December 11, 2008

Mr. Robert E. Brown
Senior Vice President, Regulatory Affairs
GE Hitachi Nuclear Energy
3901 Castle Hayne Road MC A-50
Wilmington, NC 28401

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 275 RELATED TO ESBWR DESIGN CERTIFICATION APPLICATION

Dear Mr. Brown:

By letter dated August 24, 2005, GE-Hitachi Nuclear Energy (GEH) 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.

If you have any questions or comments concerning this matter, you may contact me at 301-415-6256 or Dennis.Galvin@nrc.gov or you may contact Amy Cubbage at 301-415-2875 or Amy.Cubbage@nrc.gov.

Sincerely,

/RA/

Dennis Galvin, Project Manager
ESBWR/ABWR Projects Branch 1
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-010

Enclosure:
Request for Additional Information

cc: See next page

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Docket No. 52-010

Enclosure:
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cc: See next page
Distribution: See next page

ACCESSION NO.: ML083440278 NRO-002

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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 275 RELATED TO ESBWR DESIGN CERTIFICATION APPLICATION DATED DECEMBER 11, 2008

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Requests for Additional Information (RAIs):
ESBWR Design Control Document (DCD) Revision 5

RAI Number	Reviewer	Question Summary	Full Text
RAI 9.1-50 S02 (MFN 08-4557, July 1, 2008)	Kelly G	Justify use of a smaller minimum depth of shielding for movement of radioactive equipment	Regulatory Guide 1.13 provides guidance that the minimum safe water shielding depth associated with spent fuel assemblies is 3.05 meters (10 feet). In Section 9.1.4.1, "Design Bases," and 9.1.4.5, "Refueling Equipment," for the refueling machine and the fuel handling machine, the applicant states that a safe water shielding depth of 2.59 meters (8.5 feet) is always maintained during transit of radioactive equipment/fuel over active fuel. Please modify the minimum shielding depth or identify the differences between the design features, analytical techniques, and procedural measures proposed for the ESBWR design and the NUREG-0800, "Standard Review Plan," (SRP) acceptance criteria and address how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with NRC regulations.
RAI 9.1-96	Kelly G	Add reference to ASME standard that provides criteria for single failure proof cranes and better identify single failure proof cranes	In Section 9.1.4, "Design Bases," Section 9.1.5.2, "General," and Table 9.1-5, "Reference Codes and Standards," of the DCD, Revision 5, the applicant references only NUREG-0554, "Single-Failure Proof Cranes for Nuclear Power Plants," as containing the guidance it will follow in designing a single failure proof crane. Section 9.1-5, "Overhead Heavy Load Handling Systems," of NUREG-0800, "Standard Review Plan," Subsection 4(C)(i) calls for single failure proof, Type 1 cranes to be designed to the criteria of ASME NOG-1 2004. Please modify your write up in Sections 9.1.4 and 9.1.5, and Table 9.1-5 of the DCD Tier 2 to refer to the ASME standard for each single failure proof crane, and to more clearly articulate which of the cranes are going to be designed to be single failure proof. In particular, the staff desires clarification about the status of the RB and FB cranes.

Enclosure

RAI 9.1-97	Kelly G	Provide a reference to where the DCD justifies the suppression pool suction line can never rupture	In Section 9.1.3.2, “System Description” of the DCD, Revision 5, the applicant states that the FAPCS “suppression pool suction line is conservatively designed to preclude a rupture between the pool and the containment isolation valves.” Please provide a reference(s) where the design details and justification are provided in the DCD that this line cannot rupture under any circumstances.
RAI 9.1-98	Kelly G	Clarify statement about dual mode operation of FAPCS	Section 9.1.3.2, “System Description,” DCD, Revision 5, states that dual mode operation of FAPCS is prohibited using a single train. Please explain how that action specifically is prohibited.
RAI 9.1-99	Kelly G	How were main steam line and IC plugs considered in the heavy load drop analysis?	Section 9.1.5, “Overhead Heavy Load Handling Systems (OHLHS)”, DCD, Revision 5, describes the heavy load drop analyses performed by the applicant. Please describe how the evaluations took into account the potential for the function of main steam line and isolation condenser nozzle plugs to be affected by heavy load drops. Similarly, what is the effect of heavy load drops on SSCs that form a temporary reactor coolant boundary during shutdown activities?
RAI 9.1-100	Kelly G	The heavy loads handling system description needs to address how it meets GDC 1	The Acceptance Criteria in Section 9.1.5, “Overhead Heavy Load Handling Systems,” NUREG-0800, “Standard Review Plan” (SRP) cites that General Design Criteria (GDC) 1 of Appendix A to 10 CFR Part 50 should be addressed by an applicant regarding the design, fabrication, and testing of structures, systems, and components important to safety to maintain quality standards. In particular, for GDC 1, it states that it is acceptable for an applicant to commit to meeting design, fabrication and testing guidance in NUEG-0554 for overhead handling systems and ANSI N14.6 or ASME B30.9 for lifting devices (Note NUREG-0554 and ANSI/ASME refer to NUREG-0612 seismic guidance). Section 9.1.5, “Overhead Heavy Load Handling Systems (OHLHS),” DCD, Revision 5, does not address how the design meets the GDC 1 criteria nor does it commit to meet GDC 1. Either specifically commit to meeting the above criteria for GDC 1, or identify the differences between the design features, analytical techniques, and procedural measures proposed for the ESBWR design and the SRP acceptance criteria and address how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with NRC regulations.

RAI 9.1-101	Kelly G	Address certain acceptance criteria involving GDC 2 for heavy loads	The Acceptance Criteria in Section 9.1.5, "Overhead Heavy Load Handling Systems," NUREG-0800, "Standard Review Plan" (SRP) cites that General Design Criteria (GDC) 2 of Appendix A to 10 CFR Part 50 should be addressed by an applicant regarding the ability of the structures, equipment, and mechanisms to withstand the effects of earthquakes. In particular, for GDC 2, it states that it is acceptable for an applicant to commit to meeting the relevant aspects of Position C.2 of Regulatory Guide 1.29, "Seismic Design Classification," and Section 2.5 of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." Section 9.1.5, "Overhead Heavy Load Handling Systems (OHLHS)," DCD, Revision 5, does not address Section 2.5 of NUREG-0554 in the context of GDC 2. Either commit to meeting the above criteria for GDC 2, or identify the differences between the design features, analytical techniques, and procedural measures proposed for the ESBWR design and the SRP acceptance criteria and address how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with NRC regulations.
RAI 9.1-102	Kelly G	Provide a discussion of the physical arrangement of heavy load handling systems	In the review procedure guidance, Section 9.1.5, "Overhead Heavy Load Handling Systems," NUREG-0800, "Standard Review Plan" (SRP), an applicant is expected to describe the physical arrangement of heavy load handling systems for stored fuel and safe shutdown equipment. DCD Section 9.1.5.9 states that, "The separation (arrangement, equipment interlocks, and routing) of redundant safety-related components in relation to heavy load paths minimizes the potential to cause failure of safety-related components"; however, no arrangement is provided. Section 9.1.5.4, "System Description," DCD, Revision 5 also does not provide a description of physical arrangements. Please provide these descriptions or address how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with NRC regulations.
RAI 9.1-103	Kelly G	Explain why the applicant didn't specify metallic material should be used for slings	In Section 9.1.5.2, "General," of the DCD, Revision 5, the applicant commits to having hoists, cranes, or other lifting devices comply with among other standards, ASME/ANSI B30.9. Subsection III.4.C.ii.(2) of Section 9.1.5 "Overhead Heavy Loads handling Systems," NUREG-0800, "Standard Review Plan," states, "[s]lings should satisfy the criteria of ASME B30.9 and be constructed of metallic material (chain or wire rope)." This criterion is supported by operating experience documented in NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002." The report cites various

			examples where Kevlar slings failed or separated causing a load drop. Please justify your choice to not specify metallic material (chain or rope) for construction of slings.
RAI 9.1-104	Kelly G	Provide missing discussion of quality assurance program to monitor heavy loads handling program	In Section 9.1.5.8, "Operation Responsibilities," DCD, Revision 5, the applicant lists requirements for COL applicants to comply with regarding a Quality Assurance program to monitor, implement, and ensure compliance with the heavy load handling program. In Subsection III.4.C.i of Section 9.1.5 "Overhead Heavy Loads Handling Systems," NUREG-0800, "Standard Review Plan," (SRP) states, "the program should include at least the following elements: (1) design and procurement document control; (2) instructions, procedures, and drawings; (3) control of purchased material, equipment, and services (See also Section 10 of NUREG-0554); (4) inspection; (5) testing and test control; (6) non-conforming items; (7) corrective action; and (8) records. This guidance is typically used for developing the QA program for a non-safety system. The listing in Section 9.1.5.8 of the DCD does not appear to comply with Subsection III.4.C.i of Section 9.1.5 of the SRP. Please incorporate the missing guidance or address how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with NRC regulations.
RAI 9.1-105	Kelly G	Address lifting device criteria for recurrent load movements	Subsection III.4.C.ii.(1) of Section 9.1.5 "Overhead Heavy Loads Handling Systems," NUREG-0800, "Standard Review Plan," (SRP) states, "[a] special lifting device that satisfies ANSI N14.6 should be used for recurrent load movements in critical areas (reactor head lifting, reactor vessel internals, spent fuel casks) (See also Section 5.1.6, NUREG-0612). The lifting device should have either dual, independent load paths or a single load path with twice the design safety factor specified by ANSI N14.6 for the load." Section 9.1.5.5, "Fuel Building and Reactor Building Cranes," of the DCD, Revision 5, is silent regarding the load paths and safety factors. Either modify the DCD Tier 2 to address lifting device criteria for the FB and RB cranes or address how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with NRC regulations.
RAI 9.1-106	Kelly G	Provide functional geometric layout of the fuel handling equipment and areas	Subsection I.1 of Section 9.1.4, "Light Load Handling System (Related to Refueling)," NUREG-0800, "Standard Review Plan (SRP)," states the staff should review, "[t]he design layout, which shows the functional geometric layout of the fuel handling equipment and areas" to determine if the various handling

			operations can be performed safely." SRP Section 9.1.4, Subsection III (1), "Review Procedures," also states that the Light Load Handling System (LLHS) physical arrangements for stored fuel and fuel handling areas are to be sufficiently described to establish that the various handling operations can be performed safely. Figures showing overall system arrangement, including reactor well, the buffer pool, the upper fuel transfer pool, the inclined fuel transfer pool, the fuel building storage pool, the spent fuel storage pool, the lower fuel transfer pool, cask pool, and the inclined fuel transfer system have not been provided by the applicant. Either modify the DCD Tier 2 to address the functional geometric layout of the fuel handling equipment and areas, or show how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with NRC regulations.
RAI 9.1-107	Kelly G	Address interlocks for light load handling systems for refueling	In Section 9.1.4, "Light Load Handling System (Related to Refueling)," NUREG-0800, "Standard Review Plan," (SRP), acceptance criteria for meeting the relevant requirements of GDC 61 and GDC 62 are based on meeting the guidelines of ANSI/ANS 57.1-1992. Table 1, "Required Interlock Protection," in ANSI/ANS-57.1-1992 provides interlock protection guidelines for each component of a fuel handling system. The application describes the interlocks associated with the refueling machine and fuel handling machine, which are not complete with respect to Table 1 listed above. Additionally, Table 1 lists interlock guidelines for equipment such as the fuel building crane, reactor building crane, fuel prep machine, control component change mechanism, inclined fuel transfer system, and the upenders, which are not described in the application. Please describe in the DCD how each required interlock specified in Table 1 of ANSI/ANS 57.1-1992 is applied for each of the components listed in Table 1 and provide a markup in the DCD Tier 2 showing the above requested information or show how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with NRC regulations.
RAI 9.1-108	Kelly G	Explain how interlocks prevent the fuel handling machine from exposing the operator to too much radiation	The Fuel Handling Machine is described in Section 9.1.4.5, "Refueling Equipment," of the DCD Tier 2, Revision 5. The Fuel Handling Machine transports spent fuel assemblies over and above the spent fuel racks. If the raised fuel assembly was too close to the water surface of the spent fuel pool (SFP), excessive radiation levels might occur on the fuel handling floor. Water volume shields radiation. GDC 61 requires the avoidance of excessive

			personnel radiation exposure. DCD Tier 2 Section 9.1.4.5 states that, "The grapple in its retracted position provides sufficient water shielding of at least 2591 mm (8.5 ft.) over the active fuel during transit." Note that the variance with RG 1.13 (8.5 feet of shielding vs. the recommended 10 ft) is addressed by RAI 9.1-50 supplement 2. DCD Tier 2 Section 9.1.4.18 states that, "the retraced position is controlled by both interlocks and physical stops to prevent raising the fuel assembly above the normal stop position required for safe handling of the fuel." However, these interlocks are not described. Please explain the operating interlocks for the Fuel Handling Machine, which ensure a spent fuel assembly is not raised above a specified water level in the SFP, such that radiation levels in the FB are as low as reasonably achievable (ALARA).
RAI 9.1-109	Kelly G	What standards and codes apply to RB auxiliary hoist and refueling platform auxiliary hoist?	Section 9.1.4.9, "In-Vessel Servicing Equipment," of the DCD, Revision 5, discusses moving the instrument strongback with the Reactor Building auxiliary hoist and the instrument handling tool with the refueling platform auxiliary hoist. Please modify Table 9.1-5, "Reference Codes and Standards," in the next revision to the DCD Tier 2 to identify to which standards and codes these hoists are to be constructed and operated.
RAI 9.1-110	Kelly G	How are ends of IFTS tube to be sealed off?	In Section 9.1.4.12, "Fuel Transfer System," DCD, Revision 5, it states that there is a means to seal off the upper and lower ends of the transfer tube while allowing filling and venting of the tube. Please explain how this is to be accomplished and the implications of failure of these seals in such a manner as to drain the tube while fuel is being transported in it.
RAI 9.1-111	Kelly G	What is basis for saying no modes of operation will allow unexpected draining of upper pool?	In Section 9.1.4.12, "Fuel Transfer System," DCD, Revision 5, it states that there are no modes of operation that allow simultaneous opening of any set of valves that could cause draining of water from the upper pool in an uncontrolled manner. Please explain the engineering basis for this assertion and whether this protection is single failure proof.
RAI 9.1-112	Kelly G	Did IFTS seismic analysis consider multiple fuel assemblies in the carriage when the SSE occurs?	In Section 9.1.4.12, "Fuel Transfer System," DCD, Revision 5, it states that the inclined fuel transfer system tubes and supporting structure can withstand an SSE without failure of the basic structure or compromising the integrity of adjacent equipment and structures. In Revision 5, Section 9.1.4.12 was changed to state that cooling is provide for two instead of one freshly removed fuel assemblies in the incline fuel transfer system. Please confirm in the DCD Tier 2

			whether the engineering basis for this assertion assumes at least two fuel assemblies are contained in the transport device (i.e., carriage) during the seismic event.
RAI 9.1-113	Kelly G	What controls are in place to keep personnel safe from radiation exposure from IFTS?	In Section 9.1.4.12, "Fuel Transfer System," DCD, Revision 5, it states that (1) controls prevent personnel from inadvertently or unintentionally being left in high radiation areas or areas immediately adjacent to the Inclined Fuel Transport System (IFTS) at the time the access doors are closed, (2) that during IFTS operation or shutdown, personnel are prevented from reactivating the IFTS while personnel are in the area or entering the controlled maintenance area while irradiated fuel or components are in any part of the IFTS. Please describe these controls in the next revision to the DCD Tier 2.
RAI 9.1-114	Kelly G	Add a COL Action Item to address spent fuel casks	Guidelines specified in Section 9.1.4, "Light Load Handling System Related to Refueling," NUREG-0800, "Standard Review Plan," (SRP) state that the objective of the review is to confirm that the light load handling system design precludes system malfunctions or failures that could cause criticality accidents, a release of radioactivity, or excessive personnel radiation exposures. Section 9.1.4, "Light Load handling System (Related to Refueling)," only indirectly addresses transfer of spent fuel to a cask. Section 9.1.4.3, "Spent Fuel Cask," of the DCD, Revision 5, states that spent fuel casks are not in the ESBWR standard plant scope. Provide a COL Action Item or a DCD Tier 1 interface item that requires a COL applicant to address spent fuel casks including identifying safety and non-safety related components, a description of the safety function of each safety related component, a discussion of the seismic capacity of the spent fuel cask system, a discussion of how the single failure criterion is satisfied, a discussion of how emergency cooling is accomplished, a discussion of the need for emergency cooling of spent fuel casks, and a discussion of interlocks.
RAI 9.1-115	Kelly G	Provide elevation diagrams of pools in RB and FB	The staff's review of pools that could potentially contain new or spent fuel includes consideration of events that could lower the level of the pools and uncover fuel. Please provide an elevation diagram of the spent fuel storage pool, lower fuel transfer pool, and cask pool, including any pits in the pools and interfaces (e.g., gates or weirs) between/among the pools or pathways that could potentially lower the water level in the pools to unacceptable levels. Similarly, provide an elevation diagram of the buffer pool, reactor well, upper fuel transfer

			pool, inclined fuel transfer system, and equipment storage pool, as well as any interfaces or pits in the pools. Elevation diagrams should indicate the height of the top of fuel in any pool that could contain fuel.
RAI 14.3-441	Kelly G	Add “single failure proof” to the Tier 1 requirements for the RB and FB cranes, as well as the equipment hatch hoist and the containment maintenance hatch hoist	Table 9.1-5 “Reference Codes and Standards” of the ESBWR DCD, Revision 5 states that NUREG-0554, “Single Failure Proof Cranes for Nuclear Power Plants,” is “[a]pplicable to the RB and FB overhead cranes. Applicable to the hoist on the refueling and fuel handling machines that handles the combined fuel support and control blade grapple.” The ESBWR DCD, Tier 1 Section 2.16.1, “Cranes, Hoists, and Elevators,” and Table 2.16.1-1, “ITAAC For The Cranes, Hoists and Elevators” do not list “single failure proof” as certified design information with ITAAC for the RB crane, the FB crane, the hoist for the refueling machine or the hoist for the fuel handling machine. The staff believes that “single failure proof” design criteria for the above listed cranes and hoists should be listed in Tier 1 as described below. One design criteria, among several design criteria for Tier 1 information, is that it should include features and functions that could have a significant effect on the safety of a nuclear plant or are important in preventing or mitigating severe accidents. A drop of the reactor vessel head or a spent fuel cask could affect plant safety. Therefore, design features that reduce the risk and/or analyses that provide assurance of safety after a dropped load are important to safety. The staff considers “single failure proof” design criteria for the RB crane and the FB crane as Tier 1 safety significant design criteria. As a minimum, the following analyses would have to be performed in order to not consider “single failure proof” design criteria as safety significant criteria for the RB crane and the FB crane: • A heavy load analysis proving that a heavy load drop in safety related areas of the plant will not be the cause any of Items I through IV of section 5.1 of NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants.” • SRP 9.1.5, “Overhead Heavy Load Handling Systems,” Subsection III. 4, states that without “single failure proof” design criteria, analyses are required for a dropped load on the reactor vessel, among other analyses. The DCD does not describe results of this analysis.

			• Regulatory Guide 1.13, Regulatory Position C.5 states that an alternative to an "single failure proof" crane is that the spent fuel cask loading area be designed to withstand a drop of the heaviest load at the maximum height. As a minimum, the following analysis would have to be performed in order to not consider "single failure proof" design criteria as safety significant criteria for the containment equipment hatch hoist and the containment maintenance hatch hoist: A heavy load analysis proving that a heavy load drop in safety related areas of the plant will not be the cause any of Items I through IV of Section 5.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." Without the analyses and design criteria stated above, the "single failure proof" design feature for the RB crane, FB crane, the hoist for the refueling machine and the hoist for the fuel handling machine becomes safety significant design criteria. Please justify why the applicant did not include "single failure proof" design criteria and ITAAC in Tier 1 of the DCD, which are safety significant design criteria, for the RB crane, FB crane, the hoist for the refueling machine, and the hoist for the fuel handling machine.
RAI 14.3-442	Kelly G	Explain why some SFP and buffer pool design features do not appear in ITAAC	10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC regulations. Important to safety functions should be described in the DCD Tier 1. Based on the review of DCD Tier 2 section 9.1, several apparent important to safety design features have been omitted from Tier 1. Please explain why the applicant did not require that the following design features be added to ITAAC or specified as Tier 1 material:

			* The spent fuel pool and buffer pool are reinforced concrete structures with a stainless steel liner. * Spent fuel pool and buffer pool liner embedments are designed to meet seismic Category I requirements. * The bottoms of the spent fuel pool and buffer pool gates are higher than the minimum water level required over the spent fuel storage racks to provide adequate shielding and cooling. * Lines to fill and drain the spent fuel pool and buffer pool enter the pools above the safe shielding water level. * Redundant anti-siphon vacuum breakers are located at the high point of the pool lines in the spent fuel pool and the buffer pool to preclude a pipe break from siphoning the water from the pools and jeopardizing the safe water level. * Individual spent fuel racks are spaced less than one fuel assembly apart so that a fuel assembly cannot be inserted between racks. * Materials used for construction of the spent fuel pool and buffer pool are specified in accordance with the latest issue of applicable ASTM specifications at the time of equipment order.
RAI 14.3-443	Kelly G	Explain why aspects of the FAPCS design criteria don't appear in ITAAC	10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC regulations. Important to safety functions should be described in the DCD Tier 1. Based on the review of DCD Tier 2 section 9.1, several apparent important to safety design features have been omitted from Tier 1. Please explain why the applicant did not require that the following Fuel and Auxiliary Pools Cooling

			System (FAPCS) design criteria be included in ITAAC or specified as Tier 1 material: * The FAPCS consists of two physically separated cooling and cleanup trains. * FAPCS is designed to provide drywell spray and alternate shutdown cooling. * In Section 9.1.3.1, "System Description," for FAPCS in DCD, Revision 5, it describes the portions of the FAPCS that are not specifically defined as safety related as being seismic Category II. This quality is not mentioned in Table 2.6.2-2, "ITAAC For The Fuel and Auxiliary Pools Cooling Cleanup System." * All piping between the RWCU/SDC System and the nonsafety-related check valves (upstream of the MOVs) is designed to withstand the full reactor pressure. * With the exception of the suppression pool suction line, anti-siphoning devices are used on all submerged FAPCS piping to prevent unintended drainage of the pools. * The suppression pool suction line is conservatively designed to preclude a rupture between the pool and the containment isolation valves. * The electrical power supplies, control and instrumentation of the two FAPCS trains and their supporting systems are electrically and physically separated. * Piping and components completely separate from FAPCS pool cooling piping provide flow paths for post-accident makeup water transfer.
RAI 14.3-444	Kelly G	Clarify the ITAAC for FAPCS seismic Category I piping in Tier 1.	In Section 9.1.3.1, "System Description," for the Fuel and Auxiliary Pools Cooling System (FAPCS) in DCD, Revision 5, it describes FAPCS as being a nonsafety-related system with the exception of the piping and components required for containment isolation, refilling the IC/PCC pools and spent fuel pool, and interface with the Reactor Water Cleanup/Shutdown Cooling system. Seismic Category I piping is shown on DCD Tier 1, Figure 2.6.2-1 but it is not listed in DCD Tier 1, Table 2.6.2-1. DCD Tier 1 Table 2.6.2 design commitments

			2, 3, and 4 provide ITAAC for Seismic Category 1 piping identified in DCD Tier 1, Table 2.6.2-1, but no piping is so identified. Revise DCD Tier 1 Section 2.6.2 ITAAC for Seismic Category 1 piping to reference Figure 2.6.2-1 or modify Table 2.6.2-1.
RAI 14.3-445	Kelly G	Why aren't certain of the LLHS design features in ITAAC or Tier 1?	10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC regulations. Important to safety functions should be described in the DCD Tier 1. Based on the review of DCD Tier 2 section 9.1, several apparent important to safety design features have been omitted from Tier 1. Explain why the applicant did not require that the following design features in the Light Load Handling System (LLHS) be added to ITAAC or specified as Tier 1 material: * The reactor pressure vessel head strongback is designed to meet NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" and ANSI-14.6, "Standard for Special Lifting Devices", and is tested in accordance with American National Standard for Overhead Hoists ASME/ANSI B30.16, Paragraph 16-1.2.2.2. The reactor pressure vessel head strongback is designed so that no single component failure causes the load to drop or swing uncontrollably out of the safety-related horizontal attitude.
RAI 14.3-446	Kelly G	Add certain interlocks to ITAAC for FB handling machine	10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC regulations. Important to safety functions should be described in the DCD Tier 1.

			Table 2.5.5-1, "ITAAC for Refueling Machine," lists a few interlocks that the FB fuel handling machine will have. Add interlocks to this list based on disposition of RAI 9.1-109 which addresses interlocks for fuel handling system that are specified in Table 1 of ANSI/ANS 57.1-1992.
RAI 14.3-447	Kelly G	Why aren't certain of the OHLHS design features in ITAAC or Tier 1?	10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC regulations. Important to safety functions should be described in the DCD Tier 1. Based on the review of DCD Tier 2 section 9.1, several apparent important to safety design features have been omitted from Tier 1. Please explain why the applicant did not include the following design features for the Overhead Heavy Load Handling System (OHLHS) in ITAAC or specify them as Tier 1 material: * Cranes and hoists, or monorail hoists pass over the centers of gravity of heavy equipment that is to be lifted. * Because the Passive Containment Cooling (PCC) and Gravity Driven Cooling System (GDCS) piping and valves are spatially separated, an inadvertent load drop that breaks more than one pipe of valve in the PCC or GDCS is not credible. * The arrangement of the refueling floor precludes transporting heavy loads, other than spent fuel handled by the refueling machine or fuel handling machine, over spent fuel stored in the spent fuel storage pool.

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