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

April 6, 1994

Docket No. 52-004

Mr. Patrick W. Marriott, Manager
Advanced Plant Technologies
GE Nuclear Energy
175 Curtner Avenue
San Jose, California 95125

Dear Mr. Marriott:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING THE SIMPLIFIED BOILING WATER REACTOR (SBWR) DESIGN (Q440.7-Q440.58)

The staff has determined that it needs additional information to support its review activities related to the SBWR design certification. Some additional information on the material contained in Chapters 1, 3, 4, and 5 of the SBWR standard safety analysis report is needed (Q440.7-Q440.58).* Please respond to the enclosed questions within 90 days of the date of this letter.

You have previously requested that portions of the information submitted in the August 1992, application for design certification of the SBWR plant, as supplemented in February 1993, be exempt from mandatory public disclosure. The staff has not completed its review of your request in accordance with the requirements of 10 CFR 2.790; therefore, that portion of the submitted information is being withheld from public disclosure pending the staff's final determination. The staff concludes that this RAI does not contain those portions of the information for which you are seeking exemption. However, the staff will withhold this letter from public disclosure for 30 calendar days from the date of this letter to allow GE Nuclear Energy the opportunity to verify the staff's conclusions. If, after that time, you do not request that all or portions of the information in the enclosure be withheld from public disclosure in accordance with 10 CFR 2.790, this letter will be placed in the NRC's Public Document Room.

*The numbers in parentheses designate the tracking numbers assigned to the questions.

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This request for additional information affects nine or fewer respondents, and therefore, is not subject to review by the Office of Management and Budget under P.L. 96-511.

If you have any questions regarding this matter, please contact me at (301) 504-1178 or Mr. Son Ninh at (301) 925-1125.

Sincerely,

(Original signed by)

Melinda Malloy, Project Manager
Standardization Project Directorate
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Office of Nuclear Reactor Regulation

Enclosure:
RAI on the SBWR Design

cc w/enclosure:
See next page

Distribution (w/enclosure):

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*To be held for 30 days

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DATE	04/6/94	04/6/94	04/6/94	04/6/94

OFFICIAL DOCUMENT NAME: SBWR9423.MM

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REQUEST FOR ADDITIONAL INFORMATION (RAI) ON THE
SIMPLIFIED BOILING WATER REACTOR (SBWR) DESIGN

Reactor Systems (Chapters 1, 3, 4, and 5)

440.7 In response to the staff question on the issue of regulatory treatment of non-safety systems (RTNSS), GE Nuclear Energy (GE) stated that "Any forthcoming regulatory requirements regarding the capability and availability of the SBWR passive safety and active non-safety systems will be reviewed by GE, and compliance to these requirements will be assessed and submitted for NRC review." The regulatory requirements are identified in a draft Commission paper dated September 7, 1993. Electric Power Research Institute letter dated May 26, 1993, also describes the process for identifying the important non-safety systems as well as regulatory oversight. GE needs to submit the SBWR RTNSS assessment for staff review.

440.8 Unlike the advanced boiling water reactor (ABWR), in Section 1.2.2.2.6 of the SBWR standard safety analysis report (SSAR), the remote shutdown system is classified as a non-safety-related system. Why is the system classified as a non-safety related system?

440.9 In response to SRXB.1, GE stated that functional task analyses were performed for each identified system to establish the controls and indicators needed for remote shutdown system (RSS) control. Explain in detail why the systems traditionally included in the RSS panel are not needed for SBWR. Submit the functional task analysis performed by GE to establish the instrumentation and control for the RSS.

440.10 In SSAR Section 1.2.2.2.7, Reactor Protection System, why are there no scrams on turbine stop valve and control valve closures? In the ABWR reactor protection system, scrams are initiated on turbine stop valve and control valve closures. The SBWR design does not provide a scram upon rapid core flow decrease, nor on main steam line high radiation. It is understood that for the SBWR, which runs on natural circulation, rapid core flow decrease will be unlikely. Explain in detail why the scrams on turbine stop valve and control valve closures are not required for SBWR.

440.11 Regarding SSAR Section 1.2.2.4.3, the GE response to SRXB.5 is not satisfactory. Why is it not possible to pipe the depressurization valve discharge to the suppression pool? Why is diversity in initiation of emergency core cooling systems not necessary for the SBWR?

440.12 For SSAR Section 1.2.2.6.1 (page 1.2-56), is the following reactor coolant temperature reduction schedule based on two trains of reactor water cleanup/shutdown cooling system (RWCU/SDC) and isolation condenser system (ICS) or on one train: 140 °F in 24 hours, 130 °F in 40 hours, and 120 °F at the completion of flooding?

Enclosure

440.13 In SSAR Section 1.2.2.8.1 (Nuclear Fuel), delete the following statement which is not correct: "Fuel design for the SBWR Standard Plant is not within the scope of the certified design."

In SSAR Section 1.2.2.8.2 (Fuel Channel), delete the following statement which is not correct: "Fuel channel design for the SBWR Standard Plant is not within the scope of the certified design."

In SSAR Section 1.2.2.8.3 (Control Rod), delete the following statement which is not correct: "Control rod design for the SBWR Standard Plant is not within the scope of the certified design."

440.14 SSAR Section 1.2.2.10.6 (Turbine Bypass System) states: "The TBS does not serve or support any safety-related function and has no safety design." In the ABWR design, credit is taken in the transient analysis for the turbine bypass system (TBS) and the system is included in the technical specifications. Confirm that no credit is taken for the SBWR TBS in the transient analysis.

440.15 In SSAR Section 3.1.3.7 (page 3.1-30), Criteria 26, Reactivity Control System Redundancy and Capability, and SSAR Section 3.1.3.8 (page 3.1-31), Criteria 27, Combined Reactivity Control Systems Capability, add Reference 9.3.5, Standby Liquid Control System, to the list of SSAR chapters/sections.

440.16 GE's response to RAI SRXB.7 was not satisfactory. The staff does not agree with the GE statement that "There are no requirements for a safety-related high pressure injection system . . . ." General Design Criterion (GDC) 33 of 10 CFR 50 Appendix A requires a safety-related system for protection against small breaks in the reactor coolant pressure boundary and the SBWR design does not include a safety-related injection system to satisfy the GDC 33. (Reference SSAR Section 3.1.4.4, Reactor Coolant Make Up.)

440.17 In SSAR Table 3.2-1 (page 3.2-16), J11, add control rods to the list.

440.18 In SSAR Appendix 4A, describe the power distribution strategy for the SBWR. In Section 4A of the ABWR SSAR, the power distribution strategy is discussed and reference is given to NEDO-20953A, "Three Dimensional BWR Core Simulator," January 1977. Confirm that this document is not applicable for SBWR.

440.19 In SSAR Appendix 4A, describe the control strategy for a typical SBWR plant. In Section 4A of the ABWR SSAR, basic control strategy is given in Table 4A-1.

440.20 In SSAR Sections 4B.1 and 4C.1, add the following: "Any change to these criteria must have prior NRC review and approval."

440.21 In SSAR Appendix 4C, Control Rod Licensing Acceptance Criteria, describe the surveillance criteria for the SBWR. Section 4C.3.5 of the ABWR SSAR gives surveillance criteria for the ABWR.

440.22 In the SBWR fine motion control rod drive (CRD) system, the scram discharge volume is diverted to the reactor, rather than to the scram discharge volume pipe as in current BWRs. There is a concern that if the reactor is at high pressure during an anticipated transient without scram, there may not be sufficient differential pressure to insert the CRD into the reactor. Explain, in detail, why this is not a concern in the SBWR. Confirm that there is sufficient differential pressure to insert the CRD into the reactor for SBWR. (Reference SSAR Section 4.6.)

440.23 In the old CRD system, the major function of the cooling water was to cool the drive mechanism and its seals to preclude damage resulting from long-term exposure to reactor temperatures. What is the function of purge water flow to the SBWR control rod drives? (Reference SSAR Section 4.6)

440.24 Confirm that the motors for the fine motion control rod drives are Class 1E. If they are not, explain in detail why they need not be Class 1E. (Reference SSAR Section 4.6)

440.25 Control rod drive replacement for the SBWR is similar to current BWRs, and will use the same maintenance procedures. The CRD is withdrawn to the point where the CRD blade back seats onto the CRD guide tube. This provides a metal-to-metal seal that minimizes the reactor pressure vessel (RPV) water drainage when the CRD is removed. An unisolable loss-of-coolant accident (LOCA) with an opening of about two inches exists at the bottom of the RPV if the CRD blade and drive are simultaneously removed due to operator failure to follow the procedures. Discuss in detail the design and procedures incorporated in the SBWR design to satisfy the staff concerns. (Reference SSAR Section 4.6.)

440.26 In the ABWR design, the control rod drive pump suction design pressure was increased to 410 psig to satisfy the intersystem LOCA (ISLOCA) concerns. Describe the SBWR CRD system design features incorporated to mitigate the ISLOCA concerns. (Reference SSAR Figure 21.4.6-2, Sheet 2, Design Pressure/Temperature Table.)

440.27 SSAR Section 4.6.1.2.6 (page 4.6-21), Instrumentation and Control, states "When in the high pressure makeup mode of operation, the CRD pumps are tripped to terminate CRD system flow when the level in any two of the three GDCS [Gravity-Driven Cooling System] pools has dropped 1.64 feet below the normal level." Since the CRD pumps do not take suction from the GDCS pool, why are they tripped on low GDCS pool level?

440.28 SSAR Section 4.6.2.1, Failure Mode and Effects Analysis (FMEA), refers to Appendix 1B. Appendix 1B does not include the FMEA. Provide the FMEA for staff review.

440.29 In SSAR Figure 21.4.6-2, Sheet 1, identify the flow control valve F010.

440.30 Confirm that the standby liquid control system and fine motion control rod drive system are located in different parts of the reactor building and are not vulnerable to common-mode failures.

440.31 Why is the most limiting transient for the SBWR overpressure protection analysis the turbine trip/load rejection with failure of the bypass valves and not main steam isolation valve closure like in current BWRs and the ABWR? (Reference SSAR Section 5.2.2.3.2, Transients.)

440.32 What is meant by "overshoot the relief valve setpoint" in SSAR Section 5.2.2.3.3, Safety/Relief Valve Capacity? Figure 5.2-6 shows that peak vessel pressure is independent of valve capacity. How can the peak pressure be completely independent of the valve capacity? How many (minimum number of) safety-relief valves are required to meet the ASME Boiler and Pressure Vessel Code limit of 1375 psig? Explain in detail the significance of the operation of low and high set point valves shown in the figure.

440.33 In SSAR Figure 5.2-1, add units to all the numerical values shown in the figure.

440.34 In SSAR Figure 5.2-5, what is "Simulated Thermal" and "turbine flow" given in the some of the plots? Figures 5.2-4b and 5.2-5b are not readable. What is "deg.C/10" in Figure 5.2-4b? What parameter do the x-axis values 16, 32, etc., represent in some of the plots of Figure 5.2-5?

440.35 Why is the automatic power-actuated pressure relief function, which is part of the ABWR design, not required for the SBWR?

Reactor Systems - Isolation Condenser System (SSAR Section 5.4.6)

440.36 Provide a diagram showing the isolation condenser system (ICS) design parameters: pressures, temperatures, and flow rates. Submit the process flow diagram for the ICS.

440.37 The isolation condenser for the SBWR is a vertical heat exchanger which is significantly different from the IC in operating plants. Provide a detailed description and drawing of the SBWR IC.

440.38 Specify the codes and standards used for the design and fabrication of the SBWR isolation condenser.

440.39 Discuss isolation condenser system operation following a loss-of-power event in the safety design bases.

440.40 Explain in detail how the isolation condenser system meets GDC 4 of 10 CFR 50 Appendix A, as it relates to dynamic effects associated with flow instabilities and loads (e.g., water hammer). Explain the design features and operating procedures designed to reduce water hammer due to mechanisms such as voided discharge lines, steam bubble collapse, and water entrainment in steam lines.

440.41 Explain in detail how the isolation condenser system meets GDC 29 of 10 CFR Part 50 Appendix A as it relates to the system being designed to have an extremely high probability of performing its safety function in the event of anticipated operational occurrences.

440.42 SSAR Section 3.1.4.4, Criterion 33, Reactor Coolant Make Up, states that the requirements of Criterion 33 are met with the isolation condenser system. How can GE take credit for the ICS meeting GDC 33 of 10 CFR Part 50 Appendix A when the ICS is not a reactor makeup system?

440.43 Explain in detail how the isolation condenser system meets GDC 34 of 10 CFR Part 50 Appendix A as it relates to the system design being capable of removing fission product decay heat and other residual heat from the reactor core to preclude fuel damage or reactor coolant pressure boundary overpressurization.

440.44 Confirm that the isolation condenser system meets GDC 54 and GDC 55 of 10 CFR Part 50 Appendix A.

440.45 Confirm that the isolation condenser system is designed to seismic Category I standards.

440.46 Confirm that the isolation condenser system is protected against natural phenomena, external or internal missiles, pipe whip, and jet impingement forces.

440.47 Confirm that the isolation condenser system operation is independent of ac power. How long can the systems be operated? What is the capacity of the power supply?

440.48 Explain how the nitrogen rotary motor operator (NMO) and the nitrogen piston-operated valves work. Why are the NMOs required only for normally kept-open valves F001 and F004?

440.49 In SSAR Section 1A.2.23 (regarding TMI-2 Action Plan Item II.K.3.15, Modify Break-Detection Logic to Prevent Spurious Isolation of HPCI and RCIC), GE claims that although the isolation condenser system uses differential pressure transmitters to detect a possible pipe break, this TMI-2 action item is not applicable to the SBWR. The staff disagrees. Since the SBWR uses the differential pressure transmitters and there is a potential for inadvertent system isolation, the issue is applicable for the SBWR and it should be addressed in the SSAR.

440.50 SSAR Section 5.4.6.3 states "The ICS valve actuators are to be qualified for service inside the drywell for continuous service under normal conditions and to be operable for 4 hours with a steam environment." What is the basis for the 4 hours?

440.51 SSAR Section 1.2.2.4.2 states "The heat rejection process can be continued indefinitely by replenishing the IC/PCC pool inventory." Specify the duration and time required to connect the post-LOCA pool water makeup connections located just above grade level outside the reactor building. Where does this water supply come from?

440.52 In SSAR Section 5.4.6.2.3 (page 5.4-14), it is stated that the isolation condenser will start if the main steam isolation valve (MSIV) position on main steam line (MSL) A is less than or equal to 85 percent open. However, Figure 21.7.4-5, Sheet 5, Isolation Condenser System LD, indicates that the isolation condenser will start if the MSIV valve position on MSL A is less than 90 percent open. Which is correct?

440.53 The following isolation condenser system alarms are provided in operating plants:

(a) ISO COND SHELL TEMP HI
(b) ISO COND HI TEMPERATURE
(c) ISO COND STEAM LINE BREAK
(d) STEAM LEAK AREAS HI TEMPERATURE
(e) ISO COND AREA AIR EXHAUST

Which of these alarms are included in the SBWR design and which ones are not? For those that are not included in the SBWR design, explain why.

440.54 The following control room indications are provided for isolation condenser systems in operating plants:

(a) steam line pressure
(b) shell side level
(c) shell side temperature
(d) condenser outlet temperature
(e) vent line radiation monitors
(f) isolation condenser area exhaust temperature

Which of these indications are included in the SBWR design and which ones are not? For those that are not included in the SBWR design, explain why. (The existence of these control room indicators in the SBWR design could not be verified from the P&ID and LD diagrams.)

440.55 Submit a detailed drawing of the loop seal.

440.56 In SSAR Section 5.4.6.2.2, the catalytic converter is described on page 5.4-11 and the hydrogen recombiner is described on page 5.4-13. Both should be identified in the P&ID.

440.57 It is difficult to identify the containment isolation valves from the P&ID. The primary containment barrier should be shown in the P&ID. Identify the primary containment isolation valves and show on the P&ID.

440.58 Add a note to the P&ID stating that the power supply to all the valves are from dc power.