U. S. NUCLEAR REGULATORY COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

FINAL SAFETY EVALUATION FOR TOPICAL REPORT

WCAP-18124-NP, REVISION 0, "FLUENCE DETERMINATION WITH

RAPTOR-M3G AND FERRET"

WESTINGHOUSE ELECTRIC COMPANY

1.0 INTRODUCTION

By letter dated January 25, 2017, Westinghouse Electric Company (Westinghouse) submitted Topical Report (TR) WCAP-18124-NP, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET," for U.S. Nuclear Regulatory Commission (NRC) staff review and approval for use in licensing applications (Reference 1). A teleconference between NRC staff and Westinghouse was held on June, 1, 2017, to discuss the NRC staff's review scope (Reference 2). During the call, Westinghouse clarified that the fluence calculational methodology submitted in WCAP-18124-NP, Revision 0, is intended for reactor pressure vessel (RPV) beltline fluence estimation in general, but agreed with NRC staff that it provides limited information to support fluence evaluations for extended beltline region and reactor vessel internal components on a generic basis. After review scope clarification, the NRC staff found the material presented in WCAP-18124-NP, Revision 0, to be sufficient to begin the NRC staff's review (Reference 3).

NRC issued a request for additional information (RAI) letter on November, 21, 2017 (Reference 4). Responses to RAIs were received on January, 18, 2018 (Reference 5).

The NRC staff's technical review of WCAP-18124-NP, Revision 0, including responses to RAIs, is provided below in Section 2.

2.0 REGULATORY EVALUATION

Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," describes methods and assumptions acceptable to the NRC staff for determining the RPV neutron fluence with respect to the General Design Criteria (GDC) contained in Appendix A of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 (Reference 6). In consideration of the guidance set forth in RG 1.190, GDC 14, 30, and 31 are applicable. GDC 14, "Reactor Coolant Pressure Boundary," requires the design fabrication, erection, and testing of the reactor coolant pressure boundary so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires, among other things, that components comprising the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical. GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," pertains to the design of the reactor coolant pressure boundary, stating:

> The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) materials properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

NRC Regulatory Issue Summary 2014-011 provides additional information regarding the RPV beltline definition (Reference 7):

> Of particular interest is the reactor vessel beltline, which is defined in 10 CFR Part 50, Appendix G, Section II, "Definitions," as the region of the reactor vessel that "directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage." The beltline region experiences increased embrittlement over the operating period of the reactor vessel as a result of accumulated neutron radiation from the core. Appendix H to 10 CFR Part 50 provides the requirements to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline resulting from exposure to neutron irradiation and the thermal environment. Appendix H to 10 CFR Part 50 states that no material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods that the peak neutron fluence at the end of the design life will not exceed 1×10^{17} neutrons/centimeter-squared (n/cm^2) with energy greater than one million electron volts (E > 1 MeV). Appendix G to 10 CFR Part 50 states, "To demonstrate compliance with the fracture toughness requirements of section IV of this appendix, ferritic materials must be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of Appendix H of this part." Furthermore, Section 2.2 of NUREG-1511, "Reactor Pressure Vessel Status Report," dated December 1994, states, "The NRC staff considered materials with a projected neutron fluence of greater than 1×10^{17} n/cm^2 at end of license (EOL) to experience sufficient neutron damage to be included in the beltline." Therefore, the beltline definition in 10 CFR Part 50, Appendix G is applicable to all reactor vessel ferritic materials with projected neutron fluence values greater than 1×10^{17} n/cm^2 (E > 1 MeV), and this fluence threshold remains applicable for the design life as well as throughout the licensed operating period.

3.0 TECHNICAL EVALUATION

Background

The neutron fluence calculational methodology described in WCAP-18124-NP, Revision 0, is fundamentally the same as Westinghouse methods previously reviewed and approved by the NRC (References 8 and 9). The major difference is the replacement of either of the industry standard DORT and TORT neutron and gamma ray transport codes, which are available from the Radiation Safety Information Computational Center maintained by Oak Ridge National

Laboratory (ORNL) (Reference 10), with the Westinghouse-developed RAPTOR-M3G code. RAPTOR-M3G is fundamentally the same as the TORT code. However, Westinghouse has made improvements to the computational framework relative to TORT in order to increase computational stability and to overcome limitations associated with performing real-world neutron transport calculations on a single computer processor. Improvements in stability are described in detail in WCAP-18124-NP, Revision 0, Section 2.5, "Discrete Ordinates Transport Calculations with RAPTOR-M3G," and are based on not using zero-weighted initiating directions. That is, RAPTOR-M3G uses the technique of Lathrop and Brinkley (Reference 11). Westinghouse explains that the consistency between RAPTOR-M3G and TORT as shown in a previous study in WCAP-17993-NP indicates that use of the Lathrop and Brinkley method instead of the zero-weighting initiating directions method used in TORT are insignificant for RPV neutron fluence calculations (Reference 12).[1] Additionally, Westinghouse explains that:

> RAPTOR-M3G includes implementation of the Directional Theta-Weighted (DTW) spatial differencing scheme, whereas TORT does not. The DTW scheme is discussed and endorsed in RG 1.190...The DTW scheme generally produces improved results, as compared to traditional theta-weighted (TW) schemes.

Also, limitations in the amount of detail that can be represented in large TORT models have been removed by allowing large problems to be distributed across the memory of multiple computers via parallel processing with RAPTOR-M3G allowing for substantial decreases in problem solution time.

Since the methodology presented in WCAP-18124-NP, Revision 0, is not a major change from previously approved Westinghouse methods, the NRC staff's review focused on the differences.

<u>Method Applicability</u>

WCAP-18124-NP, Revision 0, states: "The neutron exposure calculational methodology is qualified against the requirements of RG 1.190, and estimates of the uncertainty and bias are provided. The methodology is qualified in both benchmark and power reactor neutron fields." WCAP-18124-NP, Revision 0, also states:

> Additionally, as plants enter into periods of extended operation, there has been increased interest in accurately quantifying neutron exposure levels outside of the traditional beltline region of the reactor vessel. Areas of interest include the "extended beltline" of the reactor vessel, comprising portions of the reactor vessel located above and below the elevation of the reactor core; and reactor internals components and materials, particularly those located above and below the elevation of the reactor core. Two-dimensional transport methods have shown limitations in their ability to accurately determine neutron exposure quantities in these regions. As a result, there is a need to move to efficient (three-dimensional (3D)) techniques for neutron fluence determination.

However, WCAP-18124-NP, Revision 0, only quantifies bias and uncertainty estimates for the traditional RPV beltline region approximated by the RPV region near the active height of the core. If Westinghouse would also like to apply the WCAP-18124-NP, Revision 0, methodology to components in the extended RPV beltline region, defined during license renewal, and/or for

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[1] Westinghouse notes the similarity of the values listed in the bottom two rows of WCAP-17993-NP, Table 2-2 through Table 2-4.

reactor vessel internal components of interest then: (1) bias and uncertainty estimates must be developed for all specific applications of the method, and (2) supporting measurement data must be provided. Consequently, applicability of WCAP-18124-NP, Revision 0, is limited to the traditional RPV beltline region.

Impact on the Pressure-Temperature Limits Report

The cover letter for WCAP-18124-NP, Revision 0 (Reference 1), states:

> Standard Technical Specifications (STS) for Pressure Temperature Limits Report (PTLR), Section 5.6.4.b, ‘RCS PRESSURE AND TEMPERATURE LIMITS REPORT,” states: *The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents: ...*

It further explains:

> For those licensees that have applied WCAP-14040-A, the Technical Specifications (TS) will specifically reference TR. By having the safety evaluation recognize that the methodology discussed in WCAP-18124 is an acceptable fluence determination alternate input to those discussed in Section 2.2 of WCAP-14040-A, it will be clear that a change to the TS is not required.

However, STS Section 5.6.4.b in the Revision 4 of NUREG-1431 (Reference 13) has a note that states:

> -----------------------------------REVIEWER’S NOTE--
> Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the PTLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the PTLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.
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STS Section 5.6.4 also states: “Identify the NRC staff approval document by date.”

Based on the above STS guidance, a change to the TS is required for licensees who adopt the approved version of WCAP-18124-NP, Revision 0, as the licensing basis neutron fluence calculational methodology supporting pressure-temperature limits in TS. The STS guidance is applicable to all methodologies whether these methodologies are new, revised or equivalent to the existing methodologies. The STS guidance does not differentiate between various methodologies.

Neutron Fluence Calculational Methods

In general, the guidance provided in RG 1.190 supports the validation of an acceptable RPV fluence calculation with the major components being:

1) Fluence estimation using an appropriate calculational methodology based on Regulatory Positons 1.1-1.3

2) Methodology qualification and uncertainty estimates based on Regulatory Position 1.4:

   - Analytic uncertainty analysis identifying possible sources of uncertainty.

   - Comparisons with benchmark measurements and calculations from applicable test facilities including:

     - Plant-specific operating reactor measurements
     - Pressure vessel simulator measurements
     - Calculational benchmarks

Fluence Calculations Using RAPTOR-M3G

A solution to the linear Boltzmann transport equation (BTE) is approximated using the proprietary 3D discrete ordinates code known as RAPTOR-M3G as described in Section 2, "Fluence Calculations with RAPTOR-M3G," of WCAP-18124-NP, Revision 0. Three-dimensional flux solutions are directly calculated rather than constructed using a synthesis of azimuthal, axial, and radial flux. An appropriate cross-section library based on ENDF/B-VI nuclear data is used, which is intended for use in light water reactor (LWR) shielding and RPV dosimetry applications. Numeric approximations include a P_3 (or higher) Legendre expansion to represent anisotropic scattering, and S_8 (or higher) angular quadrature for angular flux discretization. These cross-section data and modeling approximations supporting the neutron transport calculation adhere to the modeling guidance in RG 1.190 and are therefore acceptable. The response to RAI 1 describes additional spatial domain meshing, angular quadrature, and anisotropic scattering studies that support general use of P_3 for anisotropic scattering representation and S_8 angular quadrature for both the RPV beltline region and locations in the reactor cavity along the height of the core. The response to RAI 1 also clarifies that: "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."

Space and energy dependent core power distributions and associated core parameters are treated on a fuel cycle and plant-specific basis in order to obtain neutron source distributions that are appropriately averaged over each fuel cycle. Fuel cycle and plant-specific treatment includes explicit accounting of initial enrichment, burnup, and axial power distributions. Neutron source energy spectral effects, neutrons per fission, and energy per fission are accounted for by using appropriate fission fractions for the fissionable uranium and plutonium isotopes based on the initial enrichment and burnup history of the fuel assemblies that are the major contributors to the RPV fluence. The neutron source specification adheres to the modeling guidance in RG 1.190 and is therefore acceptable.

As previously discussed, WCAP-18124-NP, Revision 0, Section 2.5, lists two methodological differences between previously approved Westinghouse methods that allowed use of the TORT code and the new method described in WCAP-18124-NP, Revision 0, that uses RAPTOR-M3G as follows:

(1) RAPTOR-M3G does not use zero-weighted initiating directions when solving the BTE,[2] and

(2) RAPTOR-M3G has the added capability of using the DTW spatial differencing scheme instead of only the traditional TW scheme.

TW and DTW are both acceptable spatial differencing schemes that adhere to the guidance in RG 1.190. The main benefit of DTW is a reduction of unphysical numerical oscillations, which is expected to lead to reduced fluence calculational uncertainty (Reference 14). Simulator benchmarks, operating reactor and calculational benchmarks, and the analytical uncertainty analysis,[3] all discussed in more detail below in the following subsection, were performed with both TW and DTW schemes. There was a negligible difference in fluence estimates, therefore use of either the TW or the DTW spatial differencing scheme as described in WCAP-18124-NP, Revision 0, is acceptable when using RAPTOR-M3G.

Methodology Qualification and Uncertainty Estimates

Regulatory Position 1.4.1, 1.4.2, and 1.4.3 from Table 1, "Summary of Regulatory Positions on Calculation and Dosimetry," of RG 1.190 states the following:

> The calculational methodology must be qualified by both (1) comparisons to measurement and calculational benchmarks and (2) an analytic uncertainty analysis. The methods used to calculate the benchmarks must be consistent (to the extent possible) with the methods used to calculate the vessel fluence. The overall calculational bias and uncertainty must be determined by an appropriate combination of the analytic uncertainty analysis and the uncertainty analysis based on the comparisons to the benchmarks.

The NRC staff found that the fluence calculational methodology is consistent with the previously approved methodologies described in WCAP-14040-A, Revision 4, and WCAP-16083-NP-A, Revision 0, and with the guidance set forth in RG 1.190.

Analytic Uncertainty Analysis

The analytic uncertainty analysis identifies the important sources of uncertainty specific to RAPTOR-M3G. Westinghouse identifies the three main components of uncertainty as:

- Uncertainties in the core neutron source
- Uncertainties in the as-built thicknesses and locations of the reactor vessel and internal components

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[2] Instead a step function approximation is used for initiation which has the effect of using fewer directions for a given S_n order (Reference 11). Westinghouse also provided support that this method change is insignificant for RPV neutron fluence calculations.

[3] Westinghouse notes that, "in general, the analytic uncertainty values are consistent between the two differencing schemes; however, in cases where there are differences, the higher uncertainty values were selected." This is conservative because the higher uncertainty is applied independent of the differencing scheme used, therefore this treatment is acceptable.

- Uncertainties in the full power coolant temperatures

Sensitivity studies were performed by varying several parameters, within each main category above, over ranges of estimated variation. All major phenomena known to be important to RPV fluence estimation has been accounted for consistency with RG 1.190, Section 1.4.1, "Analytic Uncertainty Analysis." Therefore, the analytic uncertainty analysis performed is appropriate. Furthermore, considering that the response to RAI 5 has shown that the Westinghouse operating reactor measurement database is well represented by a variety of reactor designs with excellent agreement between measurements and calculations, this further supports the appropriateness of the uncertainty estimate of 11 percent assigned to analytic sensitivity studies as part of the net calculational uncertainty discussed in WCAP-18124-NP, Revision 0, Section 4.5, "Estimate of Bias and Uncertainty."

Reactor Vessel Simulator Benchmarking

Reactor vessel simulator benchmarking was performed as specified in RG 1.190. Calculations were compared with the benchmark measurements from the Poolside Critical Assembly (PCA) simulator at ORNL and the VENUS-1 experiment. The NRC staff determined these to be acceptable test facilities as they are specifically referenced in RG 1.190 for use as acceptable benchmarks for fluence method qualification.

The PCA benchmark calculations were compared (at 7 distinct locations) with the benchmark measurements and found to be in excellent agreement. This demonstrates the method's capability to model the cylindrical pressurized water reactor (PWR) vessel geometry and associated features, including dosimetry, using spatial discretization with a Cartesian grid.

Similar to the PCA benchmark, the VENUS-1 benchmark is used to demonstrate a method's capability to model the cylindrical PWR vessel geometry and associated features. However, VENUS-1 is a full scale benchmark with 41 distinct measurements at a variety of locations from the fuel to the neutron pad region. Again, there was excellent agreement between calculations and measurements. A summary of comparisons between measurements and calculations, including both PCA and VENUS-1 experiments broken down by dosimeter type, are provided in Tables 4-7, "Summary of Simulator Benchmark Measured-to-Calculated (M/C) Reaction Rate Comparisons (TW Differencing)," and 4-8, "Summary of Simulator Benchmark M/C Reaction Rate Comparisons (DTW Differencing)," in TR WCAP-18124-NP, Revision 0, in a more condensed format.

Calculational Benchmarking

Calculational benchmarking was performed as specified in RG 1.190. Both PWR and boiling water reactor (BWR) calculational benchmarks were analyzed. Fluence calculations with RAPTOR-M3G demonstrate excellent agreement between RAPTOR-M3G calculations and the reference benchmark calculations for both the PWR and BWR cases with relative differences mostly within 5 percent. This level of agreement provides reasonable assurance that the numerical procedures, code implementation, and the various modeling approximations are generally appropriate for LWR RPV traditional beltline fluence calculations.

Operating Reactor Measurement Benchmarking

Operating reactor measurement benchmarking was performed as specified in RG 1.190. Calculations were compared with the benchmark measurements from the H.B. Robinson Unit 2 (HBR-2) power reactor benchmark experiment (Reference 15). The Cycle 9 dosimetry benchmark calculation was compared with measurements and found to be in excellent agreement, demonstrating the ability to appropriately select and implement transport method options, nuclear data libraries, material specification, geometry, etc., specific to a Westinghouse 3-loop reactor design.

Westinghouse believes that the HBR-2 benchmark is applicable to most LWRs. In its response to RAI 5, Westinghouse also shows that its operating reactor measurement database is comprised of measurements from Westinghouse 2-, 3-, and 4-loop reactor designs in addition to Combustion Engineering designs. Since there is no discernible difference between average M/C reaction rate values from different PWR reactor design types in Westinghouse's operating reactor measurement database, the NRC staff agrees that the HBR-2 benchmark is applicable to most LWRs (specifically PWRs) in the traditional beltline region, and by extension the WCAP-18124-NP, Revision 0, fluence calculational methodology is also applicable to most LWRs in the traditional beltline region.

Fluence Calculational Uncertainty

RG 1.190 Regulatory Position 1 and 1.4.3 from Table 1, "Summary of Regulatory Positions on Calculation and Dosimetry," states:

> The vessel fluence (1 sigma) calculational uncertainty must be demonstrated to be less than or equal to 20 percent for RT-PTS and RT-NDT determination. In these applications, if the benchmark comparisons indicate differences greater than 20 percent, the calculational model must be adjusted or a correction must be applied to reduce the difference between the fluence prediction and the upper 1-sigma limit to within 20 percent. For other applications, the accuracy should be determined using the approach described in Regulatory Position 1.4, and an uncertainty allowance should be included in the fluence estimate as appropriate in the specific application.

Within the traditional beltline, all expected uncertainties are calculated to be less than 20 percent, as shown in WCAP-18124-NP, Revision 0, Section 4.5, "Estimate of Bias and Uncertainty," which adheres to the guidance in RG 1.190, Section 1.4.3, "Estimate of Fluence Calculational Bias and Uncertainty." The extensive evaluation of dosimetry sets, summarized in WCAP-18124-NP, Revision 0, Section 4.5, "Operating Power Reactor Comparisons," provides additional validation that the total calculational uncertainty is expected to be less than 20 percent. Examination of Table 5-11, "In-Vessel Surveillance Capsules Best-Estimate-to-Calculated (BE/C) Reaction Rate Ratios," and Table 5-12, "EVND Core Midplane BE/C Reaction Rate Ratios," suggest the presence of calculational bias factors on the order of 0.98 (standard deviation of 6 percent) and 0.92 (standard deviation of 6 percent), respectively. This means that based on rigorous consideration of both calculation and measurement uncertainties using least squares adjustment with FERRET, calculated fluences are slightly over-predicted on average.[4] While the data indicates that the WCAP-18124-NP,

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[4] The NRC staff notes that bias indication based on least squares adjustment with FERRET is theoretically more reliable than bias indication provided by M/C values because M/C values are not

Revision 0, calculational fluence methodology is slightly conservative on average, the bias uncertainty is not significantly smaller than the bias for bias correction to be warranted.

After reviewing the various methods qualification benchmarking activities and uncertainty analyses supporting the calculation of RPV neutron fluence, the NRC staff has reasonable assurance that RPV neutron fluence can be appropriately estimated using the methodology described in TR WCAP-18124-NP, Revision 0.

Least Squares Adjustment with FERRET

Westinghouse notes that least squares adjustment with FERRET is identical to that described in WCAP-16083-NP-A, Revision 0. As there is no significant change when implementing FERRET for determining best-estimate fluence values using the least squares adjustment technique previously reviewed and approved by the NRC staff and documented in WCAP-16083-NP-A, Revision 0, the NRC staff verified that the implementation of FERRET is unchanged and remains qualified for determining best-estimate fluence values as documented in the following summaries.

Validation of the Transport Calculation

RAPTOR-M3G, a neutron transport code qualified for RPV fluence determination, was discussed in the previous sections. RAPTOR-M3G is an essential part of the FERRET procedure to determine the initial input neutron spectrum at sensor measurement locations.

Uncertainty Input to the Least-Squares Adjustment Procedure

Input neutron spectrum uncertainty corresponding to plant-specific transport calculations at sensor locations is accounted for. The uncertainty is specified to be consistent with the overall uncertainty assigned to RAPTOR-M3G based on benchmarking results.

Reaction Rate Measurement and Uncertainties

Input of the qualified measured reaction rates and associated uncertainties is accomplished by adhering to relevant dosimeter-dependent American Society for Testing and Materials (ASTM) standards. Qualification of neutron fluence measurement methods, the subject of RG 1.190, Regulatory Position 2, is discussed as a part of FERRET qualification, which provides: (1) A well characterized measurement procedure, (2) a validated measurement procedure, and (3) an acceptable method for deriving neutron fluence (or reaction rates) from detector responses.

Dosimetry Cross Sections and Uncertainty

Acceptable dosimetry cross-sections with uncertainties from the extensively tested industry standard SNLRML cross-section library are used.

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subject to any rigorous treatment of measurement and calculation uncertainties. Consequently, it is not surprising that Table 5-10, "In-Vessel and Ex-Vessel Capsules Threshold Reactions M/C Reaction Rate Ratios," indicates a slight under-prediction for in-vessel fluence calculations on average when Table 5-11 indicates slight over-prediction on average.

Data Comparison in the National Institute of Standards and Technology U-235 Fission Field

The National Institute of Standards and Technology (NIST) U-235 fission field experiment with known fission spectra [5] was used to assess the: (1) correct SNLRML cross-section library implementation, and (2) proper broad-group cross-section library generation, by using reference neutron field spectra at measurement locations as a weighting function.[6] FERRET, using the SNLRML cross-section library, produces acceptable best-estimate fluence values for the NIST U-235 fission field experiment. That is, FERRET-adjusted calculated reaction rates, from the NIST U-235 fission field experiment, yielded results that were consistent with prescribed input uncertainty bounds for typical power reactor sensor sets and the sensor set used in the PCA benchmark.

Evaluation of the H. B. Robinson Benchmark

FERRET-adjusted calculated reaction rates from the HBR-2 benchmark experiment yield BE/C[7] reaction rate ratios with less variance about the average than unadjusted M/C values as expected. Additionally, the adjustments were consistent with prescribed input uncertainties.

FERRET Sensitivity Studies

Westinghouse originally performed FERRET sensitivity studies using DORT/TORT (as documented in the WCAP-16083-NP-A) and later performed similar sensitivity studies, as documented in WCAP-18124-NP, Revision 0, using RAPTOR-M3G to evaluate the impact of: (1) spectral uncertainties and (2) variation in composition of the dosimetry set on the least squares adjustment of the calculated spectrum. The conclusions remain the same with respect to minimum acceptable dosimetry sets with fission monitors (i.e., Fe, U-238, and Np-237). In addition a new minimum acceptable dosimetry set without fission monitors was identified (i.e., Nb-93 and Fe), which is acceptable based on minimizing uncertainty in the adjusted calculated fluence while maintaining adequate spectral coverage. The spectral uncertainty sensitivity study results were also consistent with the results of the previously approved implementation of FERRET in WCAP-16083-NP-A. That is, (1) the adjusted calculated fluence is very insensitive to the magnitude of the input spectral uncertainties and (2) there is increasing sensitivity of the associated calculational uncertainty with increasing magnitude of the input spectral uncertainties. As noted by the NRC staff in WCAP-16083-NP-A, "these studies are not a necessary part of the adjustment procedure but are instructive to the dosimetry analyst."

Summary

The only changes to the use of FERRET noted by Westinghouse are:

- Minor updates made to some of the uncertainty estimates to reflect the latest versions of ASTM standards and current practices.

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[5] Two benchmark quality reference fields are used: a Cf-252 spontaneous fission field and a U-235 thermal fission field.

[6] See WCAP-18124-NP, Table 5-3, "Comparison of Calculated U-235 Fission Spectrum Averaged Cross-Sections."

[7] Used interchangeably with adjusted-calculated-to-calculated (A/C) by Westinghouse.

- Minor updates made to some of the historical reaction rate measurements to reflect the best available data.

- The addition of Nb-93 as an acceptable neutron sensor with measurement procedures and associated uncertainties adhering to relevant ASTM standards.

- The input uncertainty sensitivity study was re-evaluated with RAPTOR-M3G instead of DORT/TORT and a different reference data set, but the conclusions are unchanged relative to the previously approved implementation of FERRET.

The NRC staff also found that the PCA simulator benchmark was not analyzed using FERRET; however, it was analyzed with FERRET in the WCAP-16083-NP-A TR. The NRC staff finds this acceptable because the H. B. Robinson benchmark was re-evaluated and this evaluation is sufficient to demonstrate proper implementation of FERRET for commercial LWRs with dosimetry typical of Westinghouse fluence evaluations.

Based on confirming that the implementation of FERRET as described in WCAP-18124-NP, Revision 0, is essentially the same as that previously approved in WCAP-16083-NP-A, Revision 0, with minor changes that serve as methodology enhancements, the NRC staff finds the least squares adjustment of calculated fluence using the FERRET code to be acceptable.

4.0 LIMITATIONS AND CONDITIONS

1. Applicability of WCAP-18124-NP, Revision 0, is limited to the RPV region near the active height of the core based on the uncertainty analysis performed and the measurement data provided. Additional justification should be provided via additional benchmarking, fluence sensitivity analysis to response parameters of interest (e.g., pressure-temperature limits, material stress/strain), margin assessment, or a combination thereof, for applications of the method to components including, but not limited to, the RPV upper circumferential weld and reactor coolant system inlet and outlet nozzles and reactor vessel internal components.

2. Least squares adjustment is acceptable if the adjustments to the M/C ratios and to the calculated spectra values are within the assigned uncertainties of the calculated spectra, the dosimetry measured reaction rates, and the dosimetry reaction cross sections. Should this not be the case, the user should re-examine both measured and calculated values for possible errors. If errors cannot be found, the particular values causing the inconsistency should be disqualified.

5.0 CONCLUSION

The NRC staff has reviewed the calculational fluence methodology described in WCAP-18124-NP, Revision 0, and finds that the method adheres to the guidance in RG 1.190. Therefore, this methodology is acceptable for use in calculating RPV neutron fluence provided that the limitations and conditions listed in Section 4.0 of this safety evaluation report are met.

6.0 REFERENCES

1. Submittal of Westinghouse Electric Company Topical Report WCAP-18124-NP, Revision 0, “Fluence determination with RAPTOR-M3G and FERRET,” LTR-NRC-17-7 dated January 25, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17030A377).

2. Letter from E. Lenning (U. S. Nuclear Regulatory Commission) to J. Gresham (Westinghouse Electric Company), “Summary of Closed Call with Westinghouse Electric Company to Discuss Acceptance Review Findings for WCAP-8124-NP, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET,” dated July 26, 2017 (ADAMS Accession No. ML17164A228).

3. Letter from D. Morey (U. S. Nuclear Regulatory Commission) to J. Gresham (Westinghouse Electric Company), “Acceptance for Review of Westinghouse Electric Company Topical Report WCAP-18124-NP, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET,” dated June 27, 2017 (ADAMS Accession No. ML17166A063).

4. Letter from E. Lenning (U. S. Nuclear Regulatory Commission) to J. Gresham (Westinghouse Electric Company), “Request for Additional Information Re: Westinghouse Electric Co. Topical Report WCAP-18124-NP, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET,” dated November 21, 2017 (ADAMS Accession No. ML17290A147).

5. Letter from J. Gresham (Westinghouse Electric Company) to U. S. Nuclear Regulatory Commission “Response to the NRC Request for Additional Information on the RAPTOR-M3G and FERRET Topical Report,” LTR-NRC-18-5 dated January 18, 2018 (ADAMS Accession No. ML18018B347).

6. U. S. Nuclear Regulatory Commission Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” 2001 (ADAMS Accession No. ML010890301).

7. U. S. NRC Regulatory Issue Summary 2014-11, “Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components,” 2014 (ADAMS Accession No. ML14149A165).

8. Submittal of Westinghouse Electric Company Topical Report WCAP-14040-A, Revision 4, “Methodology Used To Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” 2004 (ADAMS Accession No. ML050120209).

9. Submittal of Westinghouse Electric Company Topical Report WCAP-16083-NP-A, Revision 0, “Benchmark Testing of the FERRET Code for Least Squares Evaluation of Light Water Reactor Dosimetry,” May 2006 (ADAMS Accession No. ML061600256).

10. “RSICC Computer Code Collection CCC-650, DOORS 3.2a, One- Two- and Three Dimensional Discrete Ordinates Neutron/Photon Transport Code System,” Oak Ridge National Laboratory, 2007.

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Attachment: Comment Resolution Table

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