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Subject: **Revised Response to Portion of NRC Request for Additional Information Letter No. 375 Related to ESBWR Design Certification Application – RAI Number 12.3-14**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) revised response to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) 12.3-14 sent by NRC Letter No. 375, Reference 1. GEH's original response was transmitted via Reference 2. This revision provides additional DCD markups requested by the NRC staff. No other changes were made, although the entire GEH response is provided for completeness.

GEH response to RAI Number 12.3-14 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,

Richard E. Kingston

Richard E. Kingston
Vice President, ESBWR Licensing

References:

1. MFN 09-624, Letter from U.S. Nuclear Regulatory Commission to Jerald G. Head, *Request for Additional Information Letter No. 375 Related to ESBWR Design Certification Application*, October 1, 2009
2. MFN 09-713, Submittal of Response to Portion of NRC Request for Additional Information Letter No. 375 Related to ESBWR Design Certification - RAI Number 12.3-14, December 9, 2009

Enclosures:

1. Revised Response to Portion of NRC Request for Additional Information Letter No. 375 Related to ESBWR Design Certification Application - Auxiliary Systems - RAI Number 12.3-14
2. Revised Response to Portion of NRC Request for Additional Information Letter No. 375 Related to ESBWR Design Certification Application - Auxiliary Systems - RAI Number 12.3-14 – DCD Markups

cc:	AE Cubbage	USNRC (with enclosures)
	JG Head	GEH/Wilmington (with enclosures)
	DH Hinds	GEH/Wilmington (with enclosures)
	TL Enfinger	GEH/Wilmington (with enclosures)
	eDRF Section	0000-0110-6447 (original response)
		0000-0110-6447, Rev. 1 (revised DCD Markups)

Enclosure 1

MFN 09-713, Rev. 1

Revised Response to Portion of NRC Request for Additional Information Letter No. 375 Related to ESBWR Design Certification Application

Auxiliary Systems

RAI Number 12.3-14

NRC RAI 12.3-14

Regulatory Guide 1.206 (Section C.I.12.3.2) states that the applicant should provide information regarding the shielding for each radiation source identified in Chapter 11 and Section 12.2, including the assumptions, codes, and techniques used in the shielding calculations. Section 12.3.2.2.2 of the ESBWR DCD contains a description of several of the shielding codes (QADF, GGG, DORT) used by GEH in performing their shielding design. In addition, Table 12.3-1 lists these shielding codes and others identified in other sections of the DCD.

During the recent staff review of several shielding calculational packages provided to the staff by GEH in preparation for the staff's September 14-18, 2009 QA inspection at GEH offices, the staff noted that GEH's contractor, Empresarios Agrupados (E.A.), had utilized the following shielding codes, EMIR (2.0) and PANDORA (4.0) to perform the shielding calculations to determine the dose rates from the radwaste tanks and other components that are part of the Solid Waste Management System (SWMS). In addition, one of the shielding packages reviewed makes reference to two other codes, NISEIS and MCNP, as being referenced in the supplementary documents listed as part of the shielding calculational package. Even though these codes were apparently used by GEH in the determination of dose rates (and subsequent radiation zone designations) from SWMS plant components, no description of any of these codes is mentioned anywhere in Chapter 12 of the ESBWR DCD.

It is the staff's position that if GEH used these codes as part of the shielding analysis for radiation sources identified in Chapters 11 and 12 of the ESBWR DCD, then GEH should modify Section 12.3.2.2.2 of the ESBWR DCD to:

a. include reference to the use of these shielding codes,

b. include a description of the function of each of these codes, and

c. provide a general listing of what plant systems/components were modeled using each of these shielding codes.

The staff noted that Section 12.3.2.2.2 of the DCD currently describes the use of several codes (QADF, GGG, and DORT) which have been approved by the NRC and which are available through the Radiation Safety Information Computational Center (RSICC). Based on the staff's review of the above mentioned E.A. shield calculational packages, several of the codes mentioned in these packages appear to be similar in function to the shield codes referenced in the DCD.

d. Provide a comparison of the capabilities of the E.A. identified codes with the shielding codes currently referenced in the DCD.

e. Provide a justification in the DCD of why the above mentioned shielding codes in the E.A. calculational packages represent acceptable alternative shielding codes to the comparable codes which have been approved by the NRC and which are available through the RSICC.

f. Include a listing of these E.A. referenced shielding codes in DCD Table 12.3-1.

<u>GEH Response</u>

<u>Point *a* of the RAI:</u>

Included in the ***"CODE DESCRIPTION"*** section below is a reference to the use of the shielding codes used in the ESBWR calculations.

<u>Points *b*, *c*, *d*, and *e* of the RAI:</u>

The computer codes PANDORA, QAD-CGGP, and MCNPX have been used in the shielding calculations for the ESBWR. The codes ORIGEN, EMIR, and NISEIS (the latter specific for N-16) have been used to calculate the source strength, based on the activity inventory.

To respond to the NRC requirements, all the codes cited are described in the ***"CODE DESCRIPTION"*** section, including a comparison with the codes QAD, GGG, and DORT - approved by the NRC and available through the RSICC - to justify that they represent an acceptable alternative to them.

The functions of each shielding code are described in the ***"CODE DESCRIPTION"*** section (point *b* of the RAI), along with an indication what systems/components were modeled using each one (point *c* of the RAI). Also provided in the ***"CODE DESCRIPTION"*** section is a comparison of the capabilities of the codes (PANDORA and MCNPX) used in the ESBWR (point *d* of the RAI), and a justification of why they are acceptable alternative shielding codes to the comparable codes (QAD, GGG, and DORT) approved by the NRC (point *e* of the RAI).

<u>Point *f* of the RAI:</u>

A listing of these referenced codes (point *f* of the RAI) is included in DCD Table 12.3-1.

CODE DESCRIPTION

- <u>NISEIS code</u>

  - <u>Functions</u>. NISEIS calculates the N-16 activity and strength along the length of its run through a piping circuit or a circuit of pipes and tanks, simulated by a series of nodes and branches.

    The program calculates the transit time through each pipe section or tank (branch) when it is not given as an input datum. It uses this datum to calculate the specific source strength (Mev/s.g) in each node and the average specific source strength for each branch. It also calculates the activity (Ci) in each branch based on the source strength. The program output is an input datum to the N-16 radiation shielding calculation programs.

- Source. The program, written in FORTRAN, was developed by EA for execution on any IBM-PC or other compatible computer.

- Use. The program was used to determine the N-16 radiation sources (activity and source strength) in the RWCU/SDC system in normal operation.

- ORIGEN code

  - Functions. ORIGEN calculates the evolution of the radionuclides present in the spent fuel elements of nuclear reactors, combining irradiation or burn stages with other decay stages. It calculates the production and disintegration of isotopes using the "exponential matrix" method. It uses libraries (from NEA) of cross sections, fission performances, and radioactive decay contained in different files.

  - Source. The program, written in FORTRAN, was developed by the Oak Ridge National Laboratory (ORNL) and is available through the RSICC.

  - Use. The program was used to determine the inventory of fission products in spent fuel elements from the ESBWR.

- EMIR code

  EMIR is a source strength calculation code. It is a simple tool for calculating the source strength from the activity of the isotopes and organizing the results into different energy groups. It is a preprocessor for the preparation of input data to shielding calculation codes such as PANDORA and QAD-CGGP.

  - Functions. The EMIR program calculates the intensity of the emission of gamma rays from one or more radionuclides (in MeV/s) discretized to specific energy values that the user can select at will (energy groups). The calculation is performed based on the activity of each radionuclide, provided by the user, and on the values of energy and abundance of corresponding gamma rays stored in data file GAMDAT (for 1313 radionuclides).

  - Source. The program, written in FORTRAN, was developed by EA for execution on any IBM-PC or other compatible computer.

  - Use. The program was used to prepare input data to shielding calculation codes such as PANDORA and QAD-CGGP.

- PANDORA code

  - Functions. PANDORA is a dose/shielding calculation code. It uses the point-kernel method to calculate the doses of gamma radiation produced by multiple cylindrical sources of determinate sizes and intensities through one or several layers of materials normally used for shielding (concrete, steel, water, etc). Used to determine gamma radiation shield wall thicknesses, it is applied when there is no radiation scattering, which is most frequently the case and takes place in some labyrinths and shield penetrations.

    The “point-kernel” method is generally applied in NPPs for shielding design. “It may be used in the design of shielding for equipment that contains gamma emitting fluids such as demineralizers, heat exchangers, filters, pipes, tanks and steam lines” (ANSI/ANS 6.4-2006).

  - Capabilities. In a single execution, the code can process up to 30 different radioactive sources calculating doses at different points (maximum of 10) through different shield configurations (maximum of 6) formed by several layers (maximum of 5) of different materials. Its speed and power makes it especially useful for calculating gamma radiation doses from multiple pipes and cylindrical components, or with a geometry easily assimilable to a cylinder, a very common scenario in nuclear power plant rooms.

  - Source. The program, written in FORTRAN, was developed by EA for execution on any IBM-PC or other compatible computer.

  - Comparison. PANDORA is especially useful for determining the thicknesses of walls to shield against gamma radiation from multiple pipes and components. QAD (a point-kernel code) only handles one radiation source in each run, making it unsuitable for the combined handling of multiple pipes and components normally found in an NPP room, a situation which PANDORA easily solves.

    GGG and DORT are especially recommended for solving specific radiation transport problems when there is radiation scattering (in labyrinths and shield penetrations) and neutron flux. These problems are more complex than those that have to be solved in the calculation of the majority of plant shield wall and floor thicknesses where these situations do not arise and which PANDORA can easily solve. GGG (a point-kernel code) is useful for calculating gamma ray scattering, and DORT - which uses the “discrete ordinates” method - is more useful for primary shield analysis because it readily processes coupled neutron and gamma ray attenuation.

Based on the foregoing, PANDORA is an acceptable alternative to QAD, GGG and DORT for calculating shield walls and floors when there is no neutron flux or radiation scattering. It is especially useful because of its simplicity, power and speed when it comes to calculating shield wall and floor thicknesses in areas housing numerous radioactive system pipes and components.

- Use. The code was used for the shield wall design of the main components of the RWCU/SDC system (into rooms: 1151, 1161, 1251,1252, 1261, and 1262) and FAPC system (into rooms: 2251, 2261, 2102, 2150, and 2160).

• MCNPX code

- Functions. MCNPX (Monte Carlo N-Particle Transport Code System for Multiparticle and High Energy Applications) is an internationally recognized code for analyzing the transport of neutrons and gamma rays by the Monte Carlo method.

  The Monte Carlo method is used for the treatment of complex radiation transport problems in complex geometries; e.g. labyrinths, ducts, piping penetrations, and others that involve radiation scattering such as neutron streaming and skyshine.

- Capabilities. The code deals with transport of neutrons, gamma rays, and coupled transport; i.e. transport of secondary gamma rays resulting from neutron interactions. It can also treat the transport of electrons - both primary and secondary source electrons - created in gamma-ray interactions.

- Source. This code has been developed and maintained by Los Alamos National Laboratory. The code is available through the RSICC.

- Comparison. According to the ANSI/ANS 6.4-2006, the MC method is more sophisticated than the point-kernel and ordinate discrete methods, so the MCNPX code is an acceptable alternative to the GGG and DORT codes.

- Use. The program was used to evaluate the shield door thickness in the RPV sacrificial wall (room 1570) and also for the dose rate calculation of the main steam line penetrations in the grating (el +21000) around the RPV during refuelling in the event of a dropped fuel bundle, in response to RAI 12.2-19 S02.

DCD Impact

DCD Tier 2 Section 12.3 will be revised as noted in the attached markup.

Note: DCD markup changes from the initial response (MFN 09-713) to this response (MFN 09-713, Rev. 1), are shown within boxes.

Enclosure 2

MFN 09-713, Rev. 1

Revised Response to Portion of NRC Request for Additional Information Letter No. 375 Related to ESBWR Design Certification Application

Auxiliary Systems

RAI Number 12.3-14

DCD Markups

10 CFR 50, Appendix A, GDC 19. The analyses of the doses to MCR personnel for the design basis accidents are included in Chapter 15.

- The dose at the site boundary as a result of direct and scattered radiation from the turbine and associated equipment is considered.
- In selected situations, provisions are made for shielding major radiation sources during inservice inspection to reduce exposure to inspection personnel. For example, steel platforms are provided for inservice inspection (ISI) of the RPV nozzle welds and associated piping.
- The primary material used for shielding is concrete at a density of 2.35 gm/cm^3. Concrete used for shielding purposes is designed in accordance with Regulatory Guide 1.69 (Reference 12.3-12). Where special circumstances dictate, steel, lead, water, lead-loaded silicone foam, or a boron-laced refractory material is used.

12.3.2.2.2 Method of Shielding Design

The radiation shield wall thicknesses are determined using basic shielding data and proven shielding codes. A list of the computer programs used is contained in Table 12.3-1. The shielding design methods used also rely on basic radiation transport equations contained in Reference 12.3-1. The sources for basic shielding data, such as cross sections, buildup factors, and radioisotope decay information, are listed in References 12.3-2 through 12.3-10.

The shielding design is based on the plant operating at maximum design power with the release of fission products resulting in a source of noble gas after a 30-minute decay period, and the corresponding activation and corrosion product concentrations in the reactor water listed in Section 11.1. Radiation sources in various pieces of plant equipment are cited in Section 12.2. Shutdown conditions, such as fuel transfer operation, as well as accident conditions, such as a Loss-of-Coolant Accident (LOCA) or a Fuel Handling Accident, have also been considered in designing shielding for the plant.

The mathematical models used to represent a radiation source and associated equipment and shielding are established to ensure conservative calculation results. Depending on the versatility of the applicable computer program, various degrees of complexity for the actual physical situation are incorporated. In general, cylindrically-shaped equipment such as tanks, heat exchangers, and demineralizers are mathematically modeled as truncated cylinders. Equipment internals are sectional and homogenized to incorporate density variations, where applicable. For example, the tube bundle section of a heat exchanger exhibits a higher density than the tube bundle clearance circle, due to the tube density, and this variation is accounted for in the model. Complex piping runs are conservatively modeled as a series of point sources spaced along the piping run. Equipment containing sources in a parallelepiped configuration, such as fuel assemblies and fuel racks, are modeled as parallelepiped with a suitable homogenization of materials contained in the equipment. The shielding for these sources is also modeled on a conservative basis ~~with~~accounting for discontinuities in the shielding, such as penetrations, doors, and partial walls~~ accounted for~~. The dimension of the floor decking is not considered in the shielding calculation as it is part of the effective shield thickness provided by the floor slab.

Direct ~~P~~pure gamma dose rate calculations~~, both scattered and direct~~, are conducted using point kernel codes (PANDORA, QAD~~F/GGG~~ and QAD-CGGP). The source terms are divided into groups as a function of photon energy, and each group is treated independently of the others. Credit is taken for attenuation through all phases of material, and buildup is accounted for using a third-order polynomial buildup factor equation. The more conservative material buildup coefficients are selected for laminated shield configuration to ensure conservative results.

PANDORA is especially useful for determining shield wall thicknesses when the radiation sources are pipes and cylindrical equipment. The program can handle multiple radiation sources in one run when there is no neutron flux or radiation scattering. The QAD codes are especially appropriate for equipment geometries and more complex shields which are modeled with the combinatorial geometry routines, but they only handle one source in each run and this makes this unsuitable for the combined handling of multiple pipes and equipment normally found in an NPP room, a situation which PANDORA easily solves. PANDORA has been used for the shield wall thickness design of the main components of the RWCU/SDC System and FAPCS System.

QAD-CGGP is a more updated version of QAD. It includes two additional utilities: double precision "Combinatorial Geometry" (CG) routines and a "Geometric Progression" (GP) fitting function for gamma ray buildup factor accumulation; therefore, QAD-CGGP is an acceptable alternative for QAD. When QAD codes have to handle several radiation sources, a run must be executed for each source. QAD-CGGP has been used for the shield wall thickness design of the main components of the LWMS and SWMS and also for the shield wall thickness design of the Inclined Fuel Transfer Tube.

Where shielded entries to high-radiation areas such as labyrinths are required, a gamma ray scattering point kernel code (GGG) is used to confirm the adequacy of the labyrinth design. The labyrinths are designed to reduce the scattered as well as the direct contribution to the aggregate dose rate ~~outside the entry~~, such that the radiation zone designated for the area is not violated.

PANDORA is an acceptable alternative to QAD, GGG and DORT for calculating shield walls and floors when there is no neutron flux or radiation scattering. It is especially useful because of its simplicity, power and speed when it comes to calculating shield wall and floor thicknesses in areas housing numerous radioactive system pipes and components.

For combined gamma and neutron shielding situations, discrete ordinate~~s~~ techniques (DORT) and Monte Carlo technique (MCNPX) are applied. According to ANSI/ANS 6.4-2006 the "Monte Carlo" method is more sophisticated than the "point kernel" and "ordinate discrete" methods; therefore, MCNPX is an acceptable alternative for the GGG and DORT codes.

The shielding thicknesses are selected to reduce the aggregate dose rate from significant radiation sources in surrounding areas to values below the upper limit of the radiation zone specified in the zone maps in Subsection 12.3.1.3. By maintaining dose rates in these areas at less than the upper limit values specified in the zone maps, sufficient access to the plant areas is allowed for maintenance and operational requirements.

~~Where shielded entries to high-radiation areas such as labyrinths are required, a gamma ray scattering code (GGG) is used to confirm the adequacy of the labyrinth design. The labyrinths are designed to reduce the scattered as well as the direct contribution to the aggregate dose rate outside the entry, such that the radiation zone designated for the area is not violated.~~

The codes ORIGEN, EMIR, and NISEIS (specific for N-16) have been used as preprocessors for the preparation of input data (source strength) to the shielding calculation codes such as PANDORA and QAD-CGGP.

ORIGEN has been used to determine the fission product inventory and the source strength in spent fuel elements.

NISEIS has been used to determine the N-16 radiation sources (activity and source strength) in the RWCU/SDC system in normal operation.

~~EMIR has been used to prepare the input data (source strength) of the shielding calculation for use with PANDORA and QAD-CGGP.~~

12.3.2.2.3 Plant Shielding Description

Plant shielding geometry associated with major sources is summarized in Table 12.3-8. The general description of the shielding is provided below:

Containment - The major shielding structures located in the drywell area consist of the reactor shield wall and the drywell wall. The reactor shield wall in general consists of 16 cm (6.3 in) of steel plate. The primary function served by the reactor shield wall is the reduction of radiation levels in the drywell due to the reactor to equipment such that service life is not limited. In addition, the reactor shield wall reduces gamma heating effects on the drywell wall, as well as providing for low radiation levels in the drywell during reactor shutdown. The drywell is an F radiation zone during full power reactor operation and is not accessible during this period.

The containment (drywell) outside wall is a 2 m (6.6 ft) thick reinforced concrete cylinder that totally surrounds the drywell. A 2.4 m (7.9 ft) thick reinforced concrete containment top slab covers the drywell. The drywell wall attenuates radiation from the reactor and other radiation sources in the drywell to allow occupancy of the Reactor Building during full power reactor operation.

The ESBWR plant includes all necessary shielding provisions in the upper drywell in order to reduce the dose ALARA during transfer of irradiated spent fuel assemblies. The ESBWR plant includes all applicable shielding design provisions to minimize dose rates in case of a fuel handling mishap resulting in dropping a fuel assembly across the reactor flange.

Reactor Building - In general, the shielding for the RB is designed to maintain open areas at dose rates less than 6 µSv/hr (0.6 mrem/hr).

Penetrations of the containment wall are shielded to reduce radiation streaming. Localized dose rates outside these penetrations are limited to less than 50 µSv/hr (5 mrem/hr). The penetrations through interior shield walls of the Reactor Building are shielded using a lead-loaded silicone sleeve to reduce the radiation streaming. Penetrations are also located so as to minimize the consequences of radiation streaming into surrounding areas.

The components of the RWCU/SDC System are located in the RB. Both the RWCU/SDC regenerative and nonregenerative heat exchangers are located in shielded cubicles separated from the other components of the system. Neither cubicle needs to be entered for system operation.

Process piping between the heat exchangers and the demineralizers is routed through shielded areas or embedded in concrete to reduce the dose rate in surrounding areas. The RWCU/SDC

Table 12.3-1

Computer Programs Used in Shielding Design Calculations

Computer Code	Description
QADF	A multi-group, multi-region, point kernel gamma radiation code for calculating the flux and dose rate at discrete locations within a complex source geometry configuration.
GGG	A multi-group, multi-region, point kernel code for calculating the contributions due to gamma ray scattering in a heterogeneous three-dimensional space.
DORT	A discrete ordinates two-dimensional transport code. Multi-group, multi-region neutron or gamma transport.
QAD CGGP 1.0	"Quick and Dirty Combinatorial Geometry –Geometric Progression". A multi-group, multi-region, point kernel gamma radiation code for calculating the flux and dose rate at discrete locations within a complex source geometry configuration
SKYIII-PC	A Monte Carlo skyshine code designed to aid in the evaluation of the effects of structure geometry on the gamma-ray dose rate at given detector positions outside of a building housing N-16 gamma-ray sources.
PANDORA	The code uses the Point-Kernel method to calculate the gamma radiation doses produced by multiple cylindrical sources of determinate sizes and intensities through a shielding of one or several layers of materials normally used as shielding (e.g., concrete, steel, water). The code is applicable when there is no radiation scattering or it is not significant.
MCNPX	The MCNPX Code (Monte Carlo N-Particle Transport Code System for Multiparticle and High Energy Applications) is an internationally recognized code for analyzing the transport of neutrons and gamma rays by the Monte Carlo method. This method is used for treating complex radiation transport problems in complex geometries, e.g. labyrinths, ducts, piping penetrations, and others that involve radiation scattering such as neutron streaming and skyshine.
ORIGEN	ORIGEN calculates the production and disintegration of radioactive isotopes present in the spent fuel element materials of nuclear reactors, combining irradiation or burn stages with other decay stages. In the calculations, it uses the "exponential matrix" method and libraries of efficient sections, fission performances and radioactive decay contained in different files. The code can also express the activity of the radionuclides (Ci) as intensity of emission (MeV/s) in several fixed energy groups which the user can not change.
NISEIS	NISEIS calculates the N-16 source activity and strength along the length of its run through a piping circuit or a circuit of pipes and tanks.

Computer Code	Description
EMIR	EMIR is a preprocessor that only calculates the intensity of the emission of gamma rays from one or more radionuclides (in MeV/s) discretized to specific energy values that the user can select at will (energy groups). The calculation is performed based on the activity of each radionuclide, provided by the user, and on the values of energy and abundance of corresponding gamma rays stored in data file ~~GAMDAT (~~for 1313 radionuclides~~)~~ The ORIGEN code has also the capability to perform this calculation. However, the ORIGEN code has additional capabilities, as described above.