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MFN 06-371 Docket No. 52-010

October 18, 2006

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **Response to Portion of NRC Request for Additional Information Letter No. 60 – Radiation Protection – RAI Numbers 12.2-16 through 12.2-18, 12.3-6, 12.4-2, 12.4-10, 12.4-12, 12.4-21, 12.4-27, 12.5-5 and 12.7-3**

Enclosure 1 contains GE's response to the subject NRC RAIs transmitted via the Reference 1 letter.

If you have any questions about the information provided here, please let me know.

Sincerely,

Kathy Sedney for

David H. Hinds
Manager, ESBWR

D068

Reference:

1. MFN 06-342, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 60 Related to ESBWR Design Certification Application*, September 18, 2006

Enclosure:

1. MFN 06-371 – Response to Portion of NRC Request for Additional Information Letter No. 60 – Radiation Protection – RAI Numbers 12.2-16 through 12.2-18, 12.3-6, 12.4-2, 12.4-10, 12.4-12, 12.4-21, 12.4-27, 12.5-5, and 12.7-3

cc:	AE Cubbage	USNRC (with enclosures)
	GB Stramback	GE/San Jose (with enclosures)
	eDRF	0059-2467

ENCLOSURE 1

MFN 06-371

Response to Portion of NRC Request for

Additional Information Letter No. 60

Related to ESBWR Design Certification Application

Radiation Protection

RAI Numbers 12.2-16 through 12.2-18, 12.3-6, 12.4-2, 12.4-10, 12.4-12, 12.4-21, 12.4-27, 12.5-5 and 12.7-3

NRC RAI 12.2-16

DCD Tier 2, Section 12.1.2.1 refers to design features that resulted from the application of the as low as reasonably achievable (ALARA) considerations during plant design such that the ESBWR can be operated and maintained ALARA. However, the list of examples on the bottom of page 12.1-2 only refers to those features that assist in maintaining doses ALARA during decommissioning. Provide a description of the current operational experience considered and those general design features employed in the ESBWR design to maintain doses received during normal power and shutdown operations ALARA.

GE Response

A detailed description and examples of general design features to maintain doses ALARA during normal and shutdown operations are included in DCD subsection 12.3.1. Most of the design features take into account operational experience, such as minimizing pump maintenance and personnel exposure times by using quick-change cartridge-type seals on pumps. Another example is locating major HVAC equipment in dedicated low radiation areas to minimize exposures to maintenance personnel.

The revision to DCD subsection 12.1.2.1 will be incorporated in Revision 3 as shown in the attached markup.

12.1.1.3.3 Compliance with Regulatory Guide 1.8

Out of ESBWR Standard Plant scope. See Subsection 12.1.4.2 for COL license information.

12.1.2 Design Considerations

This subsection discusses the methods and features by which the policy considerations of Subsection 12.1.1 are applied. Provisions and designs for maintaining personnel exposures ALARA are presented in detail in Subsections 12.3.1 and 12.3.2.

12.1.2.1 General Design Consideration for ALARA Exposures

General design considerations and methods employed to maintain in-plant radiation exposures ALARA, consistent with the recommendations of Regulatory Guide 8.8, have two objectives:

- Minimizing the necessity for and amount of personnel time spent in radiation areas, and
- Minimizing radiation levels in routinely occupied plant areas in the vicinity of plant equipment expected to require personnel attention.

Both equipment and facility designs are considered in maintaining exposures ALARA during plant operations. Events considered include normal operation maintenance and repairs, refueling operations and fuel storage, in-service inspection and calibrations, radioactive waste handling and disposal, etc.

<u>Descriptions and examples of general design features to maintain doses ALARA during normal power and shutdown operations are provided in Subsection 12.3.1</u>

The features of the plant design that ensure the plant can be operated and maintained with ALARA exposures also serve to assist in achieving ALARA exposures during the decommissioning process.

Examples of features that assist in maintaining low occupational exposures during decommissioning include the following:

- Provisions for draining, flushing, and decontaminating equipment and piping.
- Design of equipment to minimize the buildup of radioactive material and to facilitate flushing of crud traps.
- Shielding which provides protection during maintenance or repairs and during decommissioning operations.
- Provision of means and adequate space for utilization of movable shielding.
- Separation of more highly radioactive equipment from less radioactive equipment and provision of separate shielded compartments for adjacent items of radioactive equipment.
- Provision for access hatches for the installation or removal of plant components.
- Provision of design features such as the Reactor Water Cleanup/Shutdown Cooling System and the condensate demineralizer to minimize crud buildup.

NRC RAI 12.2-17

DCD Tier 2, Section 12.1.2.3.1, third bullet, refers to "transportation of equipment or components requiring servicing to a lower radiation area." Provide examples of the application of this design feature.

GE Response

DCD Tier 2 Subsection 12.3.1.1.2 states that instruments that must be located in high radiation areas due to operational requirements are designed so that they can be removed to low radiation areas for maintenance.

Examples of equipment located in high radiation areas designed for transport to lower radiation areas for maintenance are: pumps and motor-operated valves electrical motors, and remote startup and control devices of equipment (pumps, motor-operated valves...)

In particular, the ESBWR design includes a dedicated room for maintenance of the Hydraulic Control Units (HCU). Specifically, Room 1107, at Reactor Building Elev. -11500, is designed for HCU maintenance and classified as Radiation Zone C at power operation (20 hr/wk controlled access), and as Radiation Zone B at shutdown (unlimited and controlled access), per DCD Figure 12.3-1. The HCUs normally reside in a higher radiation field (Rooms 1110, 1120, 1130, and 1140), in Radiation Zone D at power operation (4 hr/wk controlled access), and Radiation Zone C at shutdown.

Similarly, maintenance of the control rod drive (CRD) components occurs in Room 2200, which is classified as Radiation Zone B (unlimited and controlled access), while the CRDs themselves are in a high radiation environment (Radiation Zone F – controlled access & authorization required).

No DCD changes will be made in response to this RAI.

NRC RAI 12.2-18

DCD Tier 2, Section 12.1.2.3.2 refers to "central control panels [in the] lowest radiation zones practicable." and "When practicable for package units, separate highly radioactive components..." Provide examples of each.

GE Response

a) Examples of central control panels located in the lowest radiation zones practicable:

- The Remote Shutdown Control Panel in 1313 and 1323 rooms (Radiation Zones A).
- The Control Rod Drive Maintenance Control Panel in 2202 room (Radiation Zone B).
- The Control Rod Drive Panels in 1400, 1401, 1402 and 1403 rooms (Radiation Zone A).
- The Reactor Building Sample Panel, in 1107 room (Radiation Zone C).

b) Examples of separation of highly radioactive components:

- Remote startup and control devices of pumps and motor-operated valves, located in accessible zones:
  - In the TB, polishing valves are located apart in room 41F0 (Radiation Zone D).

- Sampling equipment and instruments located in low radiation zones, separate from the systems they sample:
  - In RB, the reactor building sample panel is in room 1106 (Radiation Zone B).
  - In TB, the turbine sampling panels are in room 4101 (Radiation Zone C).

- Placement of the instruments of radioactive systems (RWCU, FAPCS...) in zones, separate from the high radioactive equipment to which they correspond:
  - In RB, the RWCU/SDC valves are in rooms 1150 and 1160 (Radiation Zone D for both.)

No DCD changes will be made in response to this RAI.

NRC RAI 12.3-6

DCD Tier 2, Sections 12.2.1.1 and 12.2.1.2, state that the sources described for the containment and reactor building do not consider the deposition of corrosion or fission products in contained systems and components. Verify that the deposition of activated corrosion and wear products was considered in the ESBWR plant layout and shielding design. Justify why current operating experience does not provide an adequate basis for determining the nominal values for these expected sources, or provide a description of these sources.

GE Response:

Although deposition of activated corrosion and wear products mainly affects maintenance and decommissioning and has a minor effect on layout and shielding design, activated corrosion and wear products are considered in the ESBWR analysis as a part of the primary system inventory in water described in DCD Tier 2 Section 11.1 and Tables 11.1-1 through 11.1-17.

As indicated in DCD Tier2 Section 1.9 Table 1.9-22 (Industrial Codes and Standards Applicable to ESBWR), the ANSI/ANS-18.1-1999 Standard, "Radioactive Source Term for Normal Operation of Light Water Reactors", has been used for the ESBWR primary and fluid systems radioactivity estimation. ANSI/ANS-18.1-1999 incorporates the best available operating experience to date, providing the most appropriate basis for estimating the radioactivity in the principal fluid streams of a BWR. Note (b) to Table 8 of ANSI/ANS-18.1-1999 indicates that the effective removal terms considered mechanisms such as plate out.

Otherwise, the ESBWR design has been optimized to limit corrosion products and their deposition (see DCD Tier 2 Sections 12.1 and 12.3). Corrosion is prevented or limited by using high quality corrosion-strength materials. Components and pipes use materials with low Co and Ni content to avoid corrosion product generation. Equipment is suitably finished and protected with a coating or enamel to prevent corrosion product adherence to the components and aid in decontamination.

Pipes are seamless and slopes are adequate for avoiding stagnation. Low points, blind sections and the use of elbows are minimized. Straight passage valves are used whenever possible to reduce the amounts of corrosion product deposits in them. Tank bottoms with high radioactive solid content are curved or have conical sloped profiles with internal linings that prevent corrosion products from adhering to the walls.

Drains prevent the accumulation of corrosion products. Portions of contaminated systems that may require maintenance are cleaned by forced air or pressurized water.

In view of the design optimization to reduce corrosion product generation and deposition and the lack of operational experience with this type of plant, insufficient data is available at this time to provide an adequate basis for determining the corrosion product plate out sources. The utilization of ANSI/ANS-18.1-1999 to calculate the ESBWR nominal values for the expected sources provides a conservative approach to compensate for the lack of ESBWR operational experience.

No DCD changes will be made in response to this RAI.

NRC RAI 12.4-02

DCD Tier 2, Section 12.3.1.2.4, states that "piping containing radioactive fluids is routed through shielded pipe chases, shielded equipment cubicles, or embedded in concrete walls and floors [emphasis added]." In addition the statement that radioactive piping is embedded in concrete, is made several places in the DCD. Embedding radioactive piping in concrete is a good ALARA measure; however, it does not facilitate dismantlement of the system nor decommissioning of the facility, as required by 10 CFR 20.1406. Verify that the ESBWR design provides for the routing of radioactively contaminated piping through shielded pipe chases, in lieu of embedding in concrete, to the maximum extent practicable.

GE Response

The ESBWR design is such that piping containing radioactive fluids is routed through shielded pipe chases or shielded equipment cubicles wherever possible, in order to comply with the project ALARA objectives.

In some piping, feed-throughs with short sections, the piping may be embedded in concrete. Optimization by short sections with embedded piping to the extent practicable facilitates the dismantlement of the systems and the decommissioning of the facility, as required by 10 CFR 20.1406.

No DCD changes will be made in response to this RAI.

NRC RAI 12.4-10

It is difficult to determine the accessibility of several areas/rooms depicted on the plant layout drawings (DCD Tier 2, Figures 12.3-1 through 12.3-22). For example, are the HCU rooms (rooms 1110-1140) designed to be accessed during power or shutdown operations? For each room and area depicted on the layout drawings, indicate whether they are designed for continuous access, infrequent access or inaccessible during power and shutdown operations.

GE Response

DCD Tier 2 Figures 12.3-1 to 12.3-22 include the areas/rooms classification based on the expected radiation dose level during power and shutdown operations. Subsection 12.3.1.3 of DCD Tier 2 describes the access constraints to the different radiation zones:

Radiation Zone A:	Uncontrolled and unlimited access
Radiation Zone B:	Controlled and unlimited access
Radiation Zone C:	Controlled and limited access (20 hr/wk)
Radiation Zone D:	Controlled and limited access (4 hr/w)
Radiation Zone E:	Controlled and limited access (1 hr/wk)
Radiation Zones F and higher:	Inaccessible during power and shutdown operations. Limited and very limited controlled access with special authorization permit required.

DCD Rev 1 Figures 12.3-1 to 12.3-22 include only the description of current radiation zones dose rates; additional description of access time constraints will be included in the DCD Tier 2 Revision 3 of Figures 12.3-1 to 12.3-22.

The HCU rooms are designed to be accessible both at normal operation (classified as Zone D, controlled access limited to 4 hr/wk), and in shutdown (classified as Zone C, controlled access limited to 20 hr/wk).

NRC RAI 12.4-12

Indicate whether, and if so, how, the guidance in Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants," ANSI/ANS 6.4, "Nuclear Analysis and Design of Concrete Shielding for Nuclear Power Plants," and ANSI/ANS 6.4.2, "Radiation Shielding Materials," were followed in the design of the ESBWR radiation shielding. Specify any deviations from this guidance and describe the alternative criteria and methods applied to the shielding design.

GE Response

The ESBWR design complies with position C.3 of Regulatory Guide 1.69 in that the concrete shielding has been designed for steady-state radiation (supporting heat and missile loadings). The general calculation method and the data needed to calculate concrete shielding thickness delineated in the ANSI/ANS 6.4-1985 Standard, have been applied. The following are examples where the ANSI/ANS 6.4-1985 Standard have been applied to the ESBWR shielding design:

- Shielding calculations are performed using computer programs like QAD-CGGP, MCNP-X and others, which incorporate mass attenuation coefficients for different energy groups of gamma radiation, Table B-1 of ANSI/ANS-6.4, for the case of normal concrete with a density of 2.35 g/cm3.

- As an example of the use of the ANSI/ANS-6.4 standard in the ESBWR design, the basic calculation methodology makes broad use of the "point kernel" methods set out in section 6.2 of the standard. This is an optimum calculation method used in QAD-CGGP programs, since it describes the size and form of the radioactive source by way of a finite number of isotropic elements and calculates the distance from each one of them to the detector considered, taking into account the different material media it crosses. Other calculation methods similar to "point kernel" such as Ref. 31 of the standard, have been used in the validated computer codes employed to calculate the shielding of piping systems and cylindrical equipment.

- The accumulation factors in concrete used in the calculations follow the criteria recommended in section 7.3 of the standard. In some cases, however, due to code use requirements, the geometric progression method is used. As a general rule, the values used in the energy groups are not very small and the concrete thickness' are usually large (on the order of centimeters), resulting in limits indicated in Figure B-1 of the standard.

- The Monte Carlo method is employed in the design of the shield windows of the drywell penetrations and calculation of dose rates in the inspection platforms calculations by making use of the MCNP-X program .

ESBWR shielding materials utilized were used based upon those specified in ANSI/ANS 6.4-2-1985 "Specification for Radiation Shielding Materials".

The application of alternative criteria and methods to the shielding design is not considered necessary.

DCD Table 1.9-22 will be revised to incorporate the ANSI/ANS 6.4-2-1985 standard in Revision 3 to the Tier 2 DCD . See the attached markup.

Table 1.9-22 Industrial Codes and Standards[*] Applicable to ESBWR

Code or Standard Number	Year	Title
2.8	1992	Determining Design Basis Flooding at Power Reactor Sites
2.10-1979	1979	Guidelines for Retrieval, Review, Processing and Evaluation of Records Obtained from Seismic Instrumentation
2.11-1978	1978 (R 1989)	Guidelines for Evaluating Site-Related Geotechnical Parameters at Nuclear Power Sites
2.12-1978	1978	Guidelines for Combining Natural and External Man-Made Hazards at Power Reactor Sites
3.2-1994	1994 (R 1999)	Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants
4.5-1980	1980 (R 1988)	Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors
5.1	1994	Decay Heat Power in LWRs
6.4	1997 (R 2004)	Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants
6.4-2	1985 (R 2004)	Specification for Radiation Shielding Materials
10.4-1987	1987 (R 1998)	Guidelines for the Verification and Validation of Scientific and Engineering Computer Programs for the Nuclear Industry
18.1-1999	1999	Radioactive Source Term for Normal Operation of Light Water Reactors
52.1-1983	1983 (R 1988)	Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants
55.1-1992	1992 (R 2000)	Solid Radioactive Waste Processing System for Light-Water-Cooled Reactor Plants
55.4-1993	1993 (R 1999)	Gaseous Radioactive Waste Processing Systems for Light Water Reactor Plants
55.6-1993	1993 (R 1999)	Liquid Radioactive Waste Processing System for Light Water Reactor Plants
56.2-1984	1984 (R 1989)	Containment Isolation Provisions for Fluid Systems After a LOCA
56.3-1977	1977 (R 1987)	Overpressure Protection of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary
56.5-1979	1979 (R 1987)	PWR and BWR Containment Spray System Design Criteria
56.7-1978	1978 (R 1987)	Boiling Water Reactor Containment Ventilation Systems
56.8-2002	2002	Containment System Leakage Testing Requirements
56.10-1982	1982 (R 1987)	Subcompartment Pressure and Temperature Transient Analysis in Light Water Reactors
56.11-1988	1988	Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants

NRC RAI 12.4-21

DCD Tier 2, Section 12.3.3.1, second bullet, states that concentrations of radionuclides in air will be "kept below the limits of 10 CFR 20 during normal power operation." Revise this statement to indicate that they will be below the concentrations defined as an airborne area in 10 CFR 20, or state specifically which limits are referred to by this statement.

GE Response:

DCD Tier 2, Section 12.3.3.1, second bullet, first sentence will be changed to read as follows:

- The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance will be below the concentrations that define an airborne radioactive area in 10 CFR 20 during normal power operation.

Revision 3 to DCD Tier 2, Section 12.3.3.1 will incorporate the attached markup.

ventilating capabilities of the systems. This Subsection discusses the radiation control aspects of the HVAC systems.

12.3.3.1 Design Objectives

The following design objectives apply to all building ventilation systems:

- The systems shall be designed to make airborne radiation exposures to plant personnel and releases to the environment ALARA. To achieve this objective, the guidance provided in Regulatory Guide 8.8 shall be followed.
- The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance ~~shall~~ will be below the concentrations that define an airborne radioactive area in ~~kept below the limits of~~ 10 CFR 20 during normal power operation. This is accomplished by establishing in each area a reasonable compromise between specifications on potential airborne leakages in the area and HVAC flow through the area. Appendix 12.A to this chapter outlines the methodology by which such calculations are made.

The COL licensee will perform calculations for the expected airborne radionuclide concentrations to verify the adequacy of the ventilation system prior to fuel load. See Subsection 12.3.7.2 for COL license information.

12.3.3.2 Design Description

In the following Subsections, the design features of the various ventilation systems that achieve the radiation control design objectives are discussed. For all areas potentially having airborne radioactivity, the ventilation systems are designed such that during normal and maintenance operations, airflow between areas is always from an area of low potential contamination to an area of higher potential contamination.

12.3.3.2.1 Control Room Ventilation

The control building atmosphere is maintained at a slightly positive pressure (up to 0.5 in. Hg) at all times in order to prevent infiltration of contaminants. When offsite power is available, fresh air may be taken in via the single inlet system, which has its intake structure on the side of the building. During an isolation event if offsite or backup power is not available, bottled air can be supplied by a redundant supply system for up to 72 hours prior to requiring recharging. Under conditions when offsite or backup power is available, either bottled or filtered air may be used. The operator has manual control in the event filtered air is used to either run under filtered air or bottled air.

Outside air that enters the intake is normally filtered by a particulate filter. Under contamination conditions however, if external air is selected, the air flow is diverted through an adsorber system having:

- a particulate filter;
- a HEPA filter;
- a charcoal filter; and

NRC RAI 12.4-27

Figure 12.3-23 indicates two ARMs numbered 18, please clarify

GE Response

ARM #18, as indicated in DCD Tier 2 Table 12.3-2, is associated with the Reactor Building sump pumps (Room 1153). In DCD Tier 2, Figure 12.3-23, the duplicated ARM #18 located in room 1106 is an error that will be deleted.

DCD Figure 12.3-23 will be incorporated in Revision 3 , showing the elimination of Room 1106 ARM #18.

NRC RAI 12.5-5

Clarify the statement in the middle of page 12.4-7 (DCD Tier 2, Section 12.4.4) that "[t]he material of construction for the condenser tubesheet is titanium which reduces leakage of corrosion products into the feedwater." Verify that the ESBWR main condensers will have titanium heat exchanger tubes. The performance advantage of titanium tubes over bronze tubes minimizes the introduction of impurities from the ultimate heat sink (in the circulating system) into the feedwater system and ultimately into the reactor.

GE Response:

The condenser tube material (with compatible tubesheet material) will be corrosion resistant (titanium or stainless steel), which reduces leakage of corrosion products into the Condensate and Feedwater System. A complete commitment to the exact type of condenser material can not be made at this stage of the design, however, it has been decided that the condenser tube material will be either titanium or stainless steel, and not bronze.

The specific condenser tube materials will be specified in DCD Subsection 12.4.4 in Revision 3 to the Tier 2 DCD.

NRC RAI 12.7-3

Identify ESBWR piping or components that have a potential for leaking radioactively contaminated fluids, and are is designed to be below the grade (ground level) of the plant site. Describe design features intended to prevent acute, and minimize chronic (over the life of the plant), leakage from these systems and components. Describe how leakage from these systems and components will be detected and captured to minimize contamination of the soil and/or ground water below the site. This description should include, but not be limited to, the spent fuel pool, Radwaste Building tanks and sumps, radwaste piping and drain lines between the Radwaste Building and other plant buildings (i.e., the Reactor and Turbine Buildings), and radwaste discharge lines.

GE Response:

The ESBWR generic design is provided with the Equipment and Floor Drain System (EFDS) (DCD Subsection 9.3.3) to collect all possible liquid leaks. EFDS consists of the Clean (non-radioactive) Drain Subsystem and the following five potentially contaminated subsystems:

a. The Low Conductivity Waste (LCW) Drain Subsystem collects potentially radioactive liquid waste from equipment located inside the drywell, Reactor Building, Fuel Building, Turbine Building, and Radwaste Building.

b. The High Conductivity Waste (HCW) Drain Subsystem collects potentially radioactive liquid wastes from floor drains inside the containment, the Reactor Building, Fuel Building, Turbine Building, and Radwaste Building.

c. The Detergent Drain Subsystem collects potentially radioactive wastes from the personnel decontamination stations, and shower facility drains and transfers them to the detergent waste drain collection tank.

d. The Chemical Waste Drain Subsystem collects typically non-radioactive wastes, as well as potentially radioactive chemicals and corrosive substances from the wash down areas, laboratory drains, chemical decontamination solutions, and other miscellaneous sources in the Turbine Building and Radwaste Building. Effluents from analyzers, which add chemicals to the samples (e.g. sodium and silica analyzers), are also routed to the Chemical Waste Drain subsystem.

e. The Reactor Component Cooling Water System (RCCWS) Drain Subsystem: Equipment drains receives closed loop cooling water system water and directs it to the Reactor Building Cooling Water Drain Subsystem.

These five subsystems collect liquid drainage from various plant areas and transfer them to the Liquid Waste Management System (LWMS) (DCD Section 11.2) in the Radwaste Building.

The Condensate Purification System (CPS) (DCD Subsection 10.4.6), located in the Turbine Building, collects and transfers potentially radioactive wastes (water, spent resins and sludge) to the Liquid Waste Management System (LWMS) (DCD Section 11.2) and the Solid Waste Management System (SWMS) (DCD Section 11.4) in the Radwaste Building.

The Spent Fuel Pool, as well as the other pools in the Reactor Building and Fuel Building, are continuously cleaned by the Fuel and Auxiliary Pools Cooling System (FAPCS) (DCD Subsection 9.1.3). The Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System (DCD Subsection 5.4.8) continuously cleans the reactor coolant during normal operation. FAPCS and RWCU/SDC collect and transfer potentially radioactive wastes to the Liquid Waste Management System (LWMS) (DCD Section 11.2) and the Solid Waste Management System (SWMS) (DCD Section 11.4) in the Radwaste Building.

The transfer piping from buildings to the Radwaste Building is routed through the Radwaste Tunnel. This tunnel is designed to meet the same requirements as the Radwaste Building.

The ESBWR Radwaste Building, and its structures, systems and components (SSCs), are classified as safety class RW-IIa (high hazard) according to Regulatory Guide 1.143. The foundation and walls up to the spill height of the building protect the radwaste systems from the effects of failure of the housing structure.

DCD Subsection 15.3.16, Liquid Containing Tank Failure, states how the liquid pathway is not considered because of the mitigation capabilities of the Radwaste Building.

The Condensate Storage Tank is provided with a retaining basin (dike) designed to contain the total content of the tank and prevent uncontrolled runoff in the event of a tank failure (DCD Subsection 9.2.6).

DCD Subsection 11.2.3 provides the evaluation of radwaste system liquid releases. The radwaste discharge line activity concentration is consistent with the discharge criteria of 10 CFR 20 and dose commitment in 10 CFR 50, Appendix I.

No DCD changes will be made in response to this RAI.