March 2, 1988

Docket No. 50-601

Mr. W. J. Johnson
Nuclear Safety Department
Westinghouse Electric Corporation
Water Reactor Division
Box 355
Pittsburgh, PA 15230

Dear Mr. Johnson:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION ON RESAR SP/90

As a result of our ongoing review of the RESAR SP/90 PDA application, we require additional information in order to complete our review of the reactor systems aspects of the design. Enclosed are review questions Q440.242-440.262.

Please respond to this request within 60 days of the date of this letter. If you have any questions regarding this matter, call me at (301) 492-1120.

Sincerely,

Original Signed By:

Thomas J. Kenyon, Project Manager
Standardization and Non-Power
Reactor Project Directorate
Division of Reactor Projects III, IV,
V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure: As stated
cc: See next page
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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Enclosure: As stated
cc: See next page

cc: Brookhaven National Laboratory
Building 130
Upton, New York 11973
Attention: Dr. Robert Bari

ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION ON THE PDA APPLICATION FOR
WESTINGHOUSE ADVANCED PRESSURIZED WATER REACTOR (WAPWR)
DOCKET NUMBER 50-601
RESAR-SP/90

440.242 (Volume 1, Page 3-32) Discuss the safety design classification of the pressurizer heaters. If they are not designed to safety grade standards, confirm that the reactor can be brought to cold shutdown conditions without the operation of pressurizer heaters. BTP RSB 5-1 requires that the plant be able to reach cold shutdown conditions using only safety grade equipment.

440.243 (Volume 1, page 3-11) You have indicated that Westinghouse will outline a program for emergency response guideline development prior to receiving a PDA for the WAPWR design. Provide the above stated outline for the staff review.

440.244 (Volume 1, page 3-42) You have stated that the feed and bleed mode for reactor coolant system operation can be used to remove decay heat from the reactor. Has any thermal-hydraulic analysis been performed to confirm the viability of the feed and bleed process based on the plant configuration of the WAPWR? If so, please submit the results of the analysis.

440.245 THIS QUESTION INTENTIONALLY LEFT BLANK.

440.246 (Volume 1, page 4-35) You have stated that the current design basis is to be able to accommodate a loss of all ac power for a minimum of 2 hours with an ultimate goal of 10 hours. Please explain how the ultimate goal of 10 hours station blackout could be achieved if the current design has only 2 hours capability for a complete loss of ac power.

440.247 (Volume 1, page 4-38) You have stated that the primary side and secondary side safeguards system options being considered for inclusion in the WAPWR design provide the capability of removing decay heat from the reactor core while maintaining sufficient water inventory to ensure adequate core cooling. Confirm that all systems and components used to perform the above stated function will be designed to safety grade standards.

440.248 (Volume 1, page 5-21) Expand your discussion on the design criteria for the main steam and main feedwater isolation valves. Confirm that these valves will be designed to safety grade standards or that the transient and accident analyses will not give credit to these valves for performing their safety related function.

440.249 THIS QUESTION INTENTIONALLY LEFT BLANK.

440.250 (Volume 1, page 5-80) You have stated that passive failures which are considered to have a low probability may not be considered. Please discuss the criteria for identifying the passive failures with low probability.

440.251 (Volume 1, item 21) Per the requirements of RTP RSB 5-1, please confirm that a boron mixing and natural circulation test will be performed in the first plant with WAPWR design.

440.252 (Module 1, Section 6.3.2) You stated that the ECCS pumps are protected against low flow or no flow-operation by the miniflow path. It is not clear that how pump protection could be achieved by the miniflow lines under no flow or low suction pressure conditions. It is the staff's position that the ECCS pump protection should be provided by a safety grade low flow alarm system and to assure that the pump could withstand those operating conditions during the time delay for operator actions to manually trip the pump in response to the alarms.

440.253 (Module 1, Section 15.6.4) The Westinghouse LOCA evaluation model approved by the staff may not be applicable to WAPWR design with respect to plant specific configurations in node arrangement and control systems. Confirm that a new LOCA evaluation model will be prepared for the WAPWR design.

440.254 (Module 3, Section 1.1.1.2) You stated that the WAPWR design includes a NSSS with a thermal rating of 3816 megawatts, which includes a core thermal power of 3800 megawatts plus 16 megawatts from the reactor coolant pump heat. Are the primary coolant heat losses included in calculating the NSSS thermal rating? If not, why not?

440.255 (Module 4, Section 5.2.2) Section 5.2.2 on page 5.2-3 states that the liquid relief valves of the residual heat removal system (RHRS) are used to protect RCS at low temperatures when the RHRS is in operation. Section 5.2.2.10 states that the pressurizer PORVs will be used for the low temperature overpressure protection (LTOP) function. Please clarify the LTOP design for the WAPWR.

440.256 (Module 4, Section 5.2.2) Expand this section to address the assumptions used for a mass addition event relative to the LTOP system design.

440.257 (Module 4, page 5.2-11) Item 4A states that to preclude inadvertent ECCS actuation during heatup and cooldown, blockage of the safety injection signal actuation logic below 1975 psia is required. Discuss the impact of this design relative to a LOCA during modes 3 and 4.

440.258 (Chapter 15) For transient and accident analyses of WAPWR, provide the following:

a) A list of all transient and accident analyses cross referencing the modules of the PDA application where each event is addressed.

b) Acceptance criteria for each transient and accident analyzed should be clearly stated.

c) Initial conditions for each event including consideration of all modes of plant operation.

d) For each event, include the best estimate analysis using realistic plant data and emergency operating procedures and the licensing analysis using most conservative assumptions and only safety grade equipment for event mitigation.

e) Identify the most limiting single failure used for the analysis of each event with respect to different acceptance criteria of the event (e.g. Peak RCS pressure, DNB, radiological consequences).

440.259 (Module 9, page 15.0-3) Item 2a indicates that plants may be operated at power with a rector coolant pump out of service. You should provide analyses for N-1 loop operation to support the WAPWR design.

440.260 (Module 9, page 15.0-9) Discuss why the nominal plant operating parameters are assumed for the analyses of transients and accidents which are DNB limited. For the events leading to increase of RCS pressure, why isn't a higher RCS pressure assumed as the initial condition. For each event, with respect to peak RCS pressure or fuel performance, it is nessary to assure different initial conditions in the analyses in order to predict the worst consequences of the event.

440.261 (a) Generic Letter (GL) 87-12 requested information regarding lowered RCS inventory operation. Please provide a response to the generic letter with respect to the PESAR-SP/90.

(b) Please describe instrumentation provided to the operator during shutdown operations which characterize the state of the reactor coolant system (RCS). Include RCS level, RCS temperature, and residual heat removal (RHR) system performance and provide a description of the appropriateness and accuracy of each instrument with respect to its intended function. Also, include identification of audible and visual alarms used to delineate out-of-range conditions, including the values which constitute those conditions.

(c) The staff has identified that Diablo Canyon, Unit 2, was in a condition not previously analyzed by the NRC staff during the loss of RHR event of April 10, 1987 (NUREG-1269). Please describe the steps that have been taken and the future plans which will be taken to alleviate this situation for the SP/90.

(d) NUREG-1269 contains the statement "Design of the nuclear steam supply system (NSSS) did not appear to provide detail provisions for mid-loop operation." Please address this identified deficiency in PWR design with respect to the SP/90

design. Include identification of and discussion of each of the design changes in the SP/90 which represents and improvement over existing designs and establish the adequacy of the SP/90 design for lowered RCS inventory operation.

(e) NUREG-1269 identified that containment was open throughout the April 10, 1987 event, and there were no procedures to reasonably assure containment closure in the event of progression of the accident to a core damage condition. Address this situation with respect to the SP/90 design and the anticipated methods that will be used to operate the plant. Include such design considerations as the need for removal of the equipment hatch and improvements in the SP/90 design which facilitate rapid replacement of the hatch should the need arise. Similarly address other containment penetrations and potential bypass paths.

(f) The Diablo Canyon event and subsequently obtained information has shown operating procedures to be inadequate for lowered RCS inventory operation. What plans exist for recommending improved procedures and administrative controls to SP/90 owners/operators so that this situation is eliminated in the SP/90.

(g) What equipment exists in the SP/90 that can be used to assure adequate core cooling in the event of a complete loss of RHR?

(h) Evidence exists that certain Technical Specifications (TSs) may not be optimum when consideration is given to operation during non-power conditions. For example, requirements for RHR

suction valve interlocks impact upon RHR reliability, RHR flow rate requirements may overly restrict flow rate range and increase the likelihood of loss of RHR due to vortexing, and TSs written on the basis of time (such as one may remove RHR from operation for an hour) perhaps are more reasonable when written on the basis of the state of the NSSS and/or of containment. Please address this topic with respect to the SP/90 design and provide recommendations for improvement, particularly with respect to the unique design aspects of the SP/90.

(i) Safety analysis reports (SARs) typically concentrate on power operation when consideration is given to many of the potential operational transients. The recent experience from the Daiblo Canyon event indicated that further evaluation for plant operation at lower modes may be required. Hence, it may be prudent to address non-power operation in more depth than has been traditional. What plans exist, if any, with respect to this topic and the SP/90 program?

440.262 Our review has identified several areas in which unique aspects of the SP/90 design do not appear to have been exploited to achieve the maximum reasonable safety. These include:

(a) The diesel start and loading time requirements of a few seconds do not appear necessary with the SP/90 ECCS design. The staff believes that longer start times will enhance safety by reduction of stress and wear to the diesels. Please discuss why such short loading time are necessary.

(b) The four train primary side safeguards system was originally conceived, with one option, as having one diesel with each system. What are the quantitative difference in plant cost and safety when this is changed to the present two diesel design. Please also address the possibility that a four diesel approach may offer a diverse diesel design possibility that has not been included in the two diesel concept.

(c) Please address the use of four diesels of diverse design and with relaxed start and load time requirements with respect to the fraction of severe accidents associated with loss of all ac power.

(d) Early conceptual design of the RCS included large diameter connections which could be used for rapid depressurization. Why was this capability removed and what is the impact of the change on accident mitigation and upon risk?

(e) The containment design may allow cooling via a few nozzles which direct water onto the outside containment surface. Was consideration given to such a system of pre-installed piping and nozzles with a connection which could be used, for example, by a fire truck as a source of pumped water? If not, what would be the cost and impact upon safety if such a system were installed?

(f) Early versions of the SP/90 design included a non-safety related "pump-house" for each of the primary side safeguards systems. This appeared to offer many advantages over the present design under severe accident

conditions and for control of release outside containment under a wide range of conditions. What is the cost differential (details please) and impact upon both safety and releases between the early concept and the present design?