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Subject: Response to the NRC Request for Additional Information on the RAPTOR-M3G and FERRET Topical Report

References:

1. Westinghouse letter LTR-NRC-17-7, "Submittal of WCAP-18124-NP, Revision 0, 'Fluence Determination with RAPTOR-M3G and FERRET,' " January 25, 2017 (ML17030A377)
2. NRC letter to Westinghouse, "Request for Additional Information Re: Westinghouse Electric Company WCAP-18124-NP, Revision 0, 'Fluence Determination with Raptor-M3G and Ferret' Topical Report (CAC No. MF9141)" (ML17290A147)

Via Reference 1, WCAP-18124-NP, Revision 0 was submitted for NRC review and approval. Reference 2 requested additional information in support of that review. Please find attached our responses to that request.

This submittal does not contain proprietary information of Westinghouse Electric Company LLC.

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Attachment 1 – LTR-REA-18-2

cc:
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Attachment 1

Response to the NRC Request for Additional Information on the RAPTOR-M3G and FERRET Topical Report

NRC Request #1

Section 2.5, "Discrete Ordinates Transport Calculations with RAPTOR-M3G," of WCAP-18124-NP, Revision 0, Westinghouse Electric Company (Westinghouse) topical report (TR) states: "RAPTOR-M3G calculations are performed with an S_8 (or higher) level-symmetric angular quadrature set."

RG 1.190, Section 1.3.1, "Discrete Ordinates Transport Calculation," states: "The adequacy of the spatial mesh and angular quadrature, as well as the convergence criterion, must be demonstrated by tightening the numerics until the resulting changes are negligible (Reference 32). In discrete ordinates codes, the spatial mesh and the angular quadrature should be refined simultaneously."

Regulatory Position 1.3.5, "Cavity Calculations," states: "In discrete ordinates transport calculations, the adequacy of the S_8 angular quadrature used in cavity transport calculations must be demonstrated." Furthermore, Section 1.3.5 of RG 1.190 states: "However, when off-midplane locations are analyzed, the adequacy of the S_8 quadrature to determine the streaming component must be demonstrated with higher-order S_n calculations."

Based on the above, please explain how the adequacy of the S_8 angular quadrature (and associated spatial mesh) has been demonstrated for the entire spatial domain for which RAPTOR-M3G is applicable. Also, please define the standard method for determining when greater than S_8 angular quadrature is needed.

Westinghouse Response to NRC Request #1

RAPTOR-M3G models typically contain 150 to 250 radial intervals, 80 to 150 azimuthal intervals, and 100 to 200 axial intervals. In the case of the 4-loop reactor model used as the basis for the analytic uncertainty analysis, the geometric mesh description consisted of 209 radial intervals, 195 azimuthal intervals, and 179 vertical intervals. The problem was originally evaluated using an S_8 level-symmetric angular quadrature set and a P_3 Legendre expansion of anisotropic scattering matrices.

This problem was re-meshed with the same geometry inputs, but with a 33% reduction in default interval size, to obtain a mesh grid with 311 radial intervals, 288 azimuthal intervals, and 265 vertical intervals. The refined problem geometry was analyzed with a S_{12} level-symmetric angular quadrature set and a P_5 Legendre expansion of the anisotropic scattering matrices. Computed values of fast neutron ($E > 1.0$ MeV) fluence rate were compared to values from the reference case.

The observed difference in the beltline region of the reactor vessel was 0.3%. Observed differences at dosimetry locations in the reactor cavity opposite the top and bottom of the core were 1.6% or less.

The 13% net uncertainty attributed to fluence values at pressure vessel beltline locations in Section 4.5 of WCAP-18124-NP includes a 5% uncertainty component denoted as "Other Factors". Uncertainties that result from discretization choices (spatial and angular) are considered to be encompassed by this category.

The standard method for establishing the adequacy of the spatial mesh, angular discretization, and treatment of anisotropic scattering consists of analyzing the problem with refined parameters until the changes at the locations of interest are negligible. When results produced by the original and refined model differ by less than 2% at the locations of interest, the uncertainty imparted by the discretization choices is encompassed by the 5% "Other Factors" uncertainty component, and the original model is valid with respect to its spatial and angular discretization.

NRC Request #2

Please describe how the chi-squared divided by degrees-of-freedom statistic is used to assess the validity of FERRET results. Please define the criteria used and subsequent actions taken for unacceptable chi-squared divided by degrees-of-freedom values.

Westinghouse Response to NRC Request #2

The chi-squared divided by degrees-of-freedom (χ^2/DOF) quantity is frequently used as a means of checking "goodness-of-fit". This quantity is reported in the FERRET code output. It is an indicator of how consistent the best estimate neutron spectrum is with the calculated neutron spectrum, the measurements, and their respective uncertainties.

Values of χ^2/DOF ~= 1.0 are indicative of a good statistical fit of the best estimate neutron spectrum data to the measurements, calculations, and their assigned uncertainties. Values of χ^2/DOF < 1.0 indicate the best estimate neutron spectrum deviates from the measurements and calculations less than expected from the assigned uncertainties, perhaps indicating that the assigned uncertainties for the calculated spectrum or measurements may be unnecessarily large. By contrast, values of χ^2/DOF > 1.0 indicate that the best estimate neutron spectrum deviates from the measurements and calculations more than expected from the assigned uncertainties.

Typical uncertainties for the calculated neutron spectrum input to the FERRET code are provided in Section 3.3 of WCAP-18124-NP. The assigned uncertainty for the "fast" portion of the neutron spectrum is 15%. This value is larger than the stated methodology uncertainty in Section 4.5 of WCAP-18124-NP, and also larger than the uncertainty implied by the comparisons of calculations to measurements in Section 5.4 of WCAP-18124-NP. Therefore, Westinghouse expects to observe values of χ^2/DOF less than 1.0 for dosimetry comparisons. Situations where χ^2/DOF values exceed 1.0 warrant further investigation via re-examination of the calculations and measurements.

In the event that χ^2/DOF values exceed 1.0, the first investigative step involves reviewing the calculations. The input data is checked for errors, and the output data is compared to results from similar plants, when available. If no errors are found in the calculations, the next step involves scrutinizing the measurement data.

In some cases, there are known problems with the dosimetry. For example, some surveillance capsule designs experience problems wherein cadmium covers partially fuse with the radiometric monitor material during irradiation, resulting in very high uncertainties in the sample mass. In these cases, the agreement between measurements and calculations is poor, and the affected samples are rejected.

In other cases, individual measurements may be inaccurate. Measurement errors may result from incorrect sample mass evaluations, incorrect processing (e.g., cross-contamination, incomplete chemical separation), or abnormalities in the dosimetry itself.

Westinghouse maintains a large database of historical reactor dosimetry measurements. Measurements from similar plants are expected to exhibit consistent reaction rates when appropriately normalized, because the shape of the neutron spectrum should be nearly identical. Normalized reaction rates that deviate widely from the database average are discarded. Westinghouse discards individual measurements when they deviate by more than 3σ from the database average.

These measures are typically sufficient to reduce the χ^2/DOF quantity to an acceptable level.

NRC Request #3

The text in WCAP-18124-NP, Revision 0, Section 4.4.1, "Core Neutron Source Uncertainties," on page 4-17 implies that peripheral pin power for a given node could be affected by each of three independent uncertainties. That is the pin power could be off by as much as 5% due to in-core measurement uncertainty for a given assembly, 10% due to uncertainty in the relative pin power for a given assembly, and 10% based on variation in the axial peaking factor over the course of a fuel cycle. Please explain how the formulations given in Table 4-15, Case Numbers 3, 4, 7, and 8 relate to the assumption of 10% uncertainty in the fuel assembly relative power shapes given that the expectation based on the text descriptions of these formulations imply a direct multiplication of the nodal pin power by a factor of 1 plus or minus the assumed uncertainty.

Westinghouse Response to NRC Request #3

The formulations given in Table 4-15 are an accurate reflection of the sensitivity studies that were performed with RAPTOR-M3G and the stated results in Section 4.4.

The intent of Cases Numbers 3, 4, 7 and 8 is to capture the effects of uncertainties in the shapes of the radial pin power and axial power distributions. To that end, the total power production from each individual fuel assembly remains unchanged from the reference case; the spatial distribution of power within the assembly is the variable being permuted. The Section 4.4.1 text will be clarified as follows:

- **Pin-by-pin spatial distributions of neutron source at the core periphery** – Core management studies indicate that uncertainties in the spatial distribution of relative pin powers in peripheral fuel assemblies can be on the order of 10%. This parameter is evaluated by holding total assembly power constant while diminishing and intensifying the peripheral pin power gradients.

- **Axial power distribution** – Based on variations in axial peaking factors over the course of a fuel cycle, a 10% uncertainty in the shape of the axial power distribution is considered conservative. This parameter is evaluated by holding total assembly power constant while diminishing and intensifying the axial power gradients.

NRC Request #4

The text in WCAP-18124-NP, Revision 0, Section 4.4.1, "Core Neutron Source Uncertainties," on page 4-17 regarding the uncertainty due to the burnup of the peripheral fuel assemblies is ambiguous when considering corresponding information in Table 4-15 (and corresponding footnotes) and Table 4-16, "Source Permutation-to-Nominal Fast Neutron (E > 1.0 MeV) Fluence Rate Difference at Surveillance Capsule Locations." Please provide clarification regarding the reference burnup and perturbed burnup assumed for Cases 5 and 6 of Table 4-15.

Westinghouse Response to NRC Request #4

The reference case reflects an actual, contemporary power distribution from a Westinghouse 4-Loop plant.

While performing the analytic uncertainty analysis, a series of eleven perturbation cases were conducted by holding the assembly power levels fixed to their reference case levels while uniformly modifying the mid-cycle burnup levels for every fuel assembly in the core. This series of perturbations reveals the effect of changes to the neutron spectrum that result only from changes in the assembly burnup level. The perturbation cases were analyzed with mid-cycle burnup levels ranging from 3,000 MWD/MTU to 50,000 MWD/MTU.

The results show a gradual increase in the calculated fast neutron (E > 1.0 MeV) fluence rate from low to high burnup, consistent with a gradual "hardening" of the neutron spectrum as the distribution of fissioning isotopes shifts from being uranium-dominated at lower burnup levels to being plutonium-dominated at higher burnup levels. To reduce the volume of data presented, the two endpoints of the perturbation study were selected as Case Number 5 and Case Number 6 (3,000 MWD/MTU and 50,000 MWD/MTU), which span a burnup range of 47,000 MWD/MTU. The effect of a 5,000 MWD/MTU uncertainty was calculated by scaling the difference between Case 5 and Case 6 by a factor, F = (5,000 / 47,000).

NRC Request #5

Section 4.3.3, "Summary of Operating Reactor and Calculational Benchmark Results," of WCAP-18124-NP, Revision 0, states: "The H. B. Robinson Unit 2 benchmark represents an experimental configuration that is broadly reflective of most operating reactors: data was collected during full-power operation at a commercial LWR; the power distribution and power history data supporting the analysis were derived using methods similar to those employed by most operating LWRs; geometric dimensions specified are nominal dimensions, and not necessarily identical to their as-built configuration."

In support of WCAP-18124-NP, Revision 0, applicability determination for reactor designs other than Westinghouse three-loop PWRs, provide the reactor designs for the 18 nuclear power plants discussed in Section 5.4, "Operating Power Reactor Comparisons," that have been analyzed with RAPTOR-M3G and FERRET.'

Westinghouse Response to NRC Request #5

Table 1 provides the reactor designs associated with each of the plants listed in Section 5.4 of WCAP-18124-NP.

Table 1
Reactor Designs Reported in Section 5.4 of WCAP-18124-NP

Plant Number	Reactor Design
Domestic Plant #1	Westinghouse 4-Loop
Domestic Plant #2	Westinghouse 4-Loop
International Plant #1	Westinghouse 3-Loop
International Plant #2	Westinghouse 3-Loop
International Plant #3	Combustion Engineering
International Plant #4	Westinghouse 3-Loop
International Plant #5	Westinghouse 2-Loop
International Plant #6	Westinghouse 3-Loop
International Plant #7	Westinghouse 3-Loop
Domestic Plant #3	Westinghouse 4-Loop
Domestic Plant #4	Westinghouse 4-Loop
Domestic Plant #5	Westinghouse 4-Loop
International Plant #8	Westinghouse 3-Loop
International Plant #9	Westinghouse 3-Loop
Domestic Plant #6	Westinghouse 3-Loop
Domestic Plant #7	Combustion Engineering
Domestic Plant #8	Combustion Engineering
Domestic Plant #9	Westinghouse 4-Loop

NRC Request #6

RG 1.190, Section 1.4.2.1, "Operating Measurements," states: "Well documented fluence dosimetry measurements for operating power reactors may be used for methods and data qualification... As capsule and cavity measurements become available, they should be incorporated into the operating reactor measurements data base, and the calculational biases and uncertainties should be updated as necessary." From WCAP-18124-NP, Revision 0, Section 5.4, Westinghouse states that "there are 69 in-vessel surveillance capsules with 295 threshold foil measurements from 18 nuclear power plants that have been analyzed with RAPTOR-M3G and FERRET." However, if Westinghouse supports more than 18 plants both domestically and internationally with RAPTOR-M3G/FERRET, please provide a table similar to Tables 5-10, 5-11, and 5-12, including all of the plants supported by Westinghouse both domestically and internationally to confirm the validity of the uncertainties assigned to the results of RAPTOR-M3G/FERRET calculations. Also, please indicate the reactor design for any added data.

Westinghouse Response to NRC Request #6

The database presented in Section 5.4 of WCAP-18124-NP represents the state of the database at the time the document was written. The current database of reactor dosimetry comparisons performed with RAPTOR-M3G includes comparisons from 77 in-vessel capsules containing 332 threshold foils and 164 ex-vessel dosimetry capsules containing 824 threshold foils that have been removed from 23 nuclear power plants worldwide.

The database is updated periodically in a formal capacity, typically every two or three years, depending on the volume of new comparisons that are available.

Relative to the database that was presented in Section 5.4 of WCAP-18124-NP, comparisons of calculations and measurements applicable to the following reactor types have been added to the database in the intervening time period:

- Westinghouse 3-Loop PWR
- Framatome PWR
- Combustion Engineering PWR
- ABB-Atom BWR
- OKB Gidropress PWR

A high-level summary of the most recent database of RAPTOR-M3G comparisons appears in Table 2. The detailed database can be made available for review by NRC staff at a Westinghouse office.

Table 2
Summary of the 2017 RAPTOR-M3G Dosimetry Comparison Database

	Plant-Averaged Threshold Sensor Reaction Rate Comparisons	
	In-Vessel M/C	**Ex-Vessel Midplane M/C**
Average	1.03	0.92
Std. Dev.	5%	9%
Number of Plants	19	19

	Plant-Averaged In-Vessel Surveillance Capsule BE/C Ratios	
	BE/C Fluence (E>1.0 MeV)	**BE/C DPA**
Average	0.99	1.00
Std. Dev.	6%	5%
Number of Plants	19	19

	Plant-Averaged Ex-Vessel Midplane Capsule BE/C Ratios	
	BE/C Fluence (E>1.0 MeV)	**BE/C DPA**
Average	0.93	0.94
Std. Dev.	9%	9%
Number of Plants	19	19