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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 377 Related to ESBWR Design Certification Application – Fuel Pools – RAI Number 9.1-128**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) response to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) 9.1-128 sent by NRC Letter No. 377, Reference 1.

GEH response to RAI Number 9.1-128 is addressed in Enclosure 1. Enclosure 2 contains the DCD markups associated with this response.

If you have any questions or require additional information, please contact me.

Sincerely,

FOR

Richard E. Kingston
Vice President, ESBWR Licensing

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NRO

Reference:

1. MFN 09-645, Letter from U.S. Nuclear Regulatory Commission to Jerald G. Head, *Request for Additional Information Letter No. 377 Related to ESBWR Design Certification Application*, October 13, 2009

Enclosures:

1. Response to Portion of NRC Request for Additional Information Letter No. 377 Related to ESBWR Design Certification Application – Fuel Pools - RAI Number 9.1-128
2. Response to Portion of NRC Request for Additional Information Letter No. 377 Related to ESBWR Design Certification Application – Fuel Pools - RAI Number 9.1-128 – DCD Markups

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Enclosure 1

MFN 09-697

Response to Portion of NRC Request for Additional Information Letter No. 377 Related to ESBWR Design Certification Application

Fuel Pools

RAI Number 9.1-128

NRC RAI 9.1-128

Applicants for standard design certifications are required by 10 CFR 52.47(a)(22) to address operating experience insights. Inspection and Enforcement (IE) Bulletin 84-03, "Refueling Cavity Water Seal," was issued to address the potential failure of refueling cavity seals to assure that fuel uncovery during refueling remains an unlikely event. The bulletin required licensees to evaluate the potential for and consequences of a refueling cavity water seal failure. Additional information concerning refueling cavity seal failures was provided by Information Notice (IN) 84-93, "Potential for Loss of Water from the Refueling Cavity." IN 84-93 also noted that refueling cavities can be drained due to failures associated with other seals and as a consequence of valve misalignments. Therefore, in order to adequately address operating experience considerations and in accordance with the requirement specified by 10 CFR 52.47(a)(22), the following additional information is required:

a. Describe the design and installation of the refueling cavity seal and any other seals that will be used and whose failure could cause the refueling cavity to drain.

b. For each of the seals identified in (a), describe measures that will be implemented to ensure that the seals remain intact and do not become degraded over time.

c. For each of the seals identified in (a), evaluate the potential for and consequences of seal failure. These evaluations should address the following considerations:

- *seal failure modes (including impact by dropped fuel bundles and weld failures) and the maximum leak rate that can occur;*
- *the refueling cavity makeup capability that is assured by Technical Specifications or availability controls while in Mode 6;*
- *operator actions that are credited, including indication and alarms that are available to alert operators of the problem, and the time needed for operators to complete the required actions assuming that actions are not initiated until ten minutes after an alarm is sounded;*
- *the impact on stored fuel, fuel in transit or otherwise located in the refueling cavity for other reasons, and fuel in the reactor vessel, including the minimum height of water that will remain above the fuel and the basis for this determination; and*
- *the capability to isolate the fuel transfer tube with the maximum radiation level and flow rate of water through the transfer tube that are anticipated as a result of the seal failure.*

d. Other than the seals that are referred to in (a), identify all of the paths that are capable of inadvertently draining the refueling cavity, describe controls that will be established to prevent inadvertently draining the refueling cavity through these *paths, and evaluate the potential for and consequences of the refueling cavity to drain through these paths (similar to the evaluation referred to in (c)).*

e. *Describe actions that must be taken to restore containment integrity when in Mode 6, the time required to complete these actions, the capability to implement these actions during and/or following situations that cause the refueling cavity to drain, and controls that will be established to ensure that containment integrity can be restored as described.*

f. *Revise the Design Control Document (DCD) to adequately describe the licensing basis for the certified plant design with respect to the above considerations. Establish inspections, tests, analyses, and acceptance criteria (ITAAC), interface requirements, and combined license (COL) action items as appropriate for design features, procedures and controls that are important to ensure that occupational exposures and the release of radioactive material will not exceed NRC requirements as a consequence of inadvertently draining the refueling cavity.*

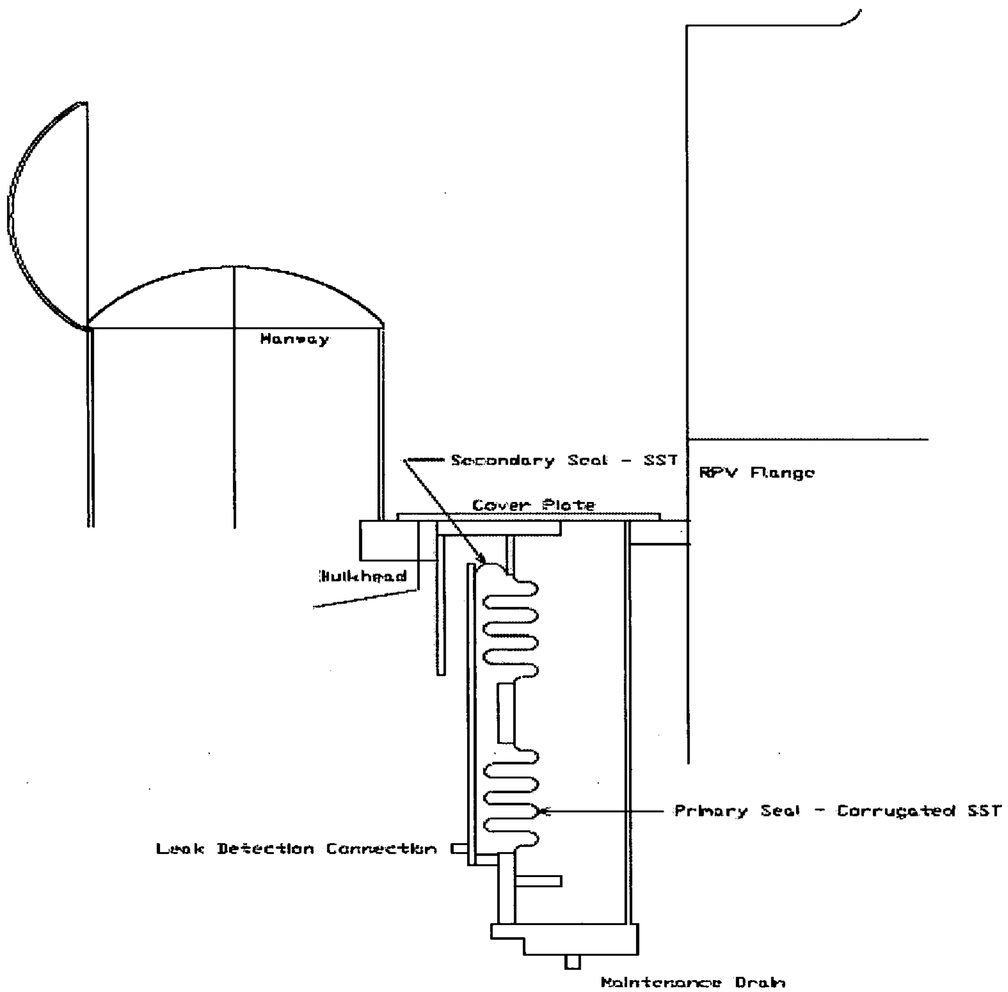
<u>GEH Response</u>

a) Design of the ESBWR refueling bellows will not vary in concept from the bellows assemblies currently in service in the GE BWR operating fleet, as operating experience has demonstrated this design to be robust with no history of significant failure. The bellows assembly is a permanent installation, unlike some bellows designs that are only installed for refueling. All bellows assembly components are steel, with the actual bellows seals fabricated from 316L stainless steel for corrosion resistance. As seen in the figure below, the bellows primary seal is corrugated and extends from the lower flange of the reactor vessel to the interior of the reactor cavity. All the component connections in the design are welded, and leakage can only occur if there is a through wall defect.

This design facilitates vessel movement due to thermal expansion. The entire refueling bellows assembly is located below the reactor vessel flange such that it cannot interfere with removal of core components for refueling. Normally closed drains on the wet side of the bellows are provided to support maintenance. A spring loaded secondary seal is provided to prevent leakage into the drywell in the event a leak occurs through the primary seal.

As committed per RAI 16.2-77 (MFN 07-247), the ESBWR bellows assembly is designed to meet Seismic Category I requirements and a rapid loss of coolant through the seals following a seismic event is not credible.

The ESBWR will not utilize any other seals whose credible failure (non catastrophic) could cause the refueling cavity to rapidly drain.



Bellows Assembly

b) As seen in the figure above, cover plates are provided for the bellows, which protect against anything (e.g., fuel assembly) that may be dropped during refueling. The cover plates remain in place during operation. The bellows are installed vertically beneath a plate welded to the bulkhead, which provides further protection in the event the cover plates are removed for cleaning or inspection. The bellows supplier shall provide guidance for routine maintenance and inspection of the bellows. The refueling bellows seals are fabricated from 316L stainless steel for corrosion resistance. Normally open leak detection connections are located on the dry side of the bellows for continuous monitoring.

c) As the bellows configuration is not susceptible to dropped objects, any seal failures are likely related to normal wear or fatigue. All welding procedures and welder qualifications shall be in accordance with ASME IX, with any exceptions approved by GEH prior to execution. Any leakage due to fatigue or weld failure will not result in a rapid drain down event and will be easily detectible and isolable. As mentioned previously, leak detection is provided on the dry side of the bellows and a second seal is in place that prevents leakage into the drywell.

In the event of a leak through the bellows, makeup to the reactor cavity, as required, is available through either the Fuel and Auxiliary Pools Cooling System (FAPCS) or the Fire Protection System (see DCD Section 9.1.3.2). Either system can provide more than sufficient flow to mitigate the cavity leaks considered in this evaluation.

If a fuel assembly were to be in transit when a leak in the refueling bellows is discovered, that assembly could be safely returned to its former location in the core or placed in the deep pit of the buffer pool, where high-density fuel storage racks are located. Safe placement of the fuel assembly in transit can occur within approximately 4 minutes of notification. Refueling activities would resume when determined safe to do so and the Technical Specification LCO on pool level is satisfied. The deep pit of the buffer pool is separated from the reactor cavity by the shallow region of the buffer pool, such that approximately 6 meters of water is present over the top of active fuel if the cavity were to drain. Sufficient water volume exists to ensure complete coverage of the fuel. Both FAPCS and the Fire Protection System are available makeup sources to the deep pit of the buffer pool.

A two-assembly-out-of-core configuration on the refuel floor is not anticipated as there is no fuel preparation machine in the buffer pool. Tasks such as fuel sipping, rechanneling, and fuel inspections will take place in the spent fuel pool, requiring that the fuel assembly subject to work be transferred from the reactor building to the fuel building.

Evaluation of the Inclined Fuel Transfer System (IFTS) has been performed through resolution of numerous RAIs, including 16.2-77, 9.1-35, 9.1-110, 9.1-111, and 9.1-112, which deal with drain down, the IFTS operation sequence, and seismic classification. IFTS components that ensure integrity of fuel assemblies in transit have been designated as Seismic Category I. As stated in the response to RAI 16.2-77 (MFN 07-247), there are no modes of operation that will allow simultaneous opening of any set of valves that could cause draining of the water from the upper pool in an uncontrolled manner.

Pool level on the refuel floor is constantly monitored and alarms are provided in the event level drops. The buffer pool is equipped with safety related water level sensors that alarm the operators that the pool level is below normal. Operators also have the ability to provide makeup water and suspend refueling operations, as needed. In addition to pool level alarms, the drywell sump also provides alarms in the event excess water is present.

d) In addition to the refueling bellows and IFTS, additional potential cavity drain down paths were considered. In each case, the path was deemed incapable of a catastrophic draining of the reactor cavity. Any leakage through these paths does not occur at a rate that exceeds makeup capability until leak isolation can occur. Therefore, there is no danger of uncovering the core or exposing fuel in transit. Drain down paths due to pipe breaks were not considered.

Manways are provided between the reactor cavity and the drywell. Prior to flooding the reactor for refueling, the manways are closed per procedure. The manway covers utilize gaskets or o-rings to provide proper sealing. Manways have leaked in the past and improvements are being incorporated in operating plants to reduce occurrences. However, the loss of water through a manway cover is minimal and presents no danger to uncovering the fuel. The water is captured in the drywell just below the manway and drained to the drywell sump. This event typically has no impact on refueling.

Plant operations with potential to drain the reactor vessel were considered and broken into two categories: process valve misalignments and maintenance configurations.

As process systems are normally aligned for operation, shutdown, refueling, startup, etc., potential exists for valve misalignment such that water could be drained from the reactor vessel. However, various system configurations are controlled by procedures that are evaluated to eliminate this potential.

Maintenance boundaries are established to ensure both a safe work environment and a safe plant during the maintenance activity. Operating experience in establishing maintenance boundaries at GE BWRs has proven to be effective in preventing adverse maintenance configurations. Maintenance is performed within Lockout/Tagout boundaries that are established to provide this assurance. Each maintenance activity is individually evaluated and the appropriate boundaries defined and established. These boundaries are controlled both procedurally and in accordance with the site's Lockout/Tagout program. Manipulation of boundary components such as valves, breakers, etc. is strictly forbidden.

Routine maintenance of valves adjacent to the reactor vessel (SRVs, MSIVs) occurs during refueling outages in parallel with fuel movement. For the ESBWR, the main steam and isolation condenser lines will be plugged, as required, to allow for maintenance activities. The plugs are designed with redundant seals (one pneumatic and one mechanical) such that failure of one will not degrade the maintenance boundary and allow loss of water from the reactor vessel. Each seal on each plug is individually leak tested prior to the outage to ensure its operability. All lines that interface with the ESBWR reactor vessel are required to have maintenance isolation valves adjacent to the nozzle. Therefore, freeze seals are not permitted and do not present a drain down potential.

Additionally, the ESBWR has several design features that provide margin or mitigate the potential for a drain down of the reactor cavity as compared to typical BWR plants. These include:

- Substantially increased water volume above the core due to the increased vertical height of the reactor pressure vessel,
- No large nozzle penetrations are present below the core,
- Spent fuel is not stored in the reactor building except during reactor shutdown, when it is temporarily stored in the deep pit that is separate from the reactor cavity (see DCD Section 9.1),
- FMCRD maintenance is not anticipated to inadvertently drain the RPV, as discussed in DCD Subsection 4.6.2.1.4,
- The buffer pool has safety-related water level sensors that automatically detect a drop in water level (see DCD Subsection 9.1.3.5),
- The movement of fuel assemblies is faster due to the state of the art refueling equipment.

Based on the rigor associated with defining and establishing maintenance boundaries, coupled with the types of devices used to establish those boundaries, a rapid drain down of the reactor vessel is not credible. In addition, makeup systems are available to mitigate a slow loss of vessel water inventory and fuel can be placed in a safe configuration within approximately 4 minutes of notification.

e) The following conclusions have been defined for a reactor cavity drain down event for the ESBWR:

1) A rapid cavity drain down event for the ESBWR is not credible.
2) The ESBWR design does not introduce any new potential drain down paths and all potential drain down paths result in a slow loss of cavity water inventory.
3) Makeup systems (FAPCS and Fire Protection) have the capacity to ensure fuel in the core remains covered.
4) Fuel in transit when a leak is identified can be placed in a safe location in the core or in the deep pit of the buffer pool.
5) Core components stored in the pools for refueling (dryer, separator, chimney partitions) will not become uncovered as a result of a slow loss of reactor cavity water, therefore, dose consequences associated with exposure of these components are not considered. It should be noted that dryers and separators are temporarily brought out of the water during removal/installation at most operating BWRs to facilitate transfer into and out of the equipment storage pool.

6) The need to restore containment integrity within the allotted time of 30 minutes is not anticipated as a result of a slow loss of reactor cavity water.

f) The DCD will be revised to point out that the all-steel refueling bellows assembly is a permanent installation and seals are fabricated from stainless steel. Additionally, the DCD will be revised to include the conclusions described in the paragraph above. An ITAAC is not necessary since the functionality and structural integrity of the refueling bellows is demonstrated by filling the reactor cavity prior to fuel load. With regard for the need for a COL action item, there is no case where a credible cavity drain down is possible that could result in a safety concern for a fuel assembly radiation release. Therefore, no COL action items are necessary.

DCD Impact

DCD Tier #2, Section 6.2.1.1.2 will be revised such that Revision 7 reflects the attached markup to enhance the description of the refueling bellows assembly.

DCD Tier #2, Section 12.4.4 will be revised such that Revision 7 reflects the attached markup to discuss drain down of the reactor cavity.

Enclosure 2

MFN 09-697

Response to Portion of NRC Request for Additional Information Letter No. 377 Related to ESBWR Design Certification Application

Fuel Pools

RAI Number 9.1-128

DCD Markups

- The containment structure shall withstand coincident fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment.
- The containment structure shall accommodate flooding to a sufficient depth above the active fuel to maintain core cooling and to permit safe removal of the fuel assemblies from the reactor core after the postulated DBA.
- The containment structure shall be protected from or designed to withstand hypothetical missiles from internal sources and uncontrolled motion of broken pipes, which could endanger the integrity of the containment.
- The containment structure shall direct the high energy blowdown fluids from postulated LOCA pipe ruptures in the DW to the pressure suppression pool and through the PCCS condensers.
- The containment system shall allow for periodic tests at the calculated peak or reduced test pressure to measure the leakage from individual penetrations, isolation valves and the integrated leakage rate from the containment structure to confirm the leak-tight integrity of the containment.
- The Containment Inerting System establishes and maintains the containment atmosphere to $\leq 3\%$ by volume oxygen during normal operating conditions to ensure inert atmosphere operation.
- PCCS shall remove post-LOCA decay heat from the containment for a minimum of 72 hours, without operator action, to maintain containment pressure and temperature within design limits.

6.2.1.1.2 Design Features

The containment structure is a reinforced concrete cylindrical structure, which encloses the Reactor Pressure Vessel (RPV) and its related systems and components. Key containment components and design features are exhibited in Figures 6.2-1 through 6.2-5. The containment structure has an internal steel liner providing the leak-tight containment boundary. The containment is divided into a DW region and a WW region with interconnecting vent system. The functions of these regions are as follows:

- The DW region is a leak-tight gas space, surrounding the RPV and reactor coolant pressure boundary, which provides containment of radioactive fission products, steam, and water released by a LOCA, prior to directing them to the suppression pool via the DW/WW Vent System. A relatively small quantity of DW steam is also directed to the PCCS during the LOCA blowdown.
- The WW region consists of the suppression pool and the gas space above it. The suppression pool is a large body of water to absorb energy by condensing steam from SRV discharges and pipe break accidents. The pool is an additional source of reactor water makeup and serves as a reactor heat sink. The flow path to the WW is designed to entrain radioactive materials by routing fluids through the suppression pool during and following a LOCA. The gas space above the suppression pool is leak-tight and sized to collect and retain the DW gases following a pipe break in the DW, without exceeding the containment design pressure.

The DW/WW Vent System directs LOCA blowdown flow from the DW into the suppression pool.

The containment structure consists of the following major structural components: RPV support structure (pedestal), diaphragm floor separating DW and WW, suppression pool floor slab, containment cylindrical outer wall, cylindrical vent wall, containment top slab, and DW head. The containment cylindrical outer wall extends below the suppression pool floor slab to the common basemat. This extension is not part of containment boundary; however, it supports the upper containment cylinder. The reinforced concrete basemat foundation supports the entire containment system and extends to support the RB surrounding the containment. The refueling bellows is an all steel, permanent installation with primary and secondary seals that are fabricated from stainless steel. The refueling bellow extends from the lower flange of the reactor vessel to the interior of the reactor cavity. This extension is also not part of the containment boundary, however, it provides a Seismic Category I seal between the upper DW and reactor well during a refueling outage.

The design parameters of the containment and the major components of the containment system are given in Tables 6.2-1 through 6.2-4. A detailed discussion of their structural design bases is given in Section 3.8.

Drywell

The DW (Figure 6.2-1) comprises two volumes: (1) an upper DW volume surrounding the upper portion of the RPV and housing the main steam and feedwater piping, GDCS pools (see Figure 6.2-3 for pool arrangement) and piping, PCCS piping, ICS piping, SRVs and piping, Depressurization Valves (DPVs) and piping, DW coolers and piping, and other miscellaneous systems; and (2) a lower DW volume below the RPV support structure housing the lower portion of the RPV, fine motion control rod drives, other miscellaneous systems and equipment below the RPV, and vessel bottom drain piping.

The upper DW is a cylindrical, reinforced concrete structure with a removable steel head and a diaphragm floor constructed of steel girders with concrete fill. The RPV support structure separates the lower DW from the upper DW. There is an open communication path between the two DW volumes via upper DW to lower DW connecting vents, built into the RPV support structure. Penetrations through the liner for the DW head, equipment hatches, personnel locks, piping, electrical and instrumentation lines are provided with seals and leak-tight connections.

The DW is designed to withstand the pressure and temperature transients associated with the rupture of any primary system pipe inside the DW, and also the negative differential pressures associated with containment depressurization events, when the steam in the DW is condensed by the PCCS, the GDCS, the FAPCS, and cold water cascading from the break following post-LOCA flooding of the RPV.

For a postulated DBA, the calculated DW pressure in Table 6.2-5 is below the design value shown in Table 6.2-1. The structure stresses are evaluated in Section 3G.5 considering the DW fluid temperature transients for multiple break locations.

Three vacuum breakers are provided between the DW and WW. The vacuum breaker is a process-actuated valve, similar to a check valve (Figure 6.2-28). The purpose of the DW-to-WW vacuum breaker system is to protect the integrity of the diaphragm floor slab and vent wall

More of the radwaste operations involve remote handling than in a typical BWR. General RW work consists of pump and valve maintenance, shipment handling, radwaste management and general cleanup activity. Maintenance collective dose estimates are captured in Section 12.4.2. The LWMS collects liquid wastes from equipment drains, floor drains, filter backwashes and other sources within the facility. Some examples of SWMS activities include movement of casks and liners, filter handling, resin transport, and movement or reconfiguration of radwaste processing skids. Generally, much of the activity is remotely performed and controlled by operators in the RW Control Room. Dose estimates for the collection, packaging and shipment of radwaste quantities are based on the assumptions below.

Operation of the RW Control Room is assumed to occur approximately once per day for one shift with a maximum dose rate of 10 μSv/hr (1 mrem/hr). Processing and packaging of DAW is assumed to occur once a day for two hours using two workers in a dose field of 50 μSv/hr (5 mrem/hr) as appropriate. This activity is assumed to occur three times per week. Shipments of concentrated wet solid waste in HICs are assumed to occur once per week for four hours in a dose field of 50 μSv/hr (5 mrem/hr) as appropriate using four workers. DAW shipments are assumed to occur once per month for eight hours in a dose field of 50 μSv/hr (5 mrem/hr) as appropriate using three workers. Finally, miscellaneous activities in high dose rate areas such as valve lineups or filter changes are assumed to be required approximately 4 person-hours per week in an average dose rate field of 150 μSv/hr (5 mrem/hr) as appropriate.

The estimated annual collective doses associated with waste processing operations appear in Table 12.4-4.

12.4.4 Refueling Operations

In the ESBWR design, refueling operations are conducted in two general areas. The Fuel Building houses the SFP and various equipment used for the receipt of new fuel assemblies. Space is also provided for fuel assembly receipt inspection and the installation of fuel channels on the new fuel assemblies. When new fuel assemblies are readied for transfer to the reactor, the assemblies are transferred using the IFTS, which is located in the upper portion of the RB. Here the new fuel assemblies are kept in the buffer pool until the refueling outage. This dual pool system is similar to that implemented in the BWR/6 product line. During the refueling outage, new fuel assemblies are placed in the reactor core using the refueling machine, the core shuffled, and spent fuel removed to the RB buffer pool. At this time, control rods or other in-core components may be replaced. Spent fuel assemblies, which were removed from the core, are then transferred through the IFTS to the FB SFP. The Fuel Building also contains facilities for the transfer of spent fuel assemblies into casks for possible storage at an onsite independent spent fuel storage installation.

Prior to commencing refueling operations, the drywell and reactor vessel heads must be disassembled and removed. Reactor vessel access and reassembly exposure times are reduced by use of a special stud tensioner for the 84 RPV head bolts. Underwater transfer of the dryer, chimney/partitions, and chimney head/separator decreases exposures during refueling operations. The improved fuel inspection equipment and increased use of remote operations significantly reduce the refueling floor exposure. Drywell access and RPV disassembly and reassembly in conventional BWRs typically require 4,500 person-hours of work at an effective dose rate of 30 μSv/hr. The ESBWR work involves the use of an automated stud tensioner for the RPV top head. This equipment, coupled with other automatic equipment available, is estimated to reduce the drywell access and RPV vessel disassembly/reassembly time to 1,200 person-hours.

ESBWR refueling is accomplished via the refueling bridge. Time for refueling operations including control rod replacement is reduced from a typical BWR value of 4,400 person-hours to approximately 1,500 person-hours. It is estimated that approximately 500 person-hours are spent in transfer of the spent fuel and other replaced components to the buffer pool in the RB, and 1,000 person-hours are spent transferring the spent fuel and removed components through the IFTS to the SFP in the FB. General area RB refueling floor and FB radiation zone effective values of 25 μSv/hr (2.5 mrem/hr) are used for the dose projections. An additional 4,000 person-hours are estimated for an optional spent fuel transfer campaign into storage casks for possible onsite storage in an independent spent fuel storage installation. Because cask loading operations are conducted entirely underwater, an effective dose rate value of 5 μSv/hr is used for the cask loading and transfer process. The total person-Sv associated with the above refueling operations is shown in Table 12.4-5.

During refueling operations, a rapid drain down of the reactor cavity is not credible for the ESBWR. The ESBWR does not introduce any new potential drain down paths and all potential drain down paths result in a slow loss of cavity water. The Fuel and Auxiliary Pool Cooling System and Fire Protection System provide make up capability and have the capacities to ensure fuel in the core remains covered. Fuel in transit when a leak is identified can be placed in a safe location in the core or in the deep pit of the buffer pool. Core components stored in the pools for refueling (i.e. steam dryer, separator, chimney partitions) will not become uncovered as a result of a slow loss of reactor cavity water, therefore, dose consequences associated with exposure of these components are not considered. The need to restore containment integrity is not anticipated as a result of a slow loss of reactor cavity water.