ANALYSIS OF PNAR SPENT FUEL SAFEGUARDS MEASUREMENTS USING THE ORIGEN DATA ANALYSIS MODULE

Jianwei Hu[1,a], Germina Ilas[1], Ian C. Gauld[1,b], Stephen J. Tobin[2], Topi Tupasela[3], Tapani Honkamaa[3], and Marita Mosconi[4]

[1]Oak Ridge National Laboratory, USA; [2]Los Alamos National Laboratory, USA; [3]Radiation and Nuclear Safety Authority, Finland; [4]Directorate General for Energy, European Commission

ABSTRACT

This paper summarizes the analysis of the Passive Neutron Albedo Reactivity (PNAR) measurements using the ORIGEN data analysis module for 23 boiling water reactor spent fuel assemblies. The measurements were performed in Finland under an international collaboration on spent fuel safeguards verification methods. PNAR is part of the proposed integrated nondestructive system to be used for safeguards verifications at the planned Finnish encapsulation facility. In addition to measuring passive neutron and gamma emission rates from an assembly like a Fork detector, PNAR also measures the PNAR ratio, which is expected to correlate with the fissile content in the assembly. The emission rates and PNAR ratio can be used to verify the operator declarations and the fissile content of an assembly, respectively. The analysis was performed using the ORIGEN data analysis module, referred to as the *ORIGEN Module* in this paper, which was originally developed to predict Fork detector neutron and gamma signals for spent fuel measurements and was integrated into the Integrated Review and Analysis Package developed by the European Atomic Energy Community and the International Atomic Energy Agency. The ORIGEN Module includes the ORIGEN burnup analysis code and integrates detector response functions pre-generated via the MCNP code to predict detector signals in several seconds per assembly. In this study, new response functions specific to PNAR measurements were generated for the ORIGEN Module. The study also analyzed the effects of using detailed fuel design and operating information vs. standard safeguards information on results that the ORIGEN Module calculated. Using detailed information reduced the standard deviation of relative differences between calculated and measured neutron count rates among the 23 assemblies from ~10 to ~4%. The results obtained via standard safeguards information for these PNAR measurements were similar to those obtained for the Fork detector. A clear trend was found between the calculated net neutron multiplications and the measured PNAR ratios of the 23 assemblies. This paper describes how the ORIGEN Module calculates the expected PNAR neutron and gamma signals and PNAR ratio and how they compare with corresponding measured values.

Keywords: spent fuel safeguards, PNAR, ORIGEN

1. INTRODUCTION

This paper summarizes the analysis of the Passive Neutron Albedo Reactivity (PNAR) measurements using the ORIGEN data analysis module for 23 boiling water reactor (BWR) spent fuel assemblies. The measurements were performed in Finland under Action Sheet 65. This action sheet is an international collaboration among the US Department of Energy (DOE)/National Nuclear Security Administration (NNSA), represented by Los Alamos National Laboratory and Oak Ridge National Laboratory, the European Atomic Energy Community (Euratom), and the Radiation and Nuclear Safety Authority of Finland (STUK). These organizations collaborated on spent fuel safeguards verification methods in the context of the Finnish spent fuel encapsulation/repository system. PNAR is part of the proposed integrated nondestructive system to be used for safeguards verifications at the planned Finnish encapsulation facility [1] [2]. Like a Fork detector, the PNAR instrument measures passive neutron and gamma emission rates from a spent fuel assembly. PNAR also measures the PNAR ratio, which is a ratio of the PNAR neutron signal with and without the Cd liner in place. Through Monte Carlo N-Particle (MCNP) simulations, Tobin et al. demonstrated that the PNAR ratio

[a] Email: huj1@ornl.gov
[b] Retired from Oak Ridge National Laboratory

correlates with the net neutron multiplication of an assembly [3], which depends on the amount of fissile content and neutron absorbers present in an assembly. Therefore, the PNAR ratio provides a way to verify the fissile content in a spent fuel assembly.

Predictions of the PNAR neutron and gamma signals and the PNAR ratio of a particular spent fuel assembly based on the provided operator declarations (e.g., burnup, initial enrichment, cooling time) of the assembly will help safeguards inspectors determine whether the measured PNAR signals and ratio are consistent with the assembly's declarations. A spent fuel assembly is a complex system that contains thousands of nuclides whose quantities could vary as functions of several factors, such as the initial U loading, burnup, and moderator density of the assembly. It is difficult, if not impossible, to predict spent fuel assembly measurement signals without a sophisticated computer software. The ORIGEN data analysis module [4], referred to as the *ORIGEN Module* in this paper, was developed previously to predict Fork detector spent fuel measurement signals in real time and was integrated into the Integrated Review and Analysis Program developed by Euratom and the International Atomic Energy Agency. This module includes the ORIGEN burnup analysis code distributed as part of the SCALE v6.1.3 nuclear systems modeling and simulation suite [5], and it integrates detector response functions pre-generated via the MCNP code [6] to predict the detector signals in several seconds per assembly. The ORIGEN Module was benchmarked against Fork measurement data for more than 300 light water reactor assemblies [7], and the module was recently expanded to VVER-440 assemblies for Fork detector measurements [8] in addition to BWR and pressurized water reactor assemblies.

In this study, the ORIGEN Module was updated for PNAR measurements by generating new response functions. The updated ORIGEN Module was then used to analyze PNAR measurements of the 23 BWR assemblies. Two datasets for the fuel design and operating conditions of these 23 assemblies were provided: (1) a set of basic data (e.g., assembly-average burnup, initial enrichments) similar to the operator declarations in a typical safeguards inspection—referred to as *safeguards data* in this report—which was provided by STUK, and (2) a set of detailed data (e.g., detailed burnup and moderator density values along the height of the assemblies) provided by the reactor operator Teollisuuden Voima Oyj (TVO). Both sets of data were used in the ORIGEN Module calculations, and this paper presents the results. Net neutron multiplication was also calculated with the ORIGEN Module and compared with the measured PNAR ratio. A more comprehensive analysis of the PNAR measurement using the ORIGEN Module is provided in Hu, Ilas, and Gauld [9].

2. PNAR MEASUREMENTS

The PNAR measurements are described in detail in other works [1] [2], so only a brief summary is provided here. In July 2019, STUK measured 23 different BWR assemblies at the Olkiluoto Nuclear Power Plant spent fuel storage facility. The cooling times of these assemblies ranged from 6.2 to 35.1 years, and burnups ranged from 18,589 to 49,698 MWd/tU. The assemblies included seven different assembly designs with three different assembly lattices: 8 × 8, 9 × 9, and 10 × 10. Three assemblies experienced noncontinuous irradiation cycles in which they were removed from the core after irradiation, stored outside the core for one or more cycles, and reinserted in the core for further irradiation. Measurements were centred on a location approximately 1.4 m from the bottom of the assembly. There is approximately 0.4 m of support structures and natural U at the bottom of the assembly below the enriched U zone [1]. The operator simulation data from TVO for these assemblies were provided for 25 equal-length axial nodes. Assuming that a natural U zone was present in the first bottom node (~15 cm) of most assemblies, the measurements were made ~115 cm above the bottom of the active fuel, which correspond to node 8 from the bottom of the assembly. The measured PNAR signals include the neutron and gamma count rates and the PNAR ratio of each assembly.

3. GENERATION OF PNAR RESPONSE FUNCTIONS

The response functions record the MCNP-simulated detector responses due to a source neutron or photon particle emitted from the spent fuel. 3D MCNP models of the measurement system are needed to adequately calculate the PNAR detector response caused by a neutron or photon originated in the spent fuel assembly, given the axial and radial variations of burnup values, nuclide compositions, and radiation emission sources in the spent fuel assembly. Only the axial burnup profiles were considered for response function generation in this study because the radial burnup variation is not as important as the axial variations.

Two BWR assembly designs were selected to generate the spent fuel nuclide compositions and neutron/gamma source terms for use in the response function generation: (1) a SVEA64 design to represent all 8×8 assemblies and (2) a SVEA100 design to represent all 10×10 and 9×9 assemblies. Both representative BWR assemblies have a burnup of 35 GWd/tU, an initial enrichment of 3.0%, and a cooling time of 20 years. The 3D spent fuel nuclide compositions and neutron and gamma source terms, which are necessary for response function generation, of the two representative BWR assemblies were calculated via the ORIGAMI code in SCALE v6.2.4 [10] based on a representative 25 node axial profile for burnup and moderator density.

Figure 1 shows the 3D MCNP model used to generate the response functions for the PNAR measurement of a SVEA64 spent fuel assembly, which is consistent with the actual configuration of the PNAR instrument used in the measurement. The top part of the assembly is not shown on the left in Figure 1 because of limited space. The Cd liner is placed between the instrument and the fuel assembly in the "with Cd liner" case, and the liner is moved down 60 cm in the "without Cd liner" case, as labeled in Figure 1. There are four detector pods in PNAR: one on each side of the assembly. Each detector pod contains a ^{3}He tube for neutron detection and an LND 52110 ion chamber for gamma detection. More details of the PNAR instrument are provided in other works [3].

For the PNAR neutron response functions, 20 discrete source neutron energy bins were used to span 0.01–20 MeV, and each energy bin was simulated in a separate MCNP model. Because most neutrons are born at ~2 MeV in spent fuel, this neutron energy discretization was deemed sufficient. In each MCNP model, a fixed-source calculation was performed with MCNP6 (v6.1) [6], and the neutron source particles were sampled uniformly in the radial direction of the fuel assembly but nonuniformly in the axial direction based on the calculated neutron emission probability along the assembly axis. The neutron capture rates in the ^{3}He gas in all the PNAR ^{3}He tubes were tallied in these models to mimic the PNAR neutron count rates. The gamma count rates were calculated based on tallies for gamma dose rates deposited in the gas of the ion chamber, given that the gamma count rates are expected to be proportional to dose rates. ANSI/ANS 1977 flux-to-dose factors [6] were used to convert gamma flux into dose rates. Sufficient neutron and gamma particle history were used in these MCNP calculations, and less than 0.5% statistical uncertainty was achieved in the neutron and gamma response functions at all energy bins, except for the bin with the lowest energy.

Figure 2 illustrates the PNAR neutron count rate response functions for the modeled SVEA64 assembly design. Neutron response functions for the "without Cd liner" case were 4–24% higher than those for the "with Cd liner" case at varying energies, which was expected because the Cd liner absorbs thermal neutrons returning to the assembly from surrounding materials and thus reduces induced fissions in the assembly. Response functions generated for the SVEA100 design were similar to the ones shown here, but they are not presented for brevity.

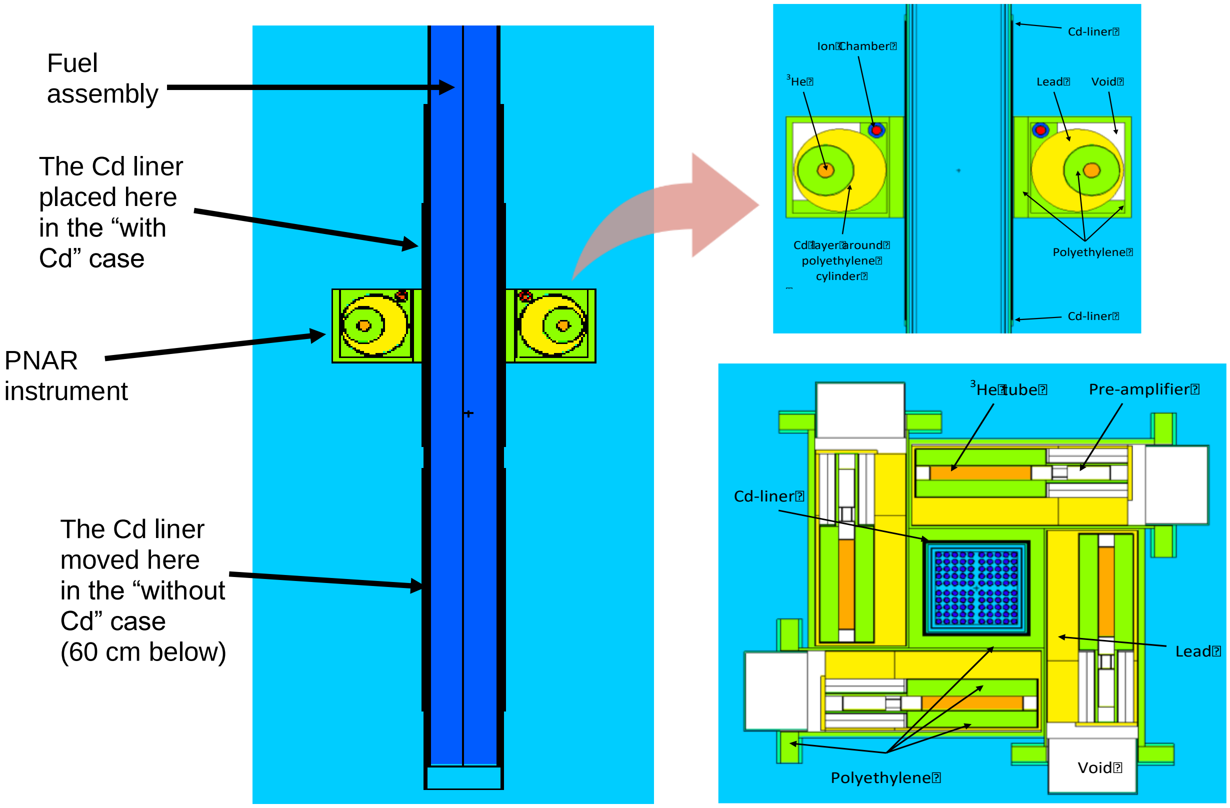

Figure 1. Side view of the MCNP model of the PNAR measurement with an assembly in the middle (top part of the assembly not shown) (left); magnified view of the PNAR detector (upper right); top view of the MCNP model of the PNAR measurement (lower right).

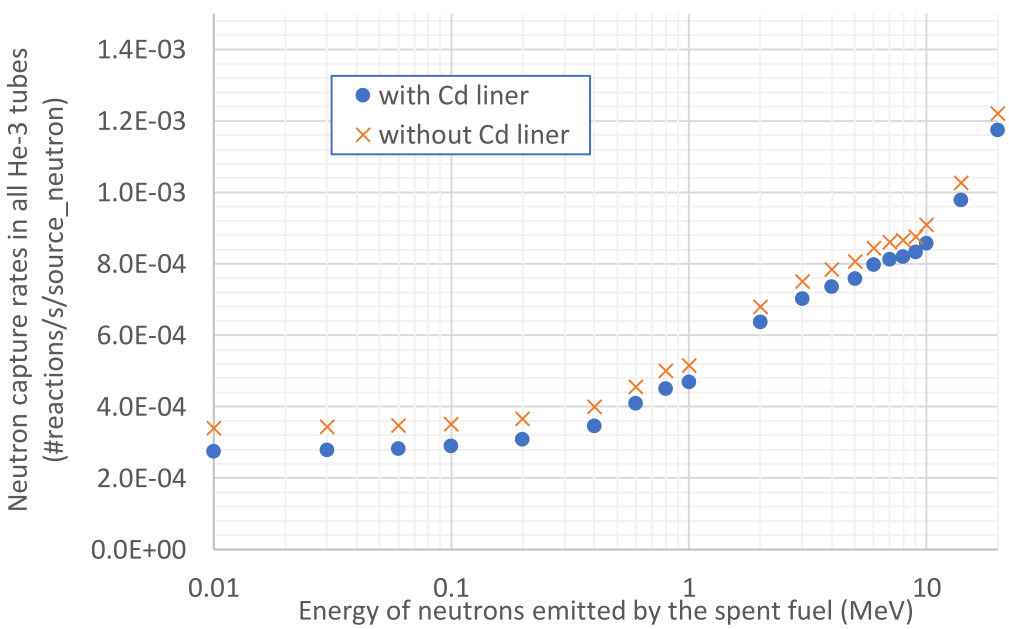

Figure 2. Response functions for the PNAR neutron count rates for the SVEA64 assembly design.

4. CALCULATION OF NET NEUTRON MULTIPLICATION

The PNAR ratio is expected to correlate with the net neutron multiplication, *Mult*. Therefore, the ORIGEN Module must calculate the *Mult* of a given spent fuel assembly. To determine the *Mult* for each assembly, the following equation was used:

$$Mult = 1/(1 - k_{eff}) \simeq 1/(1 - k_{\infty}\,(1 - L)), \tag{1}$$

where k_{eff} and k_{∞} are the effective and infinite neutron multiplication factors, respectively, of the assembly, and L is the neutron leakage factor. ORIGEN calculates k_{∞} as the ratio of total neutron production (i.e., fission cross section multiplied by the number of neutrons per fission) over total neutron absorption. No geometry is associated with this parameter other than flux-weighting the neutron cross sections for the assembly design. L is calculated as $L = 1 - k_{eff} / k_{\infty}$, where k_{eff} is determined by MCNP for a given fuel assembly geometry and composition. The leakage factor is largely constant for a fixed PNAR measurement configuration and is only weakly dependent on the fuel composition. L was calculated to be 0.55 for the 8 × 8 SVEA assembly design and 0.56 for the 10 × 10 SVEA assembly design. For the remaining three 9 × 9 assemblies, L is assumed to be 0.56. The values stored in the detector response files are the neutron non-leakage probabilities $(1 - L)$.

5. CALCULATION RESULTS AND COMPARISON WITH MEASUREMENTS

The PNAR neutron and gamma signals were calculated via the ORIGEN Module by using safeguards data and operator data as inputs in the calculations. This section presents the results of PNAR measurement analyses obtained with each input dataset. The STUK-provided safeguards data included assembly type, assembly-average burnup, assembly-average initial enrichment, cooling time, and U and Pu mass at discharge from the reactor for each assembly [9]. The initial U mass in each assembly was inferred by using the ORIGEN burnup code based on the provided information for these assemblies.

Besides the initial U loading, initial enrichment, and cooling time for each assembly, the operator data also include node-by-node burnup and moderator densities along the height of each assembly, which were extracted from the CASMO/SIMULATE spent nuclear fuel output files that TVO provided. The burnup values and moderator densities of node 8 from the operator data were used in the ORIGEN calculations to reflect the assembly characteristics in proximity to the assembly axial measurement position. In this work, the main difference between using safeguards data and operator data in the ORIGEN Module calculations is that the former uses assembly-average burnup and moderator density (0.4555 g/cm^3) values, and the latter uses nodal burnup and moderator density values near the measurement height. In both cases, the ORIGEN Module simulations assumed a continuous cycle history and a constant specific irradiation power of 24 MW/MTU, which is a representative power level for BWR assemblies. This simplification was necessary because the detailed cycle history is not usually included in safeguards data.

Once the ORIGEN Module calculated the raw PNAR neutron and gamma count rates, the neutron count rates were multiplied by *Mult* to account for the induced fissions in each assembly, and the gamma count rates were adjusted to account for the nonlinear response to gamma dose rate, which was previously found with LND 52110 ion chambers [7]. Then, the averages of the calculated and measured neutron and gamma count rates among the set of 23 BWR fuel assemblies were compared, and the ratios of the measured average signals to the calculated averages, referred to as *scaling factors*, were taken. One scaling factor common among all 23 assemblies was used to scale the calculated neutron count rates of each assembly, and another common scaling factor was used to scale the calculated gamma count rates for all 23 assemblies. These two scaling factors were used to account for factors (e.g., electronic efficiency) that were not accounted for in the ORIGEN Module calculations. Calibration factors can replace these scaling factors if sufficient PNAR measurement data were collected by the same PNAR instrument at the same spent fuel pool.

Figure 3 (left) shows the relative differences between the measured PNAR neutron count rates and the calculated values from the ORIGEN Module by using safeguards data for each assembly. Figure 3 (right) shows a similar comparison but with the operator data used in the calculations. As shown in Figure 3 (left), the standard deviation among assemblies is 11.7% for the neutron count rates without the Cd liner and 11.3% for the rates with the Cd liner. Sixteen of the 23 assemblies have calculated neutron count rates within 10% of the measurement, and the other seven assemblies (BWR1, 2, 5, 23, 31, 35, and 43) have calculation-to-measurement relative differences between -22.9 and 22.4%. BWR1 and BWR2 are two of the three assemblies with noncontinuous cycle histories. These results are similar to the results from previous Fork detector studies in which typical safeguards data were used in the ORIGEN Module calculations for BWR assemblies [7], primarily because of BWR assemblies' complexities in the fuel design and operating conditions. As shown in Figure 3 (right), the calculated results from using operator data were significantly improved compared with the measurements than from using safeguards data. The standard deviations among the assemblies were 3.9 and 4.4% for the neutron count rates with and without the Cd liner, respectively. All assemblies' calculated neutron count rates are within 10% of the measurements, and 18 assemblies show differences in the calculation-to-measurement of less than 5%.

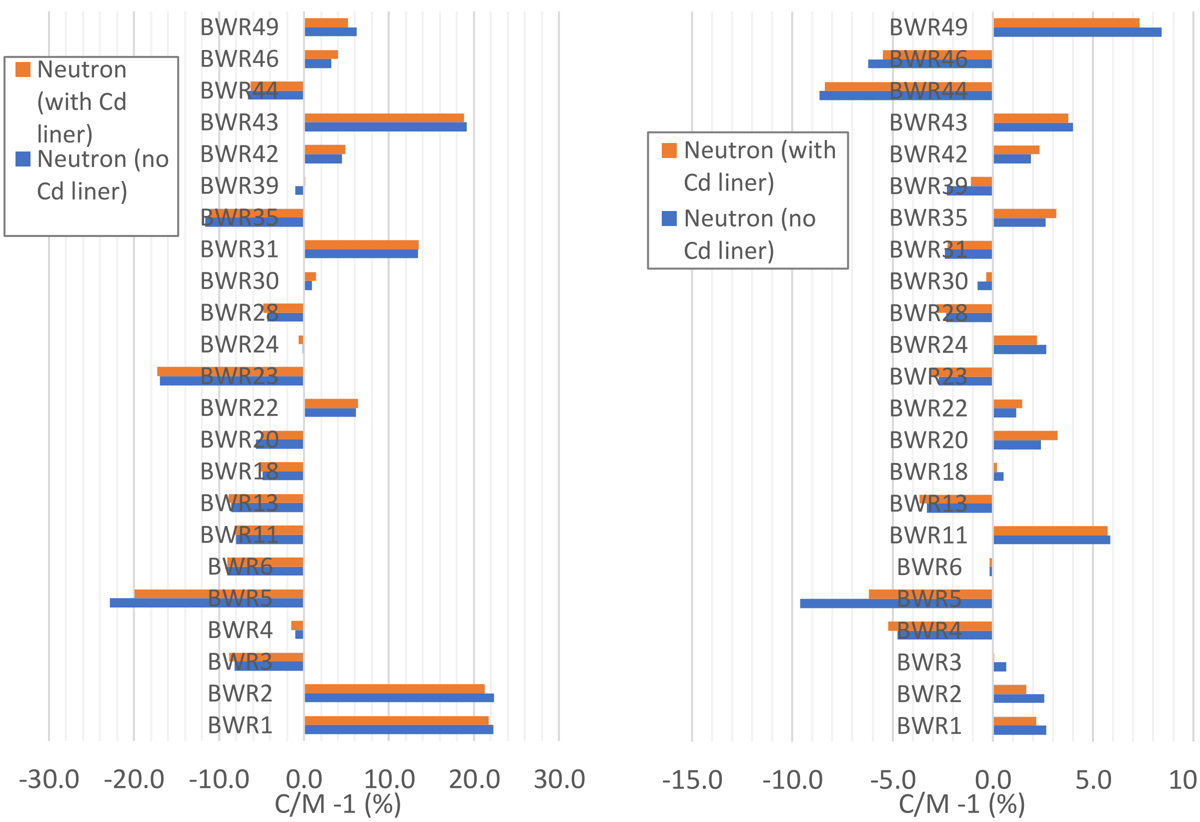

Figure 3. Neutron count rate relative differences between PNAR measurement, M, and calculation, C, using (left) safeguards data and (right) operator data for each assembly.

Figure 4 (left) shows the relative differences between the measured PNAR gamma count rates and the calculated values from the ORIGEN Module by using safeguards data for each assembly. Figure 4 (right) shows a similar comparison but with the operator data used in the calculations. As shown in Figure 4 (left), the standard deviation among the assemblies is 11.8 and 11.9% for the cases with and without the Cd liner, respectively. These differences would have been 16.3 and 15.4%, respectively, if adjustments were not made to account for the ion chamber nonlinear response. All assemblies, except BWR1 and BWR31, had calculated gamma count rates within 16% of the measured values. As shown in Figure 4 (right), the standard deviations among the assemblies were 10.9 and 11.0% for the cases with and without the Cd liner, respectively. All assemblies—except BWR1, BWR13, and BWR30—show calculated rates that are within 15% of the measured values. The gamma count rate results improved only marginally when operator data were used compared with the cases that used safeguards data, which is unsurprising because the gamma emissions are less sensitive to nodal burnup and moderator density conditions than neutron emissions. The gamma results are also similar to those observed in the ORIGEN Module analysis of Fork detector measurements for BWR assemblies [7].

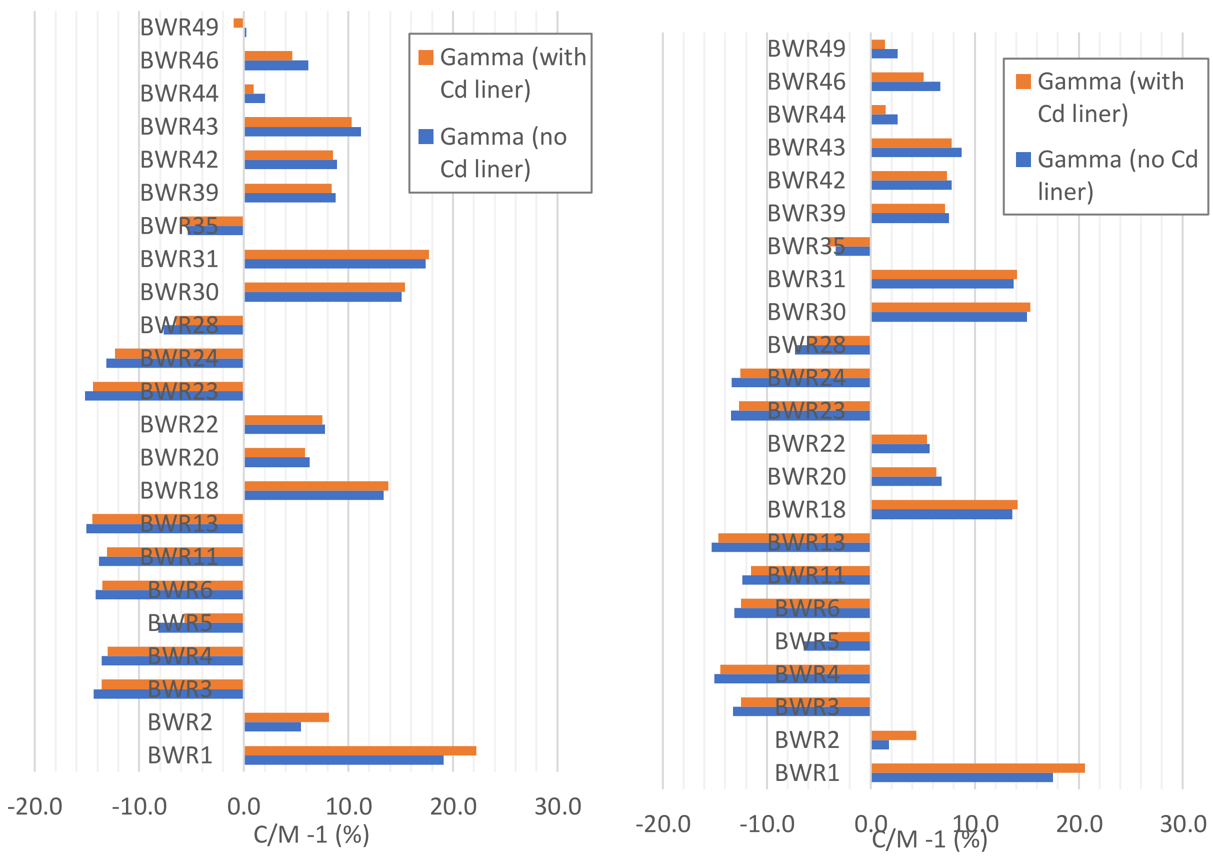

Figure 4. Gamma count rate relative differences between PNAR measurement, M, and calculation, C, using (left) safeguards data and (right) operator data for each assembly.

Figure 5 compares the measured PNAR ratio with the calculated *Mult* of each assembly with the safeguards data used in the calculations. The linear-fit trendline equation and the goodness of fit parameter (R^2) are also shown. The error bars (horizontal direction) on the measured PNAR ratios include the counting statistical uncertainty of the measurement and a repeatability uncertainty of 0.0013 [1]. The error bars for calculations are estimated as a sum of 2.7% uncertainty due to (a) the variability in the assembly-to-assembly burnup profile at the measurement position and enrichment variability at the measurement position to that of the assembly average, and (b) a 2% uncertainty associated with the calculation of *Mult* [9]. No uncertainty was assigned for cooling time or U mass. Hu, Ilas, and Gauld [9] provides a detailed uncertainty analysis. As shown, the calculated *Mult* generally follows a linear trend with the measured PNAR ratio, confirming that there is an underlying correlation between the two quantities, as a previous study suggested [3]. Most data points fall in a narrow range along the trendline that is overlapped with the uncertainty band. BWR1 was irradiated in noncontinuous cycles and was also an initial core assembly. In a previous study that used a Fork detector [7], first cycle assemblies exhibited a systematic bias compared with other assemblies, likely because of unusual reactor operations of the start-up core. These reasons might explain why BWR1 deviated the most from the trendline. BWR5 has the lowest ratio of burnup to initial enrichment values based on safeguards and operator data, indicating that it is the least "burned" assembly, so BWR5 is expected to have the highest ratio of fissile content to fission product absorbers remaining in the assembly. This explains why BWR5 had the highest *Mult* and PNAR ratio. On the contrary, BWR49 has the highest ratio of burnup to initial enrichment values, indicating that it is the most "burned" assembly, which explains why BWR49 had the lowest *Mult* and PNAR ratio. A similar trend was found between measured PNAR ratio and calculated *Mult* with operator data used in the calculations, and the results are discussed in Hu, Ilas, and Gauld [9].

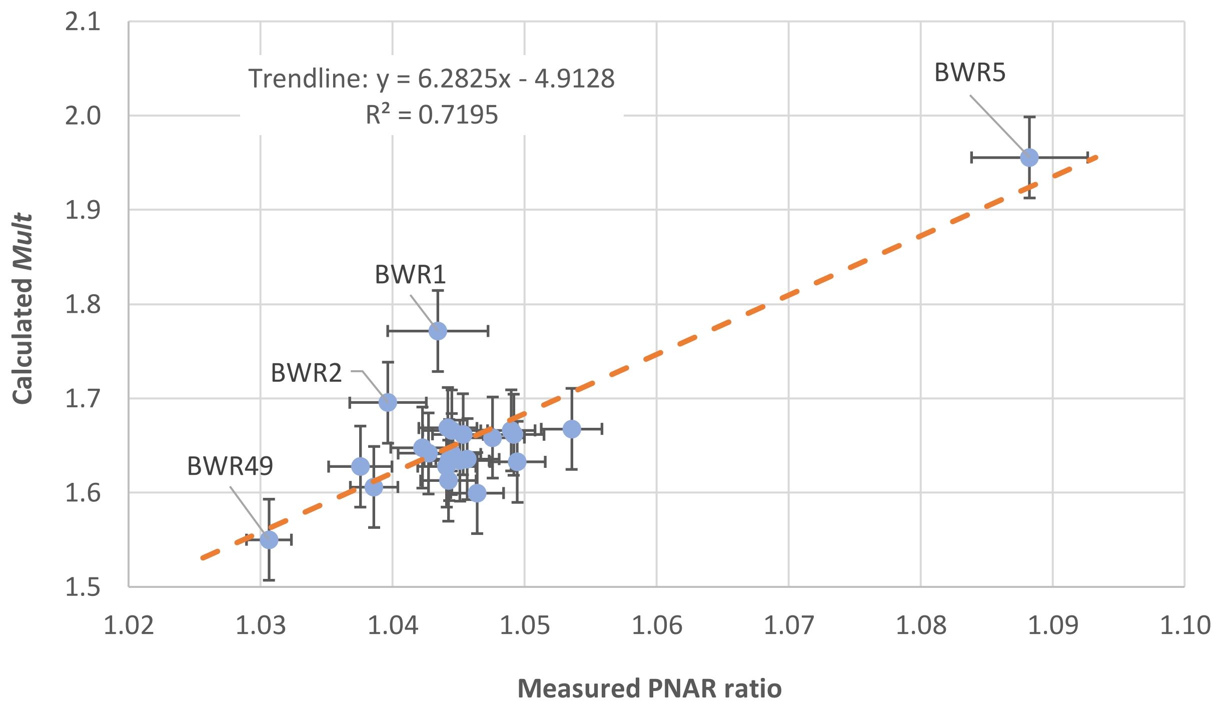

Figure 5. Calculated *Mult* using safeguards data vs. measured PNAR ratio.

6. SUMMARY AND CONCLUSIONS

The PNAR instrument was designed to measure the neutron multiplication of a fuel assembly to ensure that the assembly multiplies at a level consistent with the operator declaration and the expected fissile content in the assembly. The ORIGEN Module calculation results show that when typical safeguards data—typically including assembly-average attributes only—were used, the standard deviations between measured and calculated neutron and gamma detector count rates were similar to the results from previous studies that used the Fork detector because of complexities in BWR assemblies. The agreement between the calculated count rates and the measurement values was improved by the added nodal burnup and moderator density values included in the operator data, which resulted in a standard deviation of the calculated-to-measured value among the set of 23 assemblies of ~4% for neutron and 10% for gamma count rates.

The measured PNAR ratio was analyzed by trending the PNAR ratio with the calculated *Mult* of the assembly because a correlation between them was expected, but the exact correlation was not established before this work. The results show a clear linear trend between the measured PNAR ratio and calculated *Mult*. Most data points fall within a narrow range along the trendline that is overlapped with the uncertainty band. The predicted assemblies with the highest and lowest *Mult* calculated by the ORIGEN Module were the same as the assemblies with the highest and lowest measured PNAR ratios.

Based on comparisons between calculations and measurements, the results show that the ORIGEN Module can reasonably predict the PNAR neutron and gamma signals and *Mult*, which follows a linear trend with the measured PNAR ratio. Because an ORIGEN Module calculation takes only seconds to complete, safeguards inspectors can use these predictions to draw quick conclusions. Future work is recommended to investigate how to further reduce discrepancies between the calculated and the measured quantities while considering the complexities in fuel design parameters and operator conditions (e.g., enrichment zoning, part length rods, noncontinuous cycles) that often exist in BWR assemblies.

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