiting break location when evaluating the effects of a postulated LOCA followed by diversion of the LPCI flow.

c. Provide a sensitivity study of the PCT as a function of break size for small break LOCA's, assuming LPCI diversion will be initiated 10 minutes after the start of the accident. Perform this study for postulated breaks in the ECCS and recirculation lines. For the most limiting break, provide the following

figures: (1) the water level inside the shroud as a function of time following the postulated LOCA; (2) the reactor vessel pressure versus time; (3) the convective heat transfer coeficient versus time; (4) the peak clad temperature versus time; and (5) the ECCS flow rate versus time.

d. Provide assurance that LPCI diversion after 10 minutes will have less severe consequences than diversion at 10 minutes, considering the appropriate break sizes for diversion at times greater than 10 minutes after the accident.

e. Provide a discussion which contrasts the need for LPCI diversion for the limiting break size with the need for abundant core cooling required by GDC 35. For example, this discussion could consider the likelihood of LPCI diversion for the limiting break size.

211.83 (6.3) Provide assurance that the fast closure of a recirculation flow control valve coincident with a LOCA is not expected to occur. Alternatively, provide the results of a sensitivity study which evaluates the effects of a fast closure of a recirculation flow valve coincident with the design basis LOCA and the worst postulated ECCS failure.

211.84 (15.2) Your proposed reclassification of the transients resulting from a generator trip and a turbine trip without bypass, from a frequent to a infrequent event in Section 15.2.2.1.2.2 of the FSAR has not been accepted by us and is still under generic review. Accordingly, reanalyze the events cited above to determine the operational limit on the minimum critical power ratio (MCPR) which would not violate the minimum safe value of 1.06 for the MCPR. It is our position that the limiting transient be reanalyzed with the ODYN code cited in Item 211.49 of this enclosure.

211.85 (15.A) Modify your nuclear safety operational analysis (NSOA) drawings to show the nonsafety-grade equipment for which you take credit to mitigate transients and accidents. Such equipment includes relief valves, turbine bypass valves, and a vessel level trip indicating high water in the RPV (i.e., a Level 8 trip).

211.86 RSP (15.1.2) During recent meetings with GE, we have discussed whether nonsafety-grade equipment can be assumed to function when analyzing anticipated transients. It is our understanding that one of the more limiting events is the failure of the feedwater controller which would result in a maximum flow demand. For this transient, the plant operating equipment which has a significant role in mitigating this event, are: (1) the turbine bypass system; and (2) the reactor vessel high water level trip (Level 8) that closes the turbine stop valves. To assure an acceptable level of performance, it is our position that the availability, the setpoints and the surveillance testing of this equipment be identified in the WNP-2 Technical Specifications.

Accordingly, submit your plans for implementing this requirement along with any system modifications that may be required to satisfy our requirements in this matter.

211.87 (15.1) It is not evident to us that the drop of 100° Fahrenheit which you assume in the feedwater temperature results in a conservative evaluation of the cold feedwater transient when the recirculation flow is manually controlled. For example, a feedwater temperature drop of about 150° Fahrenheit occurred at an operating BWR in this country as a result of a single failure of an electrical component. The electrical equipment malfunction which was a break-trip of a motor control center, caused a complete loss of all feedwater heating due to a total loss of extraction steam. Accordingly, submit: (1) a sufficiently detailed failure modes and effects analysis to demonstrate the conservatism of the 100° Fahrenheit feedwater temperature drop you assume considering the potential effects of any single electrical malfunction; or (2) calculations using a limiting feedwater temperature drop which clearly bounds current operating experience.

Further, reductions in feedwater temperature less than 100° Fahrenheit can occur which would represent more realistic (i.e., slower) changes in feedwater temperature with time. In particular, slow transients with the surface heat flux in equilibrium with the reactor power when the reactor scrams due to a feedwater temperature drop smaller than 100° Fahrenheit, could result in a larger change in the critical power ratio (CPR). Accordingly, evaluate the cold feedwater transient for all sequences of events that can cause a slow transient and demonstrate the conservatism of the values of the feedwater temperature drops, including the rate of change with respect to time, which you assume in your present transient analysis.

211.88 (15.2) In your evaluation of the generator load rejection transient, you assume 0.15 seconds for the full stroke closure time of the turbine control valve and state that it is conservative compared to an actual closure time of 0.2 seconds. However, in Table 15.2-2 of the FSAR, you indicate that the turbine control valves close in 0.07 seconds. Explain this apparent discrepancy. Additionally, the pressure peaks caused by closure times from the partially open to the fully closed position are not addressed in the FSAR. For full-stroke closure, the closure time you assume appears to be conservative in light of the information in the FSAR. However, for operation in the full arc (i.e., full throttling) mode, the closure times may be significantly less than 0.15 seconds for typical cases where the control valves are only partially open. We have two concerns with respect to this particular transient. Our first concern is that the minimum closure times for part-stroke may be less than those you assumed in your analysis. Our second concern is that your analysis, which is based on initial conditions which include 105 percent, nuclear boiler rated, steam flow and the control valves wide open, may result in a less conservative

evaluation than the initial conditions at a somewhat lower power with the control valves partially open. Accordingly, demonstrate that control valve closure times smaller than 0.15 seconds do not result in unacceptable increases in the ΔMCPR and in the reactor peak pressure. Alternatively, either provide justification that shorter closure times cannot occur or indicate a minimum closure time to be incorporated into the WNP-2 Technical Specifications.

211.89 (15.1.1) For the transient resulting from a loss of feedwater heating while in the manual flow control mode, the thermal power monitor (TPM) is used to scram the reactor. Explain the need for the TPM and indicate the specific transients for which this trip signal initiates a reactor scram. Describe the surveillance testing of the TPM which will be incorporated into the WNP-2 Technical Specifications.

211.90 (15.4) Provide assurance that the plots of pressure with time in Section 15 of the FSAR are consistent with the initiation logic for the SRV's. For example, you may have modified the safety/relief system to prevent subsequent reopening of these valves during transients involving an increase in the reactor pressure to satisfy your present design bases for pool dynamic loads in the containment.

211.91 (15.4.5) Provide the initial operating MCPR determined at 56 percent of rated power (nuclear boiler) and 36 percent of the core flow for the postulated failure of the recirculation flow control system while undergoing an increasing flow transient. In addition, provide the K_f* factors as a function of the core flow for both the automatic and manual flow control modes of operation. Provide the maximum flow control setpoint calibration limit (e.g., 100 percent or 105 percent of rated flow) for the recirculation loop flow control valves used in the transient analysis. Additionally, we note that you reference the GE topical report, NEDO-10802, for the dynamic model which you used to simulate this event. However, NEDO-10802 does not describe the complete event. Accordingly, discuss in greater detail the overall method you used to calculate the change in the CPR.

211.92 (15.3.3) In Table 15.3-5 of the FSAR, you take credit for nonsafety-grade equipment to terminate the postulated accident involving seizure of the recirculation pump. However, it is our position (refer to Section 15.3.3, Revision 1, NUREG-75/087 of the Standard Review Plan) that only safety-grade equipment can be used and that the required safety functions must be accomplished assuming the worst single failure of an active component. Accordingly, reevaluate this accident with the specific criteria cited above. Indicate the resulting change in the CPR and the percentage of fuel rods which would be in boiling transition for this postulated accident.

*K_f is defined as the ratio of the MCPR at a given reactor coolant flow rate to the MCPR at 100% power (i.e., 1.20).

211.93 (15.1.2) For the transient resulting from a postulated failure of the feedwater controller during maximum flow demand, you indicate a feedwater flow of 146 percent in Table 15-1-3 of the FSAR. However, you indicate in Section 15.1.2.3.2 that the feedwater flow is 135 percent for the maximum flow setting in simulating this transient. Clarify this apparent discrepancy.

211.94 (15.1.2) When a sudden increase in feedwater flow occurs, there will be a corresponding drop in the feedwater temperature which contributes to the reactivity increase during the first part of this transient. For example, the combination of a drop in the feedwater temperature and a smaller maximum flow rate could cause a Level 8 trip with the surface heat flux close to the flux scram setpoint. If you have assumed that the feedwater temperature into the reactor vessel has remained constant, reanalyze this transient to include the effect of the variation in the feedwater temperature on the MCPR. Provide your basis for determining the time variation in the feedwater temperature at the reactor vessel. Demonstrate that a smaller increase in the feedwater flow rate than the one you analyzed, in conjunction with the change in feedwater temperature, does not result in a lower MCPR.

211.95 (15.1.4) In your analysis of an inadvertent opening of an SRV in Section 15.1.4.2.1.1 of the FSAR, you state that a plant shutdown "should" be initiated if the valve cannot be closed. Indicate how much time the operator has to initiate plant shutdown before exceeding the proposed WNP-2 Technical Specification limits for the suppression pool temperature.

211.96 (15.2.6) You indicate in your analysis of the transient resulting from a postulated loss of off-site power that closure of the MSIV's occurs at 30 seconds after the start of the transient due to a loss of condenser vacuum. Our concern in this matter is that the MSIV's may close at an earlier time in the transient, thereby causing higher system pressures than your analysis indicates. Apparently, you take credit for operation of the MSIV air accumulator since the normal air supply to the MSIV's would trip at the start of this particular transient. Discuss the design provisions incorporated into the WNP-2 facility which prevent closure of the MSIV's any earlier than 30 seconds after the start of this transient. Additionally, discuss your verification testing which will demonstrate that the MSIV performance assumed in your analysis, will be achieved.

211.97 We have a similar concern to that stated above regarding the potential for MSIV closures times that may be shorter than those assumed in your analyses of the transient resulting from a loss of off-site power since this loss of power could generate an isolation signal that would close the MSIV's. Indicate the sources of electrical power for the MSIV isolation logic and the isolation actuators. State whether these power sources would be available following a loss of off-site power. Indicate

whether the MSIV isolation logic and the isolation actuators could fail in a manner which would initiate an MSIV isolation signal on loss of off-site power.

211.98 (15.0) We are concerned that operation of the WNP-2 facility with partial feedwater heating might occur during routine maintenance or as a result of a decision on your part to operate with a lower feedwater temperature near the end of a fuel cycle. Demonstrate that this mode of operation will not result in: (1) maximum reactor vessel pressures greater than those obtained using the assumptions in Section 5.2.2 of the FSAR; or (2) a more limiting change in the MCPR than would be obtained with the assumptions used in Section 15.0. Provide the basis for the maximum reduction in feedwater heating considered in your response to this item (e.g., the specific limitations on the turbine operation).

211.99 (7.5) Since systems such as the HPCS, HPCI, and RCIC are initially aligned to draw coolant water from the CST and switch to the suppression pool following a signal indicating a low water level in the CST, it is our position that the CST water level should be included in Table 7.5-1 of the FSAR, entitled "Safety-Related Display Instrumentation." Accordingly, add the signal indicating low water level in the CST in Table 7.5-1. Alternatively, justify its omission.

211.100 (7.5) Identify which parameters are used to monitor the plant conditions following an accident and which are input to the safety-related display instrumentation shown in Table 7.5-1 of the FSAR, .

211.101 (7.5) In Table 7.5-1 of the FSAR, you identify the range of the instrument which monitors the reactor vessel pressure to be from 0 to 1500 psig. Since the design pressure of the reactor coolant pressure boundary is 1250 psig, justify the upper bound of this instrument range in light of the potential transients and accidents that may cause large pressure excursions (i.e., ATWS).

211.102 (7.4.1) Provide display instrumentation indicating the water level in the CST on the remote shutdown control panel. You state in the FSAR that the RHR flow indicator will be located on the remote shutdown panel. Verify that flow indication will be provided for both RHR systems (i.e., A and B) and that the flow range will be the same as that shown in Table 7.5-1 of the FSAR.

211.104 (9.2.7) In Table 9.2-5 of the FSAR, you show a flow rate of 7400 gallons per minute (gpm) from the standby service water system to the RHR heat exchanger. This flow rate is based on an inlet temperature of 95° Fahrenheit. However, in Section 5.4.7.2.2, the service water side flow rate of 7400 gpm to the RHR heat exchanger is based on a rated inlet temperature of 85° Fahrenheit. Explain this apparent discrepancy. Additionally, demonstrate that you have adequately selected the required flow rates for the standby service water system for heat load removal from the ECCS pumps

as shown in Table 9.2-5 of the FSAR. Provide justification for these flow rates, including a list of the design duty heat loads for the equipment identified in Table 9.2-5.

211.104 (9.2) Provide a table listing the standby service water system cooling duty loads as a function of the time intervals listed below following a postulated DBA. In this table, indicate the operating status of the appropriate safety-related equipment (e.g., the RHR pumps, the RHR heat exchangers, the CS pumps, the ADS valves, and the RCIC). The time intervals for this tabulation should be: (1) 0 to 10 minutes; (2) 10 to 30 minutes; (3) 30 minutes to 6 hours; (4) 6 hours to 24 hours; and (5) 24 hours to 30 days.

211.105 (3.9.1) Provide the following information related to the contents of Table 3.9-1 of the FSAR. This table shows the number of plant cycles or events considered for reactor assembly design and fatigue analysis.

a. Discuss the events contained in Item i for normal, upset and testing conditions and relate these to the transients analyzed in Section 15.0 of the FSAR. In particular, discuss the following events:

(1) The number of cycles (i.e., eight cycles) for the 40 year life of the WNP-2 facility shown in Table 3.9-1 of the FSAR (i.e., Item i.4) for a single safety or relief valve blowdown for upset conditions, appears to be low. Specifically, we note that Table 15.0 of the FSAR indicates that these valves will lift for a variety of transient events and that more than one valve will blow down. Accordingly, provide justification for your design basis of eight cycles.

(2) Clarify whether the loss of feedwater pumps in Item i.3 is due to MSIV closure or whether both of these events occur independently. For either case, the number of cycles (i.e., ten cycles) which you state for the 40-year life of the WNP-2 facility, appears to be low. In particular, since a number of transients can cause a trip of the feedwater pumps and close the MSIV's, more than ten events causing the above conditions can be anticipated throughout the plant lifetime. Accordingly, justify your design basis of ten cycles for this event.

b. Indicate whether Item 1(2) for emergency conditions in Table 3.9-1 of the FSAR is the automatic blowdown feature related to the ADS function.

c. Explain Item 1(2) for emergency conditions and relate it to your analysis in Sections 5.2.2 or 15.0 of the FSAR. Justify

your omission of the event in which the reactor is overpressurized, there is a scram initiated by a high flux signal and the isolation valves stay closed under "emergency conditions."

211.106 (15.0) Provide the correct units (or value) for the recirculation pump trip inertia for Item 32 of Table 15.0-2 of the FSAR.