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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 218 Related to ESBWR Design Certification Application – Radiation Protection – RAI Number 12.4-19 S03**

Enclosure 1 contains the GE Hitachi Nuclear Energy (GEH) response to the subject NRC RAI originally transmitted via the Reference 1 letter and supplemented by an NRC request for clarification and GEH responses addressed in References 2 through 5.

Enclosure 2 contains the DCD markup associated with the RAI response.

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

Sincerely,

Lee F. Dougherty for

Richard E. Kingston
Vice President, ESBWR Licensing

Reference:

1. MFN 08-561, *Letter from the U.S. Nuclear Regulatory Commission to Robert E. Brown, Request for Additional Information Letter No. 218, Related To ESBWR Design Certification Application*, dated July 1, 2008

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

3. MFN 06-499, *Response to Portion of NRC Request for Additional Information Letter No. 60 – Radiation Protection Systems – RAI Numbers 12.3-1, 12.4-14 and 12.4-19*, dated December 22, 2006

4. MFN 06-499 Supplement 1, *Response to RAI Letter Nos. 60 and 112 Related to the ESBWR Design Certification – Radiation Protection – RAI Numbers 12.4-4S02, 12.4-19S01 and 12.4-19S02*, dated April 15, 2008

5. MFN 07-558, *Letter from the U.S. Nuclear Regulatory Commission to Robert E. Brown, Request for Additional Information Letter No. 112, Related To ESBWR Design Certification Application*, dated October 17, 2007

Enclosures:

1. Response to Portion of NRC Request for Additional Information Letter No. 218, Related to ESBWR Design Certification Application – Radiation Protection – RAI Number 12.4-19 S03
2. Response to Portion of NRC Request for Additional Information Letter No. 218, Related to ESBWR Design Certification Application – Radiation Protection – RAI Number 12.4-19 S03 - DCD Markup

cc:	AE Cubbage	USNRC (with enclosures)
	RE Brown	GEH/Wilmington (with enclosures)
	DH Hinds	GEH/Wilmington (with enclosures)
	eDRF	0000-0095-4893 (Rev 2)

Enclosure 1

MFN 06-499 Supplement 2

Response to Portion of NRC Request for

Additional Information Letter No. 218

Related to ESBWR Design Certification Application

Radiation Protection

RAI Number 12.4-19 S03

NRC RAI 12.4-19 S03:

Provide additional clarifying information and correct inconsistencies and errors associated with the response to RAIs 12.4-19 S01 & S02.

Review of GEH responses to RAI 12.4-19 S01 & S02 12.4-19 S01

1. *In GEH response to RAI 12.4-19 S01 b), GEH stated that access to any area adjacent to the transfer tube (classified as being radiation zone B or above) will be controlled through a series of physical controls, interlocks and an annunciator. Specify which areas will be included under these controls (are these areas limited to Room 1702, the Trapezoidal room, and some of the rooms in the rail car bay (at elevations 4650 and 13570)?).*
2. *In GEH response to RAI 12.4-19 S01 b), specify how access will be restricted to the area near the operating platform of the fuel handling machine opposite the trapezoidal room at elevation +4650 (adjacent to the embedded IFTT).*
3. *Figure 12.3-10 is cross section A-A of the reactor building. Even though the wetwell access hatch on level 13570 is located at roughly the 1:00 o'clock position on Figure 12.3-6, discuss why this hatch appears on Figure 12.3-10 (which is a cross-section of the reactor building at the 3 o'clock position).*
4. *On the basis of GEH response to RAI 12.4-19 S01 c), it appears that the IFTT is embedded in the concrete structure to the south of the hallway on level 13570 (Room 1600). Justify why the IFTT is not shown on Figure 12.3-6. Justify why the hallway is not shown on Figure 9.1-2.*
5. *On the basis of GEH response to RAI 12.4-19 S01 c), the dose rate in the hallway on level 13570 during fuel transfer will be less than 10 μSv/hr (zone B). Explain why the Note near this hallway on Figure 12.3-6 (in Rev. 4) stated that the dose rate in this hallway during spent fuel transfer was less than 5 Sv/hr (zone I) (a factor of 5 E+5 higher).*
6. *In GEH response to RAI 12.4-19 S01 c), GEH stated that the maximum dose rate in the hallway on level 13570 from a single spent fuel assembly in the adjacent portion of the IFTT would be 3.04E-04 mSv/hr. What would be the maximum dose rate in this area from two spent fuel assemblies in the IFTT?*
7. *In GEH response to RAI 12.4-19 S01 e), GEH modified Figures 12.3-4 (not Figure 12.3-6 as stated in GEH response) and 12.3-10 to show the access plug entrance to the trapezoidal room. Describe the route that a person would take to reach this access plug entrance on elevation 4650. Modify Figure 12.3-7 (elevation 17500) to show the access plug entrance to Room 1702 (this access plug entrance is shown on Figure 12.3-10).*

12.4-19 S02

8. *In GEH response to RAI 12.4-19 S02 a), GEH provided a list of shielding assumptions for GEH new dose calculations that resulted in the revised dose rate estimates shown in Table 2 for various rooms surrounding the IFTT.*

   a) *It is unclear from this list which of these assumptions is new or has been revised in order to respond to RAI 12.4-19 S02. Provide a list of all assumptions used and specify those assumptions which have been added or revised to respond to RAI 12.4-19 S02.*

   b) *In calculating the dose rates shown in Table 2, GEH assumed two fuel assemblies in the IFTT, each with a burnup of 35 GWd/MTU. State what effect the transfer of two fuel assemblies, each having a higher burnup value (assume 58 GWd/MTU, as per the response to RAI 12.2-19 S01), would have on the dose rates provided in Table 2.*

   c) *Other than the fact that the dose rates shown in Table 2 are based on the presence of two, not one, spent fuel assemblies in the adjacent IFTT, describe, in general, how these dose rates compare with the calculated dose rates provided in response to the initial RAI 12.4-19.*

9. *In GEH list of assumptions for Table 2 (in GEH response to RAI 12.4-19 S02), GEH calculated the dose rates in the rooms adjacent to the IFTT assuming the presence of two fuel elements side-by-side in the IFTT. Describe the criteria that will be used for determining whether one or two spent fuel elements will be transferred in the IFTT and state what percentage of the time two spent fuel elements will be transferred in the IFTT and what percentage of the time only one spent fuel element will be transferred in the IFTT.*

10. *In GEH response to RAI 12.4-19 S02 b), Table 2 indicates that the concrete wall shielding thickness between the IFTT and the FB rail car bay at elevations 4650 and 13570 is 125 cm. Based on this thickness, GEH calculated that the dose rate at these two areas from two spent fuel assemblies in the adjacent portion of the IFTT would be 1.86 mSv/hr. Verify that the 125 cm. was the thickness of the concrete assumed in the original dose rate calculations referenced in the response to the initial RAI 12.4-19.*

11. *GEH's initial response to the initial RAI 12.4-19 referenced a dose rate of 3.6E+01 mSv/hr on elevation 4650 near the border of the pit 21P1. GEH response inferred that this dose rate was near an area where the IFTT was shielded with a minimum of 50 cm. of concrete. On the basis of the description, it appears that this area described on elevation 4650 (and having a dose rate of 3.6E+01 mSv/hr) is the same area described in the last row of Table 2 (in response to RAI 12.4-19 S02) with a dose rate of 1.86 mSv/hr. If this is the same area, describe why the GEH response to RAI 12.4-19 stated that the minimum shielding thickness was 50 cm. of concrete, resulting in a maximum dose rate of 3.6E+01 mSv/hr during IFTT operation and GEH response to RAI 1.4-19 S02 stated that the shielding thickness was 125 cm. of concrete, resulting in a maximum dose rate of 1.86 mSv/hr during IFTT operation. If the area described on elevation 4650 in the initial response to*

RAI 12.4-19 is not the same area as described in the last row of Table 2, state what the revised dose rate (based on GEH new shielding assumptions) would be in this area and identify the location of this area on Figure 12.3-4.

Revised Figures 12.3-1 to 12.3-10

12. *State why proposed (revised) Figures 12.3-1, 12.3-2, and 12.3-3 no longer include Notes 2, 4, and 3, respectively, relating to the existence of radiation zone J in the 21P1 pit when recently unloaded spent fuel is present.*
13. *On Figure 12.3-4, Note 2 states that the radiation level in the area between the trapezoidal room and the inclined fuel transfer tube pit increases to zone F during spent fuel transfer. Verify that this specified area is the same area as the location near the operating platform of the fuel handling machine opposite the trapezoidal room at elevation +4650 (adjacent to the embedded IFTT).*
14. *Figure 12.3-9 appears to include several modifications. Explain the addition of the circular ring (between coordinates RB-RC and R1-R3) and the two "wavy" lines (at coordinates RB and RF between R1 and R5).*
15. *Explain the rationale for deleting Note 3 (zone increase during spent fuel transfer) in area 17P3 and Note 4 (zone increase in the presence of the dryer/separator) in area 18P0.*
16. *Figure 12.3-4 (elevation 4650) shows that the radiation level at the fuel handling machine is Zone C (<5mRem/hr). A Zone C classification indicates that the area dose rates are between 1 mRem/hr (Zone B) and 5 mRem/hr (Zone C). Since this is an area where personnel will be present during refuelings, specify whether the expected dose rates will be closer to 1 mRem/hr or 5 mRem/hr.*

GEH Response:

1 Room 1702 and the trapezoidal room are permanently closed, except when maintenance is being carried out on the components (valves, seals) that service the tube, which will be done when there is no fuel being transferred and, consequently, with the appropriate controls referred to in the question. A system of key-locks in both the Inclined Fuel Transfer Tube (IFTT) main operation panel and in the control room is provided to prevent access to any IFTT maintenance area during IFTT operation.

Additional areas with controls only apply to the FB rail car bay (at elevations 4650, and 13570).

2 Access to Fuel Building Rail car bay (at elevations 4650) area adjacent to the transfer tube, with radiation zone classification "F", is controlled, in accordance with 10CFR 20.1601 and 10CFR 20.1902, through a system of physical controls (barriers, interlocks), and annunciators. Specifically:

- Chains, locked doors, annunciators and posted signals are located at areas close to IFTT rooms where high radiation dose is expected. An audible alarm

and flashing red lights are provided both inside and outside of these rooms, or areas, to indicate IFTT operation;

- Radiation monitors with alarms are provided both inside and outside any maintenance areas.

COL item 12.1-3-A states:

The COL applicant will provide the criteria and conditions under which various operating procedures and techniques will be implemented to ensure occupational radiation exposures ALARA are implemented using the guidance of NUREG-1736 (Reference 12.1-5) (Subsection 12.1.3), to the level of detail provided in Regulatory Guide 1.206 (Reference 12.1-6).

3. The location of the wetwell access hatch is on level 13570 at the 1:00 o'clock position (between Columns RA and RB and Lines R5 and R6). In general, the section drawings provided in the DCD are intended to provide the elevation and "location with respect to a given building's centerline" of various structures, which are judged to be significant to the design. It should be noted that DCD Tier 1 Figure 1.2-10 also depicts the wetwell access hatch in the same manner.

4. DCD Tier 2, Figure 12.3-6 will be revised to show the IFTT. DCD Tier 2, Figure 9.1-2 has been revised to show the hallway as shown in the Enclosure.

5. DCD Rev. 5 Figure 12.3-6 will be revised and the hallway is classified as Zone B. The Zone "B" classification for the hallway does not change during spent fuel transfer because of the concrete thicknesses between the hallway and the inclined fuel transfer tube.

6. The maximum dose rate in this area from two spent assemblies in the IFTT was reported in Table 2 of the response to RAI 12.4-19 S02 (MFN 06-499, Supplement 1 and 06-512, Supplement 2, dated April 15, 2008). It should be noted that there is a typographical error in that table; the value reads as 6.84E-04 mSv/hr, while the actual value is 6.48E-04 mSv/hr for two elements. This corresponds to 3.24E-04 mSv/hr for one element. The value provided (3.04E-04 mSv/hr) in the response to RAI 12.4-19 S01 (MFN 06-499, Supplement 1 and 06-512, Supplement 2, dated April 15, 2008) should be 3.24E-04 mSv/hr

7. The route a person would take to reach the access plug entrance is through the access to the Fuel Building from the Service Building at elevation 4650 near the fuel storage pool. DCD Tier 2, Figure 12.3-7 will be modified to show the access plug entrance to Room 1702.

8.a For the responses to RAI 12.4-19 S02 and RAI 12.4-19, the assumptions were:

<table>
<tr><th>Point</th><th>RESPONSE TO RAI 12.4-19 S01 & S02</th><th>RESPONSE TO RAI 12.4-19</th></tr>
<tr><td>01</td><td>The intensity of gamma radiation source for a GE-14 fuel element in an ESBWR reactor, with a burn-up of 35 GWd/MTU, and after one day of decay following shutdown, is indicated in the following Table 1.
Table 1
Spent Fuel Element Radiation Source Intensity
<table>
<tr><th>Energy, MeV</th><th>MeV/s</th></tr>
<tr><td>0.8</td><td>6.02E+16</td></tr>
<tr><td>1.3</td><td>3.99E+15</td></tr>
<tr><td>1.7</td><td>1.14E+16</td></tr>
<tr><td>2.5</td><td>1.19E+15</td></tr>
<tr><td>4.0</td><td>5.37E+12</td></tr>
<tr><td>5.0</td><td>4.08E+10</td></tr>
<tr><td>6.4</td><td>1.02E+10</td></tr>
</table></td><td>Same hypothesis</td></tr>
<tr><td>02</td><td>We have considered a parallelepiped shaped radiation source with dimensions 13.8 x 13.8 x 424.5 cm, representing the spent fuel element.</td><td>Same hypothesis</td></tr>
<tr><td>03</td><td>The density of the fuel element homogenized in water is 4.26 g/cm3, and the percent composition is: UO2-59.9%, Steel-1.9%, Zircalloy-2-25%, and water-13.2%.</td><td>Same hypothesis</td></tr>
<tr><td>04</td><td>The spent fuel element is located inside a steel tube with an outer radius of 30.48 cm (12"), representing the transfer tube.</td><td>The preliminary radius was 38.1 (15”).</td></tr>
<tr><td>05</td><td>The thickness of transfer tube is 1.27 cm (0.5").</td><td>Same hypothesis</td></tr>
<tr><td>06</td><td>The space between the fuel element and the transfer tube is considered to be full of water, with density 1 g/cm3.</td><td>Same hypothesis</td></tr>
<tr><td>07</td><td>In those rooms where the transfer tube is embedded in concrete, there is a guard tube with an outer radius of 45.72 cm (18").</td><td>The preliminary radius was 40.65 (16”).</td></tr>
<tr><td>08</td><td>The thickness of guard tube is 1.27 cm (0.5").</td><td>Same hypothesis</td></tr>
<tr><td>09</td><td>The space between the transfer tube and the guard tube is considered to be full of air.</td><td>Same hypothesis</td></tr>
<tr><td>10</td><td>The guard tube is not included in the model when the transfer tube passes through the interior of a room. In these cases, the air-filled space between the transfer tube and the concrete wall is considered to be 15.24 cm (6”).</td><td>Guard tube was considered in calculation.</td></tr>
</table>

11	The guard tube is not included when calculating the dose rate inside the rooms crossed by the transfer tube.	Doses in the room crossed by the transfer tube were not calculated.
12	A normal concrete cylinder with density 2.35 g/cm3 is considered to surround the guard tube. In this cylinder there are dose rate detection points at distances of 30 cm 50 cm, 100 cm, 150 cm and 200 cm, representing the different thicknesses of the shielding of the room next to the inclined tube.	Same hypothesis
13	Additional lead shielding of 10 cm or 15 cm is included in the model when the transfer tube passes through the inside of a room.	Lead shielding calculations were not considered
14	The dose rates were multiplied conservatively by 2 in order to take into account the possibility of the simultaneous presence of two fuel elements.	Only one element was assumed to be in the tube.
15	To account for a fuel burnup of 58 GWd/MTU instead of 35 GWd/MTU, a multiplier of 2.0 was conservatively applied to adjust the dose in order to account for the potential simultaneous presence of two fuel elements.	A fuel burnup of 58 GWd/MTU was not considered.
16	The calculation was carried out using the QAD-CGGP program, duly verified and incorporated into EA's QA system.	Same hypothesis

8.b. The results set out in table 2 accounted for the fuel burnup level (see assumption 15 of response 8.a above).

8.c Because there are significant differences with respect to the first response, no detailed comparison is performed but the differences in the results obtained are considered to be justified. The differences are:

- The identified incorrect input data to the QAD-CGGP shielding code were corrected and new calculations performed.
- The radius of the transfer tube has changed (see response to 8.a) from the preliminary values used to the verified input data.
- Under the present hypotheses (see response to 8.a), the results are multiplied by two to take into account the possibility of two fuel elements being in the transfer tube at the same time.
- Under the present hypotheses (see response to 8.a), the results obtained with two elements are multiplied by two to take into account a fuel burnup of 58 GWd/MTU instead of 35 GWd/MTU.
- In areas that require greater concrete thicknesses compared to what was indicated in the DCD General Arrangement drawings, we have calculated lead shielding with thicknesses equivalent to the required concrete thicknesses.

9. Two fuel elements have been considered because this is permitted in the design of the transfer system.

   For the shielding analysis, no consideration is given to the percentage of time one or two fuel elements will be transferred at the same time. GEH has conservatively considered that the two fuel elements will be transferred at the same time every time.

10. GEH confirms that the concrete thickness is 125 cm.

    Elevation 13570: The minimum thickness of the shielding between of the inclined fuel transfer tube and the fuel building is 50 cm; assuming that the thickness of the crane girder cantilever is around 75 cm, the total equivalent concrete shielding would be 125 cm. In the response to RAI 12.4-19, no consideration was given to the concrete crane girder width.

11. Elevation 4650: the wall between the "trapezoidal room" and room 2400 has been assumed to be designed up to a minimum 125 cm concrete shielding equivalent (e.g., 50 cm of concrete plus 15 cm of lead), thus maintaining room 2400 as a radiation Zone F or lower during IFTT operation. The radiation zone figures (12.3-4 and 12.3-10) will include notes indicating the additional lead shielding thickness considered. The additional lead shielding was not considered in the response to RAI 12.4-19 S02, which specified a thickness of 50 cm. This is the same area described in the last row of Table 2 from RAI 12.4-19 S02.

12. In pit 21P1, the fuel being transferred is the recently unloaded fuel. Therefore, in pit 21P1 (DCD Rev. 4), Note 1 of figures 12.3-1, 12.3-2 and 12.3-3, "during spent fuel transfer" is equivalent to the note "recently unloaded spent fuel" (Note 2 in Figure 12.3-1, Note 4 in Figure 12.3-2 and Note 3 in Figure 12.3-3) so only one note (note 1 in figures 12.3-1, 12.3-2 and 12.3- 3 "Zone J during spent fuel transfer") in pit 21P1 is necessary. DCD Rev. 5 was revised and included only one Note 1. Notes 2, 3 and 4 for pit 21P1 are unnecessary.

13. The area between the trapezoidal room and the inclined fuel transfer tube pit is the same area as the location near the operating platform of the fuel handling machine opposite the trapezoidal room at elevation +4650 (adjacent to the embedded IFTT).

    The second part of note 2 on figure 12.3-4 states, "…is zone F at the area between trapezoidal room and the inclined fuel transfer tube pit, during spent fuel transfer". It refers only to that precise zone of area 2400, located alongside the trapezoidal area. See figure 12.3-4. This zone is a hot spot during spent fuel transfer inside the inclined tube. This is also indicated in the second part of Note 4 on figure 12.3-10 section A-A.

14. The circular ring corresponds to the storage drywell head, which is stored in this position when the reactor vessel is opened. The DCD Rev. 5 Figure 12.3-9 did not include a depiction of this equipment item because it is not normally in this position.

The wavy lines mark the edge of the structure supporting the beams of the crane in the reactor building. The DCD Rev. 5 Figure 12.3-9 did not include a depiction of this support structure. DCD Figure 12.3-9 will be updated to include the structure in DCD Rev. 6.

15. The note is unnecessary because the areas are between elevations 27000 and 34000. The note and the classification of the areas are included in Figure 12.3-8. Rooms 17P3 and 18P0 in Figure 12.3-9 are also shown in Figure 12.3-8 and have been qualified therein. Notes 3 and 4 of Figure 12.3-9 correspond to Notes 1 and 2 respectively in Figure 12.3-8. In the new DCD Rev. 5 Figure 12.3-9, Notes 3 and 4 have been deleted so as not to be repetitive. For greater clarity, please see Figure 12.3-10 (Section A-A).

16. A Zone C classification denotes that the radiation level will not exceed 5 mRem/hr. While that indicates the dose rate is between 1 mRem/hr (Zone B) and 5 mRem/hr, the dose rate may fall below 1 mRem/hr on occasion. It can not be determined whether the expected dose rates will be closer to 1 mRem/hr or 5 mRem/hr; the received doses to personnel will be limited to the lowest extent possible in accordance with ALARA procedures.

The Chapter 12 figures modified for this RAI response (Figures 12.3-4, 12.3-6, 12.3-7, 12.3-9 and 12.3-10 modified for sub questions 4, 7, 11 and 14) are created by adding the requested radiologically significant features to the corresponding DCD general arrangement figures in DCD Chapter 2 Tier 2 Chapter 1. Currently, those controlling (or governing) figures are being updated for reasons not related to DCD Chapter 12. For that reason, the updated Chapter 12 figures (Figures 12.3-4, 12.3-6, 12.3-7, 12.3-9 and 12.3-10) have not been included in this response, but the aforementioned changes have been added. This response will be revised to include the updated figures, and will be transmitted to the NRC prior to the issuance of DCD Rev. 6.

DCD Impact:

DCD Tier 2 Figure 9.1-2 will be revised for DCD Rev. 6, as shown in Enclosure 2.

Figures 12.3-4, 12.3-6, 12.3-7, 12.3-9 and 12.3-10 will be revised in DCD Rev. 6 and transmitted as described in the response.

Enclosure 2

MFN 06-499 Supplement 2

Response to Portion of NRC Request for

Additional Information Letter No. 218

Related to ESBWR Design Certification Application

Radiation Protection

RAI Number 12.4-19 S03
DCD Markup