Central File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 5, 1994

Docket No. 52-004

Mr. Patrick W. Marriott, Manager
Licensing & Consulting Services
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

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 instrumentation and control systems described in Chapter 7 of the SBWR standard safety analysis report (SSAR) is needed (Q420.1-Q420.98).* In order to meet the SBWR RAI response date in SECY-93-097, "Integrated Review Schedules for the Evolutionary and Advanced Light Water Reactor Projects," dated April 14, 1993, please provide a written response to the enclosed questions by January 31, 1994.

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 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.

DF03
1/1

This RAI 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,

Melinda Malloy

Melinda Malloy, Project Manager
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of Nuclear Reactor Regulation

Enclosure:
RAI on the SBWR Design

cc w/enclosure:
See next page

Distribution (w/enclosure):

*Central File	PDST R/F	MMalloy
*PDR	SNinh	RHasselberg
JWermiel, 8H3	MChiramal, 8H3	ELee, 8H3
MFinkelstein, 15B18	GSuh (2), 12E4	WDean, 17G21
DCrutchfield/WTravers	RBorchardt	JNWilson
PShea	TMurley/FMiraglia,12G18	WRussell, 12G18
BSheron/TKing, NLS007	JMoore, 15B18	ACRS (11)

***To be held for 30 days**

OFC	LA:PDST	PM:PDST	PM:PDST	SC:PDST	D:PDST
NAME	PShea	MMalloy	SNinh	JNWilson	RBorchardt
DATE	1/5/94	1/5/94	1/5/94	1/5/94	1/ /94

DOCUMENT NAME: SBWR9407.MM

Mr. Patrick W. Marriott
GE Nuclear Energy

Docket No. 52-004

cc: Mr. Laurence S. Gifford
GE Nuclear Energy
12300 Twinbrook Parkway
Suite 315
Rockville, Maryland 20852

Director, Criteria & Standards Division
Office of Radiation Programs
U.S. Environmental Protection Agency
401 M Street, S.W.
Washington, D.C. 20460

Mr. Sterling Franks
U.S. Department of Energy
NE-42
Washington, D.C. 20585

Mr. John E. Leatherman
SBWR Licensing Manager
GE Nuclear Energy
175 Curtner Avenue, MC-781
San Jose, California 95125

Mr. Steven A. Hucik
GE Nuclear Energy
175 Curtner Avenue, MC-780
San Jose, California 95125

Mr. Frank A. Ross
Program Manager, ALWR
Office of LWR Safety & Technology
U.S. Department of Energy
NE-42
19901 Germantown Road
Germantown, Maryland 20874

Mr. Victor G. Snell, Director
Safety and Licensing
AECL Technologies
9210 Corporate Boulevard
Suite 410
Rockville, Maryland 20850

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

Instrumentation and Control (I&C)

420.1 Identify the reports that will be provided to support any aspects of the design that are substantially different relative to designs previously reviewed by the staff. Subjects addressed in these reports could include, but not necessarily be limited to, the following:

a. Overall block diagram(s) and descriptions of the reactor protection system (RPS) and engineered safety features actuation system (ESFAS), showing the architecture of the system, the allocation of functions to modules, and the communication channels among modules. Digital and analog modules should be identified. Methods for assuring required independence should be clearly identified, as well as power supply dependencies, division boundaries, and non-safety system interfaces. A description of the scope of on-line and diagnostic testing features for the proposed systems should be provided with regard to the diagram(s) to illustrate compliance with testability requirements.

b. The applicant's overall design verification program, covering development of the functional requirements, criteria, specifications, design, manufacture, test, and qualification methods and procedures. This should include a plan for software design verification and validation (V&V).

c. Failure modes and effects analysis for the I&C system.

d. A defense-in-depth analysis, demonstrating the diversity in the system that provides for defense against potential common-mode failures.

e. System (and significant component) reliability goals, assumptions, methodology, model, analysis, and evaluation.

f. Methodology, basis, and acceptance criteria for qualifying the system and equipment to the design-basis electromagnetic interference (EMI) and radio frequency interference (RFI) environment.

g. Methodology, basis, and acceptance criteria for qualifying the system and equipment to the design basis thermal environment established by localized heat transfer within these electronic equipment, including non-accident environments. This should also address the requirements for humidity controls to preclude damage from (1) electrostatic discharges and (2) moisture in the air.

Enclosure

h. Methodology, basis, and acceptance criteria for qualifying the system and equipment to the design-basis surge withstand capability.

j. Methodology, basis, and acceptance criteria for qualifying the system and equipment to the design-basis radiation environment, including environments normally considered "mild" for insulation materials.

k. Task analysis for the man-machine interface to the system.

420.2 Electromagnetic interference and radio frequency interference, including surge and electrostatic discharge, could reduce the reliability of the safety-related digital system. Provide a list of standards with which the SBWR design will comply to minimize and withstand EMI/RFI in the SBWR's environment.

420.3 Provide a discussion of SBWR's overall software development program. This should include development of the functional requirements, criteria, specifications, design, manufacture, test, and qualification methods and procedures. The discussion should also include a list of standards with which the SBWR software development program will comply (consider ANSI/IEEE/ANS-7-4.3.2-1993, "Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," for guidance).

420.4 Provide a list of all actuation devices of the reactor protection system and engineered safety features actuation system that cannot be fully tested during reactor operation. How will these devices be periodically tested to ensure that they are capable of performing their safety functions, in compliance with the guidance of Regulatory Guide (RG) 1.22?

420.5 The last sentence on page 7.1-9 of the standard safety analysis report (SSAR) in the discussion of compliance with RG 1.47 states that those portions of the bypass indications that, when faulted, could reduce the independence between redundant safety-related systems are electrically isolated from the protection circuit. Identify which are the portions of the bypass indications that are referred to on SSAR page 7.1-9.

420.6 Are there any limitations on the SBWR design concerning the use of expert systems? Are there any limitations on the use of technology not specifically described? Provide the requirements for using such systems and technology.

420.7 The application of high technology semiconductor electronics components has resulted in high current densities in some portions of equipment used in non-nuclear application. Identify how these higher current densities, which can result in localized hot-spots that can damage the electronic components, will be considered in the design. Is there provision in the design for monitoring hot-spots

and high localized temperature? When designing the electronic equipment, will thermal analysis be performed of the electronic boards? What method of cooling is being considered in the design, forced or natural circulation?

420.8 The SBWR design has active non-safety systems that perform important functions. These non-safety systems need to be operated reliably. To address the needed reliability, provide a discussion of the following:

a. Overall design verification program for the non-safety equipment that are important to safety;

b. Software development program, as described in Question 420.3;

c. Self-test requirements and surveillance test requirements;

d. Reliability/availability goals; and

e. The applicable standards and RGs.

420.9 Using a block diagram, describe the operation of the essential mulitiplexing system (EMS). The description should explain how the EMS transmits serial, time-pulsed data streams representing the status of plant variables, from local sensors to the logic processor. It should also explain how the EMS transmits alarm and trip status data to the safety system logic and control (SSLC) and display controllers in the main control room. (Reference SSAR Section 7.3.4.4.)

420.10 Describe the data transmission process between safety-related systems and non-safety-related systems, including the interface criteria. This should also describe the data bus used (protocols and error detection). Describe what happens when a single card on a data bus fails. Identify the design features that prevent errors from propagating into the safety systems. (Reference SSAR Section 7.1.1.)

420.11 Provide a discussion of the error tolerance of the bus and multiplexer. Explain how errors are detected and how the systems are tolerant to the errors.

420.12 What are the reliability/availability goals for the reactor protection system and engineered safety features (ESFs) systems? In addition, what testing will be done to demonstrate reliability and what is the scope of each test? The discussion should also include the method used in determining the system reliability/availability.

420.13 Provide a discussion of the availability of the reactor protection system monitoring systems. The discussion should include the conditions and functions that are being monitored to inform the operator of the status of both the long-term and short-term availability of the RPS. (Reference SSAR Section 7.2.1.2.)

420.14 Provide a list of the reactor protection system supporting equipment, such as air conditioning systems. If these supporting equipment are non-Class-1E equipment, what are the reliability requirements of the supporting equipment, and explain how they are isolated from the RPS. Would the failure of any of the supporting equipment reduce the reliability of the RPS? (Reference SSAR Section 7.2.1.)

420.15 Using a block diagram, describe the reactor protection system power distribution system. In addition, identify any non-Class-1E equipment connected to the Class 1E power supply. If any non-1E equipment is connected to the RPS power distribution system, explain how this non-1E equipment is isolated from the 1E power system, and explain the reasons for connecting non-1E equipment to 1E power supply. In addition, explain how the SBWR design complies with General Design Criteria 17 and 18, IEEE Standard 308-1974, and RG 1.32. (Reference SSAR Section 7.2.1.)

420.16 The second sentence of the second paragraph of SSAR page 7.2-13 on bypass indication states that indicator lights indicate which part of a system is not operable. Clarify whether these indicator lights indicate the bypass or inoperability of portions of a system that performs a function important to safety. (Reference SSAR Section 7.2.1.2.1 and RG 1.47.)

420.17 Describe the self-diagnostic features of the computer-based safety system. (Reference SSAR Section 7.2.1.4.)

420.18 Identify any on-line test equipment or circuits that are not part of the safety-related system. Also describe the interface between the safety-related system and the on-line test equipment. Show that faults in the test equipment will not challenge the system or equipment being tested. Explain how all four channels of reactor protection system are tested without violating independence/isolation criteria. Describe the process (configuration management) that will be incorporated at operating facilities when on-line diagnostics uncovers an error in the computer system. (Reference SSAR Section 7.2.1.4.)

420.19 Discuss the reactor protection system automatic testing features' compliance with RG 1.22, RG 1.118, and IEEE Standard 338. (Reference SSAR Section 7.2.1.4.)

420.20 Provide a single failure analysis of the reactor protection system as part of the failure modes and effects analysis (FMEA) in response to Question 420.1.c. (Reference SSAR Section 7.2.1.)

420.21 RG 1.47 requires that manual capability exist in the control room to activate each system-level indicator provided in accordance with Regulatory Position C.1. This position states that administrative procedures should be supplemented by a system that automatically indicates at the system level the bypass or deliberately induced inoperability of the protection system and the systems actuated or controlled by the protection system. Explain how SBWR complies with this RG 1.47 position. (Reference SSAR Section 7.2.1.5.3.)

420.22 Explain how the reactor protection system complies with RG 1.62, Regulatory Positions C.2 and C.3. (Regulatory Position C.2 of RG 1.62 on manual initiation of protective actions requires that manual initiation of a protective action at the system level perform all actions performed by automatic initiation. Regulatory Position C.3 states that the switch for manual initiation of protective action at the system level should be located in the control room and be easily accessible to the operator so that action can be taken in an expeditious manner.) In addition, explain how the RPS complies with Regulatory Position C.5 of RG 1.62, which states that manual initiation of protective actions should depend on the operation of a minimum of equipment. (Reference SSAR Section 7.2.1.)

420.23 Provide the alarm system requirements of SBWR design. The discussion should also include how the alarm system complies with Position II.T in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993, on control room annunciator reliability. (Reference SSAR Section 7.2.1.2.4.)

420.24 Do all system-level bypass indicators have the manual capability to be activated according to Regulatory Position C.4 of RG 1.47? List any bypass that does not have manual-activation capability and explain the reasons for not having it. (Reference SSAR Section 7.2.1.3.)

420.25 Describe the built-in interlocks that will prevent a simultaneous bypass of more than one channel. Provide a list of bypasses that do not have this interlock capability and provide justifications for not having it. (Reference SSAR Section 7.2.1.3.)

420.26 Does all equipment have a bypass status indication local to the equipment to provide information to maintenance personnel. (Reference SSAR Section 7.2.1.3.)

420.27 Describe how the bypass indicators are grouped in the control room. (Reference SSAR Section 7.2.1.3.)

420.28 Identify the reports that will be provided to support any aspects of the neutron monitoring system design that are different relative to designs previously reviewed by the staff. (Reference SSAR Section 7.2.2.)

420.29 Using block diagrams, describe the operation of the reactor protection and safety monitoring system for a average power range monitor upscale trip. The description should trace the transmission of the initiating signals from the sensors through the integrated protection cabinets, the engineered safety features actuation cabinets, and the monitoring and controls at the control room work station to the actuated devices. The diagram should also include all the major components, such as the sensors. the signal conditioners, the isolation devices, the multiplexers, the data buses, the indicators, the protection cabinets, and control rod drive system. The diagram should show all channels and components and interfaces. (Reference SSAR Section 7.2.1.)

420.30 Describe a startup range neutron monitor (SRNM) signal and the connections between a SRNM detector and preamplifier in the reactor building. Explain how the SRNM detector signals transmitted to preamplifiers are protected from the noises and interferences in their environment. (Reference SSAR Section 7.2.2.2.)

420.31 Provide a discussion of how the neutron monitoring system (NMS) instruments are tested. The discussion should also include the requirements with which NMS instruments must comply. (Reference SSAR Section 7.2.2.4.)

420.32 Provide a description of how all four NMS channels are tested without violating independence/isolation criteria. (Reference SSAR Section 7.2.2.4.)

420.33 Describe the methods and design criteria used to reduce the common mode failure vulnerabilities in the hardware and software of the NMS. (Reference SSAR Section 7.2.2.3.)

420.34 Explain how the SBWR complies with anticipated transient without scram (ATWS) mitigation requirements. In addition, describe the manual actuation system of the automatic depressurization system (ADS). (Reference SSAR Section 7.3.1.1.)

420.35 Describe the manual initiation features of the engineered safety features actuation system. The description should include how the manual features comply with (1) IEEE Standard 279 and RG 1.62 and (2) SECY-93-087, Position II.Q, "Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems." (Reference SSAR Section 7.3.1.1.2.)

420.36 Unlike previous boiling water reactor ADS actuation sequencing, the SBWR ADS actuation sequencing initiates only on water level. Explain the change. Would this reduce the system reliability? (Reference SSAR Section 7.3.1.1.2.)

420.37 Provide a discussion of how ADS channel integrity is maintained. This should include (1) the reliability of ADS and (2) environmental qualification of ADS. (Reference SSAR Section 7.3.1.1.2.)

420.38 Describe how the gravity-driven cooling system pool water level is monitored. Is the level constantly monitored in the control room? (Reference SSAR Section 7.3.1.2.2.)

420.39 The last sentence of the first paragraph on page 7.3-9 states that the deluge valves actuate immediately upon sensing extreme lower drywell base mat temperature. This sentence, however, is not consistent with the second sentence of the last paragraph on page 6.3-5 (SSAR Section 6.3.2.2) which states that the deluge line flow is initiated by thermocouple which sense high lower drywell region base mat temperature indicative of molten fuel on the lower drywell floor. This needs to be clarified. (Reference SSAR Section 7.3.1.2.2.)

420.40 The third sentence of the second paragraph of page 7.3-9 states that any division sending an input signal will generate a divisional output logic signal which is then sealed in for 30 minutes. Is this 30-minute seal-in different from the 30-minute delay for the suppression pool equalizing line valves? Also, explain what would happen if the input signal goes away. (Reference SSAR Section 7.3.1.2.2.)

420.41 The leak detection and isolation system (LD&IS) isolates the sources of leaks from the containment. Are all LD&IS isolations backed up by manual actuation in the control room? If not, explain why. (Reference SSAR Section 7.3.3.1.)

420.42 Using block diagram(s), describe the arrangement of the fiber-optic data links for inter-cabinet communications. Identify all the components (including power supply arrangements) to be used for inter-cabinet communications. List all the data links between the integrated protection cabinets, and explain how the data links in a cabinet are protected from faults in other cabinets. In addition, explain how the integrated protection cabinets communicate with other cabinets. (Reference SSAR Section 7.3.4.2.)

420.43 Describe the channel bypass provision in the reactor trip logic. This should include a detailed description of the design of hardware and software for reverting the 2-out-of-4 logic to a 2-out-of-3 logic, 2-out-of-4 logic to automatic trip, other logic reverting, alarm provision, and the basis for permitting indefinite time bypass of one channel for testing or maintenance. Is the "channel bypass" limited to the same function (e.g., high containment pressure) or can it be applied to different functions (e.g., one high containment pressure and one low water level)? Describe the relationship between channel bypass and the trip design. Describe the method of the bypass indication at the work station in the main control room. (Reference SSAR Section 7.3.4.2.)

420.44 Provide a list of manual actuation controls that are not independent of safety system logic and control or the essential mulitiplexing system. Provide a list of manual system-level and component-level actuation controls that are independent of the SSLC and EMS. (Reference SSAR Section 7.3.4.3.)

420.45 The second sentence of paragraph 2 on page 7.3-27 states that the testing shall not cause actuation of the driven equipment. Describe how this will be accomplished. (Reference SSAR Section 7.3.4.4.)

420.46 Describe the qualification of surveillance test equipment and diagnostic equipment. In addition, describe the interfaces between the test equipment and the safety equipment. Could the test equipment (1) compromise the separation between channels or (2) potentially degrade the safety-related equipment or system that they are testing? (Reference SSAR Section 7.3.4.4.)

420.47 Describe how protection systems are tested end to end. If some portions of the systems are not tested, explain why. In addition, explain (1) how failures in on-line testing systems will not prevent the safety circuits from performing their safety functions and (2) how the test configuration does not violate the separation requirements. (Reference SSAR Section 7.3.4.4.)

420.48 Describe any design or testing requirements that deviate from Section 3.6.1, "Testability Requirements," of Chapter 10 of EPRI Advanced Light Water Reactor Utility Requirements Document, Volume III, Passive Plant. (Reference SSAR Section 7.3.4.4.)

420.49 Provide a discussion of the use of commercial dedication software in safety systems. The discussion should also include the criteria for selecting commercial software, the accuracy of tools, and the process by which the developer notifies the end user of changes. (Reference SSAR Section 7.3.4.5.)

420.50 Although there are some differences in the systems aspects of the advanced boiling water reactor (ABWR) and the SBWR design, the electronic components and modules used for the SBWR I&C are very similar to those of the ABWR. Therefore, the requirements met by the SBWR design also should be very similar to the ABWR requirements. Provide a list of the standards and RGs with which the ABWR design complies, but the SBWR design does not. Also provide a list of standards and RGs which are unique to the SBWR design. In addition, provide a justification for each difference. (Reference SSAR Section 7.3.4.5.)

420.51 Describe the methods used to program firmware. The discussion should address the programming process that is implemented to improve the reliability of the firmware. (Reference SSAR Section 7.3.4.3.)

420.52 Identify the reports that will be provided to support any aspects of the software development requirements that are different relative to software development requirements previously reviewed by the staff. (Reference SSAR Section 7.3.4.5.)

420.53 Describe how software errors are tracked during software development. (Reference SSAR Section 7.3.4.5.)

420.54 Paragraph 3 on page 7.3-27 states that the use of interrupts for processing safety-related functions is discouraged. What are the requirements for using interrupts when they <u>are</u> used? (Reference SSAR Section 7.3.4.5.)

420.55 Describe the local area networks and communication systems and provide a list of standards with which the SBWR will comply. In addition, provide the installation requirements for fiber optic lines. (Reference SSAR Section 7.3.5.2.)

420.56 The fiber optic line protects signals from the noises in the environment; however, the fiber optic line driver and receiver are susceptible to the noises in their environment. What are the environmental qualification criteria for these drivers and receivers? (Reference SSAR Section 7.3.5.2.)

420.57 Show how the independence criteria in accordance with IEEE Standard 603 and IEEE Standard 379 are satisfied with the proposed configuration of fiber optic links. (Reference SSAR Section 7.3.5.2)

420.58 Describe the data highway system for the essential mulitiplexing system. This description should include error handling and error recovery of the system. Does the SBWR have sufficient error handling capability so that the discovery of an error would not cause a data highway traffic jam? In addition, describe the data handling capability of the EMS. Explain whether data traffic would increase during abnormal plant conditions? (Reference SSAR Section 7.3.5.2.)

420.59 Provide a safety and hazard analysis, sneak circuit analysis, and timing analysis for the protection systems. (Reference SSAR Section 7.3.5.2.)

420.60 Provide an explicit discussion of how the systems conform to IEEE Standard 279, paragraph 4.5 on channel integrity, as supplemented by RG 1.75 and IEEE Standard 384. (Reference SSAR Section 7.3.5.3.)

420.61 Confirm whether system-level failures of any multiplexer system detected by automatic diagnostic systems are indicated to the operators consistent with the requirements of IEEE Standard 279 and IEEE Standard 603 regarding safety system status indication. (Reference SSAR Section 7.3.5.4.)

420.62 Describe how the essential mulitiplexing system interfaces with non-safety-related equipment. (Reference SSAR Section 7.3.5.2.)

420.63 Describe the equipment that are tested by the on-line testing and automatic testing, and describe how the essential mulitiplexing system is tested end to end. (Reference SSAR Section 7.3.4.5.)

420.64 Unlike the ABWR design, the SBWR design has numerous passive safety systems that perform important functions. Provide a discussion of any precaution included in the SBWR design to prevent or minimize the inadvertent initiation of non-safety systems.

420.65 Describe the standby liquid control system's (SLCS's) manual initiation system. Is the manual system independent from the automatic initiation system? In addition, describe the interface between SLCS and the safety system logic and control (essential mulitiplexing system). (Reference SSAR Section 7.4.1.)

420.66 Explain how the SBWR design complies with 10 CFR 50.62 (requirements for reduction of risk from ATWS events for light-water-cooled nuclear power plants). (Reference SSAR Section 7.4.1.)

420.67 Explain which standby liquid control system parameters are monitored and displayed in the control room. (Is SLCS flow monitored?) (Reference SSAR Section 7.4.1.)

420.68 Provide a discussion of the reliability of the standby liquid control system's instrumentation and control system. This discussion should include how the SLCS will perform its function with acceptable reliability. (Reference SSAR Section 7.4.1.)

420.69 Provide a list of systems that interface with the remote shutdown system (RSS) and provide a description of the interface between the RSS and other systems. In addition, describe the RSS's defense against failures in the interfaces and interconnections. Could failures in the RSS prevent the I&C systems from performing their functions? (Reference SSAR Section 7.4.2.)

420.70 Provide a discussion of the remote shutdown system's environmental qualification criteria and reliability goals. (Reference SSAR Section 7.4.2.)

420.71 Explain how standby liquid control system or leak detection and isolation system actuation signals prevent the containment isolation valves from opening, or close them when they are open. In addition, provide a discussion of how reactor water cleanup (RWCU)/shutdown cooling (SDC) system actuation signals are isolated from SLCS and LD&IS actuation signals. (Reference SSAR Section 7.4.3.)

420.72 Provide a discussion of the I&C system reliability for the reactor water cleanup system/shutdown cooling system. This discussion should also include how the RWCU/SDC system will perform its function with acceptable reliability. (Reference SSAR Section 7.4.3.)

420.73 Provide a discussion of (1) the reactor water cleanup system/shutdown cooling system parameters monitored, and (2) how monitored data are processed. (Reference SSAR Section 7.4.3.)

420.74 Describe which isolation condenser (IC) parameters are monitored to ensure that (1) the isolation condenser system is ready to accomplish its safety function, and (2) the IC pool has sufficient water. (Reference SSAR Section 7.4.4.)

420.75 Provide a discussion of the isolation condenser's instrumentation and control system reliability. This discussion should include how the IC will perform its function with acceptable reliability. (Reference SSAR Section 7.4.4.)

420.76 Provide a discussion of the independence between alternate rod insertion (ARI) and the reactor protection system. Explain how the ARI design complies with 10 CFR 50.62. (Reference SSAR Section 7.4.5.)

420.77 Describe the interface between alternate rod insertion and other systems. In addition, describe the ARI's defenses against common-mode failures. (Reference SSAR Section 7.4.5.)

420.78 Describe how the required data is processed and displayed to comply with RG 1.97. (Reference SSAR Section 7.5.)

420.79 Explain how the SBWR design complies with NUREG-0737, Item I.D.2, which requires each applicant to install a safety parameter display system that will display a minimum set of parameters for operating personnel to determine the safety status of the plant. (Reference SSAR Section 7.5.)

420.80 Provide a list of the primary variables for the considered events listed in SSAR Tables 7.5-5 through 7.5-7 that are associated with called-for manual action. (Reference SSAR Section 7.5.1.3.)

420.81 Provide a discussion of the equipment classification of containment atmospheric monitoring system (CAMS). The discussion should include how CAMS achieves its required reliability (i.e., single failure criteria, defense against failures, etc.). In addition, describe what would happen if the hydrogen concentration measured by one of the two channels is different from that measured by the other channel. (Reference SSAR Section 7.5.2.3.)

420.82 Explain how the SBWR design complies with NUREG-0737, Item II.F.1. This item requires provisions for (1) instrumentation to measure, record, and readout in the control room: containment pressure, water level, hydrogen concentration, radiation intensity (high level), and noble gas effluent at all potential accident release points, (2) continuous sampling of radioactive iodine and particulates in gaseous effluent from all potential accident release points, and (3) onsite capability to analyze and measure samples. (Reference SSAR Section 7.5.2.3.)

420.83 Provide a discussion of equipment classification of process radiation monitoring system (PRMS). In addition, provide a discussion of how the PRMS achieves its required reliability (i.e., single failure criteria, defense against failures, etc.). Provide a description of the design and operation of the PRMS. (Reference SSAR Section 7.5.3.)

420.84 Provide a discussion of the test requirements of the process radiation monitoring system. (Reference SSAR Section 7.5.3.)

420.85 Provide a discussion of the equipment classification of the nuclear boiler system (NBS). In addition, provide a discussion of how the NBS achieves its reliability (i.e., single failure criteria, defense against failures, etc.). (Reference SSAR Section 7.7.1.1.)

420.86 Describe how the nuclear boiler system is tested. Provide a list of the RGs and standards with which it will comply. (Reference SSAR Section 7.7.1.4.)

420.87 Explain the differences between the ABWR and SBWR rod control and information system (RC&IS) designs. This should include any ABWR RC&IS requirements that are not met by the SBWR RC&IS and an explanation of the differences. (Reference SSAR Section 7.7.2.1.)

420.88 Section 7.7.2.1 states that the rod control and information system is highly reliable. Explain how this is achieved. The explanation should include defenses against common-mode failures. (Reference SSAR Section 7.7.2.1.)

420.89 Describe the power supplies for the fine motion driver cabinets and rod brake controller cabinets. (Reference SSAR Section 7.7.2.2.6.)

420.90 Describe the power supplies of the non-safety systems that perform important functions described in SSAR Section 7.7. The description should also include the sources of power. Are these power supplies redundant and uninterruptable?

420.91 Provide a description of the rod control and information system interface with other systems. In addition, describe the communication between the RC&IS and others. (Reference SSAR Section 7.7.2.2.8.)

420.92 Are control rod blocks, bypasses, or any detected rod movement errors displayed in the control room? (Reference SSAR Section 7.3.4.5.)

420.93 SSAR Section 7.7.2.1 states that the rod control and information system, a dual-channel system, is designed to be single failure proof. Explain how this can be achieved when placing one of the two RC&IS channels in bypass. The discussion should include defenses against common-mode failures. (Reference SSAR Section 7.7.2.2.9.)

420.94 Describe the RC&IS self-test and on-line diagnostic test features that identify failures of the instrumentation and control electronics. (Reference SSAR Section 7.7.2.4.)

420.95 Explain the design of the fault-tolerant digital controller of the feedwater control system (FWCS). In addition, describe the operation of the field voter and lock-up voter. (Reference SSAR Section 7.7.3.2.)

420.96 Provide a discussion of the feedwater control system's defenses against common-mode failures. In addition, explain how the FWCS' controls are qualified for its environment. (Reference SSAR Section 7.7.3.5.)

420.97 The second sentence on page 7.7-37 states that automatic power regulator system (APRS) is designed such that functionalities of safety-related systems in the plant are not affected by the APRS, a non-safety related system. Explain how this is achieved. (Reference SSAR Section 7.7.4.3.)

420.98 Describe the qualification of surveillance test equipment and diagnostic equipment for non-safety-related systems that perform important functions. In addition, describe the interfaces between these test equipment and the systems being tested. Could these test equipment (1) compromise the separation between channels or (2) degrade the safety-related equipment or system that they are testing? (Reference SSAR Section 7.7.4.5.)