May 17, 2006

Mr. David H. Hinds, Manager, ESBWR
General Electric Company
P.O. Box 780, M/C L60
Wilmington, NC 28402-0780

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 29 RELATED TO ESBWR DESIGN CERTIFICATION APPLICATION

Dear Mr. Hinds:

By letter dated August 24, 2005, General Electric Company (GE) submitted an application for final design approval and standard design certification of the economic simplified boiling water reactor (ESBWR) standard plant design pursuant to 10 CFR Part 52. The Nuclear Regulatory Commission (NRC) staff is performing a detailed review of this application to enable the staff to reach a conclusion on the safety of the proposed design.

The NRC staff has identified that additional information is needed to continue portions of the review. The staff's request for additional information (RAI) is contained in the enclosure to this letter. This RAI concerns Section 4.5.1, "Control Rod Drive System Structural Materials," Section 4.5.2, "Reactor Internal Materials," and Section 5.2, "Integrity of Reactor Coolant Pressure Boundary," of the Tier 2 ESBWR design control document. The RAI for Sections 4.5.1 and 4.5.2 was sent to you via electronic mail on April 10 and April 13, 2006, and was discussed with your staff during a telecon on May 4, 2006. The RAI for Section 5.2 was sent to you via electronic mail on March 20, 2006, and was discussed with your staff during telecons on April 25 and May 1, 2006. During the calls, GE and the NRC staff discussed the RAI and you clarified some of the staff's questions. You agreed to respond to this RAI on the following schedule:

RAI 4.5-1 through 4.5-32: June 2, 2006, and
RAI 5.2-6 through 5.2-29: June 6, 2006

If you have any questions or comments concerning this matter, you may contact me at (301) 415-4115 or mcb@nrc.gov. You may also contact Lawrence Rossbach at (301) 415-2863 or lwr@nrc.gov.

Sincerely,

/RA/

Martha Barillas, Project Manager
ESBWR/ABWR Projects Branch
Division of New Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 52-010

Enclosure: As stated

cc: See next page

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ACCESSION NO. ML061360248

OFFICE	NRBA/PM	NRBA/BC
NAME	MBarillas	ACubbage
DATE	05/17/2006	05/17/2006

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Request for Additional Information (RAI)
ESBWR Design Control Document (DCD) Sections 4.5.1 and 4.5.2

RAI Number	Reviewer	Summary	Full Text
4.5-1	Tsao J	Revise Table 4.5-1 to include a complete list of all reactor internal and core support structure components.	Design Control Document Tier 1 (DCD Tier 1), Figure 2.1.1-1, identifies chimney restraint, in-core housing, and drain line, which are parts of reactor internals, but which are not identified in DCD Tier 2, Table 4.5-1. Also, many reactor internal components identified on pages 2.1-3 to 2.1-5 of DCD Tier 1 are not mentioned in DCD Tier 2, Table 4.5-1. This discrepancy shows that DCD Tier 2 Table 4.5-1 is inadequate in identifying the core support structures and reactor internal components. (A) Justify why Table 4.5-1 is adequate or revise Table 4.5-1 in DCD Tier 2 to include a list of all core support structures and reactor internal components with corresponding material selection. (B) The components identified under the “Materials Used for the Core Support Structures” heading in Table 4.5-1 should not be considered as, or grouped with, the reactor internals because the core support structures have different ASME Code Class classification than some of the reactor internal components (e.g., non-ASME Code materials). Separating the core support structure components from the reactor internal components provides regulatory clarification. The staff recommends the title of DCD Table 4.5-1 be changed to “Material Specifications for Core Support Structures and Reactor Internals.” Also, include a heading in DCD Table 4.5-1 (e.g., “Materials Used for the Reactor Internals”) so that components listed under this heading would be identified as reactor internal components (e.g., steam dryers, steam separators, and chimney). The purpose of these two suggestions is to distinguish and separate the core support structure components from the reactor internal components.

RAI Number	Reviewer	Summary	Full Text
4.5-2	Tsao J	Provide drawings of core support structures and reactor internal components.	DCD Tier 2, Section 4.5.2 contains no drawings of the core support structures nor reactor internals. Figures 4.6-1 to 4.6-10 in DCD Tier 2 contain only schematics of control rod drive mechanisms. DCD Tier 1, Figure 2.1.1-1 contains a sketch of reactor internals without details. (A) Provide the detailed drawings and/or diagrams of all significant core support structures and reactor internal components. (B) Provide assembly drawings and diagrams to show how the core support structure components and reactor internal components are attached to each other and/or to the reactor vessel. ©) Please include the drawings and diagrams in Section 4.5.2 of DCD Tier 2.
4.5-3	Tsao J	Address degradation and inspectibility of cast austenitic stainless steel.	DCD Table 4.5-1 identifies cast austenitic stainless steel, ASTM or ASME Grade CF3/CF3M, as a material used in the reactor internals and core support structures. Cast austenitic stainless steel is susceptible to the loss of fracture toughness due to thermal aging embrittlement, neutron irradiation embrittlement and void swelling in the reactor vessel. The staff's concern was documented in a letter from Christopher I. Grimes of NRC to Douglas J. Walters of Nuclear Energy Institute, subject: License Renewal Issue No. 98-0030, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components," May 19, 2000 (ADAMS ML003717179). In addition, ultrasonic examinations of cast austenitic stainless steel have not been reliable. Address the loss of fracture toughness due to thermal aging embrittlement, neutron irradiation embrittlement and void swelling in cast austenitic stainless steel in the reactor vessel. Address the inspectibility of cast austenitic stainless steel by ultrasonic examinations.
4.5-4	Tsao J	Clarify the discrepancy in Table 4.5-1 on Alloy 600 material.	DCD Table 4.5-1 indicates that niobium modified nickel-chromium-iron-alloy 600 per ASME Code Case No. –580-1 will be used in the shroud support. However, Code Case –580-1 discusses adding columbium, not niobium, in Alloy 600. Discuss the discrepancy.

RAI Number	Reviewer	Summary	Full Text
4.5-5	Tsao J	Justify the use of non-L grade 304 and 316 stainless steel.	DCD Table 4.5-1 identifies that non-L grade 304 and 316 stainless steels are used for the reactor vessel internals and core support structures. Justify the use of non-L grade 304 and 316 stainless steels for the core support structures and reactor internals in light of the industry history of intergranular stress corrosion cracking in the 304 and 316 stainless steels and potential neutron irradiation embrittlement or irradiation assisted stress corrosion cracking in the BWR.
4.5-6	Tsao J	Provide information on the non-coded material.	DCD Section 4.5.2.1 states that "...The other reactor internals are non-coded, and they may be fabricated from ASTM or ASME specification materials or other equivalent specifications...". The statement seems to be contradictory because "non-coded" generally means that a component is not fabricated using ASME specifications. (A) Clarify the above statement. (B) Identify the specific material specification for each of the reactor internal components and include this information in DCD Table 4.5-1.
4.5-7	Tsao J	Justify the use of non-coded material in the reactor internals.	(A) Discuss the operating experience (i.e., degradation) of the non-coded materials used in the reactor internals in the current BWR fleet. (B) Demonstrate that the non-coded material will provide the necessary strength, resistance to corrosion, and fracture toughness to maintain the safe operation of the ESBWR. ©) Discuss whether the non-coded components are designed for and analyzed with the same loading combinations per the ASME Code, Section III, as that used for the ASME Code components. If not, demonstrate by analysis that the failure of the non-coded components will not affect the structural integrity of the ASME Code components (e.g., the core support structures). (D) Clarify whether the non-coded components are considered as safety or non-safety category components.
4.5-8	Tsao J	Discuss the inspection and welding of the non-coded materials.	Discuss the industry standards that the non-coded components will follow in terms of material selection, fabrications (including welding), construction, design (e.g., stress analysis), testing, and inspections.

RAI Number	Reviewer	Summary	Full Text
4.5-9	Tsao J	Revise Section 4.5.2.2 to include welding requirements and follow the ASME Code for the reactor internals.	(A) The title of DCD Section 4.5.2.2 is “Controls on Welding”. However, welding is not explicitly mentioned in this section. To clarify the intent, revise the first sentence in Section 4.5.2.2 to read: “...Core support structures are welded in accordance with...”. (B) Standard Review Plan (SRP) Section 4.5.2.II.2, Draft Revision 3, April 1996, specifies that methods and controls for core support structures and reactor internals welds must be in accordance with the ASME Code, Section III, Division 1, NG-4000, and the welds must be examined and meet acceptance criteria as specified in NG-5000. The second sentence in DCD Section 4.5.2.2 discusses the welding of the reactor internals without referring to the above ASME Code sections. Justify why the welding of the reactor internals does not follow ASME Code, Section III, NG-4000 and NG-5000, and ASME Code, Section IX. ©) Identify the core support structure and reactor internal components that require welding and describe the welding technique/procedures that will be used.
4.5-10	Tsao J	Add the acceptance criteria of the examination in Section 4.5.2.3	SRP Section 4.5.2.II.3, Draft Revision 3, April 1996, specifies that “...the acceptance criteria of the nondestructive examination shall be in accordance with the requirements of ASME Code, Section III, Division I, NG-5300...” DCD Section 4.5.2.3 does not specify the acceptance criteria for the nondestructive examination. Revise DCD Section 4.5.2.3 to include the acceptance criteria for nondestructive examination and identify the appropriate ASME Code section or justify why the acceptance criteria are not needed.
4.5-11	Tsao J	Revise Section 4.5.2.3 to require the examination of all core support structures and reactor internals.	DCD Section 4.5.2.3 discusses the nondestructive examinations of control rod drive housings and peripheral fuel supports. SRP Section 4.5.2.I.3, Draft Revision 3, April 1996, recommends that each product form in the reactor internals and core support structures be examined. (A) Justify why product forms other than control rod drive housings and peripheral fuel supports do not need nondestructive examinations or revise the title and contents of DCD Section 4.5.2.3 to require nondestructive examinations of all product forms in the reactor internals and core support structures. (B) Identify the specific tubular products that will be hydrostatically tested.

RAI Number	Reviewer	Summary	Full Text
4.5-12	Tsao J	Justify the use of cold-worked materials in steam vanes.	DCD Section 4.5.2.4 states that significantly cold-worked stainless steels are not used in the reactor internals except for vanes in the steam dryers. There are advantages and disadvantages of using cold-worked materials in certain applications. Justify the use of cold-worked materials in vanes considering the adverse impact of the cold work on the microstructure of the material and its susceptibility to stress corrosion cracking.
4.5-13	Tsao J	Address the issues related to Alloy X-750 material.	DCD Section 4.5.2.5 states that "...Alloy X-750 components are fabricated in the annealed and aged condition. Where maximum resistance to stress corrosion is required, the material is used in the high temperature (1093EC) annealed plus single aged condition...". Alloy X-750 materials are susceptible to intergranular stress corrosion cracking due to equalized and aged heat treatment conditions (Reference: BWRVIP-41, "BWR Vessel and Internals Project: BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," EPRI 1012137, page 2-3). (A) Identify the reactor internal components fabricated from Alloy X-750. (B) Provide information on aging heat treatment process of Alloy X-750 (i.e., aging temperature and holding time). (C) Justify how this aging process will help to prevent/minimize stress corrosion cracking. (D) Provide the optimal hardness value that is required to minimize the susceptibility to stress corrosion cracking. (E) Discuss the discrepancy in Alloy X-750 not being identified in DCD Table 4.5-1 although this material is identified in DCD Section 4.5.2.5 as a material used in reactor internals.
4.5-14	Tsao J	Discuss the pre-service and inservice inspection of all core support structure and reactor internal components.	(A) Discuss the pre-service inspection and inservice inspection program of all core support structure and reactor internal components. For each component, the discussion should include specific examination technique, frequency of the inspection, acceptance criteria, the area/coverage of the inspection, and the industry codes/requirements used. (B) Provide a list of components that will not be inspected during the pre-service inspection or inservice inspection activities and explain why the inspection is not needed.

RAI Number	Reviewer	Summary	Full Text
4.5-15	Tsao J	Discuss applicability of BWRVIP guidance.	The BWR Vessel and Internals Project (BWRVIP) has published many guidelines in BWRVIP reports related to the reactor internals. The NRC has approved some of the BWRVIP reports. Discuss briefly which guidance/reports will be used for which components in the ESBWR. If none of the BWRVIP guidelines will be followed as a matter of practice or policy, provide explanation.
4.5-16	Tsao J	Discuss the maintenance program for bolts and threaded fasteners used inside the reactor vessel.	Discuss the maintenance program for the bolts and threaded fasteners used in the core support structures and reactor internals to ensure their structural integrity (i.e., prevent bolt cracking) and to prevent them from becoming loose parts in the reactor coolant system.
4.5-17	Tsao J	Provide information on chimney and associated components.	(A) Identify the ASME Code requirements for material selection, inspection, design, fabrication, and construction of the chimney, chimney partitions, and chimney head. (B) Describe the fabrication, assembly, and installation of the chimney, chimney partitions, and chimney head. ©) Discuss whether a mockup test of the chimney assembly in a reactor vessel environment has been conducted to verify the structural integrity of the chimney assembly.
4.5-18	Tsao J	Discuss the design of the core shroud supports.	The core shroud supports in the current BWR fleet are supported from and attached to the bottom of the reactor. The ESBWR design is attached and supported at the side wall of the reactor vessel which may result in a bending moment on the vessel wall. (A) Discuss whether the design of the core shroud supports considered the potential bending of the reactor vessel wall (i.e., the shroud supports may not sustain the loads as calculated in the structural analysis because the vessel wall may not be as rigid as assumed in the analysis). (B) Discuss whether the stress analysis of the reactor vessel shell considered the bending moment generated by the core shroud supports. ©) The core shroud supports use Niobium-modified Inconel 600 alloy. Alloy 600 material is susceptible to stress corrosion cracking. Justify the selection of this material in the reactor vessel or describe the design features that will be used to mitigate stress corrosion cracking. (D) Provide the drawings and design details including the location and installation of the core shroud supports.

RAI Number	Reviewer	Summary	Full Text
4.5-19	Tsao J	Clarify the CRD housing design.	DCD Tier 1, Page 2.1-2, states that "...The control rod drive (CRD) housings... are welded to CRD penetrations (stub tubes) formed in the reactor pressure vessel (RPV) bottom head...". DCD Tier 1, Page 2.1-4, states that "...the upper end of the CRD housing is welded to a stub tube that is directly welded to the bottom of the vessel...". It is not clear whether the stub tube is welded or formed as part of RPV bottom head forging. (A) Provide assembly drawings of the CRD housing and stub tube to show how they are attached to each other and to the bottom of the vessel. (B) Discuss weld joint details, welding processes, post-weld heat treatments, materials to be used, and the fabrication sequence that will be used to prevent sensitization of the stainless steel material (based on operating experience in the current BWR fleet, i.e., Oyster Creek).
4.5-20	Tsao J	Provide information on the in-core guide tubes and associated lateral supports.	Section 2.1.1 in DCD Tier 1 states that a latticework of clamps, tie bars, and spacers provide lateral support and rigidity to the in-core guide tubes. (A) Provide assembly drawings of the lateral support components, in-core guide tubes, how the lateral support components are inter-connected, and how the in-core guide tubes are attached to the shroud. The drawings should be included in Section 4.5.2 in DCD Tier 2. (B) Identify materials used for the lateral support components and in-core guide tubes. (C) Identify the number of penetrations.
4.5-21	Tsao J	Discuss hydrogen water chemistry in the reactor vessel.	Discuss whether a hydrogen water chemistry program will be implemented in the reactor vessel to mitigate stress corrosion cracking. If so, discuss briefly the hydrogen water chemistry program and associated requirements.
4.5-22	Tsao J	Discuss material fabrication of austenitic stainless steel.	DCD Tier 1, Page 2.1-5, states that special controls on material fabrication processes are exercised when austenitic stainless steel is used for construction of RPV internals. Describe the special controls that are used for material fabrication.

RAI Number	Reviewer	Summary	Full Text
4.5-23	Tsao J	Discuss potential cracking of feedwater nozzles and spargers.	DCD Tier 1, Page 2.1-4, discusses the feedwater spargers. Cracking of the feedwater spargers in the current BWR fleet is discussed in the NRC report, NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking." Describe design features, fabrication processes, and water chemistry to minimize or prevent cracking in feedwater nozzles and spargers in ESBWR. Discuss the inspection program for the feedwater spargers (examination scope, technique, and frequency).
4.5-24	Tsao J	Discuss the prevention of loose parts in the reactor vessel.	DCD Tier 1, Section 2.9, and DCD Tier 2, Section 4.4.5, describe the loose parts monitoring system to detect metallic parts in the reactor vessel. The system uses acoustic sensors to detect loose parts and alarms to notify the operators. There is no discussion on the measures that will be taken to prevent the generation of loose parts. Describe the programs that will be used to prevent and manage metallic loose parts in the reactor vessel during fabrication/assembly, maintenance, normal operation, and refueling activities.
4.5-25	Tsao J	Address failure modes in the core support structures and reactor internals.	Discuss the likelihood of the following degradation mechanisms affecting all significant core support structures and reactor internal components: ductile and brittle fractures, fatigue failures, distortion failures, wear failures, erosion (cavitation and liquid-impingement), corrosion (pitting, leaching, galvanic, and intergranular), creep, corrosion-fatigue, hydrogen-damage failures, embrittlement (neutron irradiation and thermal), and stress corrosion cracking. Discuss the steps taken to minimize or prevent the applicable degradation mechanisms (e.g., inspection and mitigation techniques).
4.5-26	Davis R	Provide correction to reference of RG 1.85.	DCD Section 4.5.1.1 "Materials" indicates that the applicant used Regulatory Guide (RG) 1.85 for the design of the ESBWR. However, RG 1.85 was withdrawn by the NRC on June 10, 2004. RG 1.84 contains guidance on all Section III Code Cases, including those related to materials and related testing in Division 1 which were previously contained in RG 1.85. Please update Section 4.5.1.1 to reflect the correct Regulatory Guide used in the design of the ESBWR.

RAI Number	Reviewer	Summary	Full Text
4.5-27	Davis R	Provide discussion of the use of Cobalt and non-Cobalt bearing hard surfacing alloys in the CRDs.	Please discuss the selection, basis for selection, and operating experience with the materials selected and used in the Cobalt bearing and non-Cobalt bearing hard surfacing alloys in the ESBWR design.
4.5-28	Davis R	Provide the basis for Alloy X-750 heat treatment for CRD components.	Given the relationship between mechanical and thermal processing of Alloy X-750 and its susceptibility to stress corrosion cracking, please provide a basis for the heat treatment described in DCD Subsection 4.5.1.1.
4.5-29	Davis R	Provide clarification on intent to comply with RG 1.44.	DCD Section 4.5.1.2.1 states that the degree of conformance to RG 1.44 is presented in Subsection 4.5.2.4. Subsection 4.5.2.4 states that “These controls are employed to comply with the intent of RG 1.44.” The word “intent” does not make it clear if the application meets the RG positions. If the ESBWR design does not meet all of the provisions of RG 1.44, please list the deviations and provide a basis. If the ESBWR design does meet the RG positions, please correct the language in Subsection 4.5.2.4 and any other applicable Subsection of the DCD to reflect that the design complies with RG 1.44. Specify the test used to comply with the guidance provided in RG 1.44. Provide the response in a global context as it applies to the entire ESBWR design.
4.5-30	Davis R	Provide clarification on use of NQA-1, part 2.2.	DCD Subsection 4.5.1.4 references NQA-1, part 2.2 and RG 1.37. Please specify the edition of NQA-1 that is applicable. The staff notes that the ESBWR DCD references NQA-1-1983 in Chapter 17 but the applicable section related to requirements for cleaning of fluid systems and associated components is located in NQA-2-1983. Please provide clarification and state if all positions of RG 1.37 are being met. Provide the response in a global context as it applies to the entire ESBWR design.

RAI Number	Reviewer	Summary	Full Text
4.5-31	Davis R	Provide ESBWR design special controls limits associated with incidental cold work.	DCD Section 4.5.1.1 states that for incidental cold work introduced during fabrication and installation, special controls are used to limit the induced strain and hardness, and bend radii are kept above a minimum value. Please provide the values of the ESBWR design special controls limits on hardness, 0.2% offset yield strength and induced strain. Also discuss abrasive work controls for limiting cold working and the introduction of contaminants during abrasive work. Provide the response in a global context as it applies to the entire ESBWR design.
4.5-32	Davis R	Verify that the ESBWR design meets provision of RG 1.31 .	DCD Subsection 4.5.2.4 describes delta ferrite content for weld materials and specifies the minimum and maximum values for the ESBWR design but the application, as it relates to Subsection 4.5.1, does not state if the ESBWR design meets all of the provisions of RG 1.31. If the ESBWR design does not meet all of the provisions of RG 1.31, please list the deviations and provide a basis. If the ESBWR design does meet the RG positions, please update Subsection 4.5.2.4 and any other applicable Subsection of the DCD to reflect that the design complies with RG 1.31.

Requests for Additional Information (RAIs)
ESBWR Design Control Document (DCD) Section 5.2

RAI Number	Reviewer	Question Summary	Full Text
5.2-6	Hammer G Thomas G	Applicable General Design Criteria (GDC)	DCD Tier 2, Section 5.2.2 lists the applicable GDC. In addition to GDC 15, the overpressure protection and pressure control devices inboard of the main steam isolation valves (MSIVs) are considered part of the reactor coolant pressure boundary (RCPB), therefore, GDC 14 and 30 are applicable. Provide a discussion of how GDC 14 and 30 are met.
5.2-7	Thomas G Hammer G	TMI-2 Action Item II.D.1 and anticipated transients without scram (ATWS)	DCD Tier 2, Section 5.2.2 states that "The ESBWR meets the recommendations of the TMI action plan items of NUREG 0737 regarding testing to qualify reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents and the provision of direct indication of relief and safety valve position." TMI-2 action item II.D.1 requires licensees to provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves for all fluid conditions expected under operating conditions, transients and accidents. Consideration of ATWS conditions shall be included in the test program. For current operating plants, this issue was resolved with a generic test program for current valve designs and plant-specific responses for individual plant piping configurations and system responses. Confirm that the generic test program for the currently operating plants is applicable for ESBWR transients, ATWS, and accidents, or provide a commitment to perform the required testing. Also, provide a commitment to provide necessary plant-specific responses as a combined operating license (COL) action item.

RAI Number	Reviewer	Question Summary	Full Text
5.2-8	Thomas G	TMI-2 Action Item II.D.3	In accordance with the requirements of TMI-2 action item II.D.3, safety relief valve (SRV) position indications are provided in the control room. In some operating plants, thermocouples and acoustic monitors are provided at the SRV discharge piping for redundancy and diversity. Why are these diverse and redundant features not included in the ESBWR design?
5.2-9	Thomas G	TMI-2 Action Item II.K.3.3	Address TMI-2 action item II.K.3.3 in regards to reporting SRV challenges and failures. This may be a COL action item which is to be tracked.
5.2-10	Thomas G Hammer G	Rupture Disc at SRV discharge	Confirm that American Society of Mechanical Engineers (ASME) Code NB-7623 provisions will be met for rupture disc installation. Provide details of the specific SRV design features which ensure that the rupture disc will not impede SRV response time or operation.
5.2-11	Thomas G Hammer G	Safety mode operation of SRVs	It is stated in DCD Tier 2, Section 5.2.2.2.2 on page 5.2-5, that “[e]ight of the SRVs are opened by steam pressure initiated if the direct and increasing static inlet pressure overcomes the restraining spring and the frictional forces acting against the steam inlet pressure at the main or pilot disk and the main disk moves in the opening direction at a faster rate than corresponding disk movements at higher or lower inlet steam pressures.” Essentially identical language is used to describe the 10 SRV/automatic depressurization valves (ADS). This statement is confusing. Should the statement read: “the main (or pilot) disk opens quickly when the steam inlet pressure exceeds the restraining spring force and frictional forces”?

RAI Number	Reviewer	Question Summary	Full Text
5.2-12	Hammer G	SRV discharge line vacuum breaker failure analysis	Operating experience shows that there have been events involving failure of the SRV discharge line vacuum breakers to open or close properly. Some events have occurred as a result of condensation pressure oscillations which resulted from routine SRV seat-leakage. Provide an analysis of the failure of a vacuum breaker to either open or close properly during or following SRV discharge to demonstrate that discharge lines remain operable and that reactor coolant system and containment design criteria are met.
5.2-13	Hammer G	SRV pilot disk Corrosion	In some currently operating boiling water reactors, there has been excessive corrosion bonding of SRV pilot disks to their seats, causing excessively high set-point drift. A contributing cause was found to be the presence of radiolytic oxygen buildup, and one corrective action was to recombine the oxygen with hydrogen using a catalyst to form water. Aside from maintaining oxygen concentration through hydrogen addition, what other provisions will be made to prevent excessive set-point drift from corrosion bonding in SRVs for the ESBWR?
5.2-14	Thomas G	Depressurization Valve (DPV) and SRV Discharge	DPVs and non-ADS SRVs are discharged into the drywell. Is there any safety grade equipment in the drywell which is required for safe shutdown of the plant which will be affected by the DPV and SRV discharge into the drywell?
5.2-15	Thomas G	DPV relief capacity and overpressure analysis	DPV capacities are given in DCD Tier 2, Table 5.2-2. It is not clear whether credit is taken for the DPVs in the overpressure analysis. Confirm that credit is not taken for DPVs in the overpressure analysis.

RAI Number	Reviewer	Question Summary	Full Text
5.2-16	Hammer G	SRV operability and liquid discharge	Operational history for boiling water reactors indicates that SRV liquid discharge may occur as a result of reactor vessel overfilling into the main steam lines. Provide information regarding SRV liquid discharge in the event of vessel overfilling to demonstrate that the SRVs remain operable and that the SRV discharge lines and main steam lines do not exceed the applicable stress limits.
5.2-17	Hammer G Thomas G	SRV capacity and adequate margin	DCD Tier 2, Section 5.2.2.3.2, states that “SRV capacities are based on establishing an adequate margin from the peak vessel bottom pressure to the vessel code limit.” Explicitly define “adequate margin.” If the margin includes any factors besides the 3% SRV set-point pressure tolerance, describe these additional factors.
5.2-18	Thomas G	TRACG version and addition to the reference section	Specify the version of the TRACG code used for the overpressure protection analysis, and provide the appropriate reference in DCD Tier 2, Section 5.2.7.
5.2-19	Thomas G	Power actuated pressure relief function	Why is the automatic power-actuated pressure relief function not included in the ESBWR design?
5.2-20	Thomas G Hammer G	SRV performance, set-point drift and seat-leakage	SRV set point drift and seat leakage are generic problems. Describe specific design features of the ESBWR SRVs. Compare the relative performance of ESBWR SRVs and SRVs currently installed in operating reactors. Provide a detailed description of any improvements between ESBWR SRV designs and presently operating plant SRVs in the areas of seat-leakage, set-point drift, and actuator reliability.

RAI Number	Reviewer	Question Summary	Full Text
5.2-21	Thomas G Hammer G	SRV valve operator type and valve operator design	Include a discussion of improvements in the air actuator, especially materials used for components such as diaphragms and seals. Discuss the safety margins associated with the air accumulator design. Discuss the pressure indications in the accumulator and how this information is relayed to the operator.
5.2-22	Thomas G Hammer G	SRV specifications and environmental qualification.	What provisions have been employed to ensure that valve and valve actuator specifications include design requirements for operation under expected environmental conditions (i.e. radiation, temperature, humidity, and vibration)?
5.2-23	Hammer G Thomas G	SRV position indication	In DCD Tier 2, Section 5.2.2.5, why are SRV position indication and seat-leakage detection not instrumentation requirements for the ESBWR?
5.2-24	Thomas G Hammer G	Valve testing	What programs have been instituted to ensure that valves are manufactured to specifications and will operate as designed? For example, what tests are performed to ensure that the blowdown capacity is within specifications?
5.2-25	Thomas G Hammer G	Valve inspection and overhaul frequencies and procedures	Operating experience has shown that SRV failure may be caused by exceeding the manufacturer's recommended service life for the internals of the SRV or air actuator. What frequency is necessary for visual inspection and overhaul of the SRVs (both safety/relief and ADS valve components)? In addition to periodic testing, valve inspection and overhaul should be performed in accordance with the manufacturer's recommendations. Provide assurance that procedures will be in place to ensure that the design service life will not be exceeded for any component of the SRV?

RAI Number	Reviewer	Question Summary	Full Text
5.2-26	Thomas G Hammer G	SRV control systems	DCD Tier 2, Section 5.2.2.2.2 states: "The power-actuated SRVs can be operated individually by remote manual controls from the main control room." This implies that only the ten ADS valves can be opened from control room. Can the other eight non-ADS valves be remotely opened and closed? Can the safety relief valves be closed by operators when these valves are actuated as part of the ADS function? If so, how long after ADS actuation can this be accomplished?
5.2-27	Thomas G Hammer G	Vessel Pressure and SRV operation	Provide a figure showing the peak vessel bottom pressure as a function of the number of SRVs assumed to open in the isolation overpressure analysis. Identify the minimum number of SRVs that must open to remain within ASME limits on vessel pressure.
5.2-28	Thomas G	SRV Spring Set Pressure Values in Inspectrions, Tests, Analysis and Acceptance Criteria (ITAAC)	DCD Tier 1, Table 2.1.2-1 does not include spring set-points as part of the Inspections, Tests, Analysis and Acceptance Criteria (ITAAC) for the SRVs. Add the spring set pressures for all the SRVs to the ITAAC.

RAI Number	Reviewer	Question Summary	Full Text
5.2-29	Thomas G	Additional ITAAC Items	The following ITAAC were included in the ABWR DCD. Explain why similar ITAAC are not provided the ESBWR DCD. (a) The ADS can be initiated manually. (b) The reactor pressure vessel (RPV) water level instrumentation considers the effects of dissolved noncondensable gasses in the RPV water instrument lines. ©) The mechanical portion of each division of the safety-related NBS instrumentation located in the Reactor Building is physically separated from other divisions. (d) Motor operated valves designated in Section 2.1.2 as having an active safety function will close under differential pressure, fluid flow, and temperature conditions. (e) Control valves designated in Section 2.1.2 as having an active safety-related function will actuate (open, close, or both open and close), under differential system pressure, fluid flow, and temperature conditions (f) The ADS accumulator can open the SRV with the drywell pressure at design pressure following failure of the pneumatic supply to the accumulator

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