A Complex-Geometry Validation Experiment for Advanced Neutron Transport Codes

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INTRODUCTION

The Idaho National Laboratory (INL) has initiated a focused effort to upgrade legacy computational reactor physics software tools and protocols used for support of core fuel management and experiment management in the Advanced Test Reactor (ATR) and its companion critical facility (ATRC) at the INL.. This will be accomplished through the introduction of modern high-fidelity computational software and protocols, with appropriate new Verification and Validation (V&V) protocols, over the next 12-18 months. Stochastic and deterministic transport theory based reactor physics codes and nuclear data packages that support this effort include MCNP5[1], SCALE/KENO6[2], HELIOS[3], SCALE/NEWT[2], and ATTILA[4]. Furthermore, a capability for sensitivity analysis and uncertainty quantification based on the TSUNAMI[5] system has also been implemented. Finally, we are also evaluating the Serpent[6] and MC21[7] codes, as additional verification tools in the near term as well as for possible applications to full three-dimensional Monte Carlo based fuel management modeling in the longer term.

On the experimental side, several new benchmark-quality code validation measurements based on neutron activation spectrometry have been conducted using the ATRC. Results for the first four experiments, focused on neutron spectrum measurements within the Northwest Large In-Pile Tube (NW LIPT) and in the core fuel elements surrounding the NW LIPT and the diametrically opposite Southeast IPT have been reported [8,9]. A fifth, very recent, experiment focused on detailed measurements of the element-to-element core power distribution is summarized here and examples of the use of the measured data for validation of corresponding MCNP5, HELIOS, NEWT, and Serpent computational models using modern least-square adjustment methods are provided.

FACILITY DESCRIPTION

The ATR (Fig. 1) is a light-water and beryllium moderated, beryllium reflected, light-water cooled system with 40 fully-enriched (93 wt% $^{235}U/U_{Total}$) plate-type fuel elements, each with 19 curved fuel plates separated by water channels. The fuel elements are arranged in a serpentine pattern as shown, creating 5 separate 8-element "lobes". Gross reactivity and power distribution control during operation are achieved through the use of rotating control drums with hafnium neutron absorber plates on one side. The ATR can operate at powers as high as 250 MW with corresponding thermal neutron fluxes in the flux traps that approach 5.0 x 10^{14} n/cm^2-s. Typical operating cycle lengths are in the range of 45-60 days.

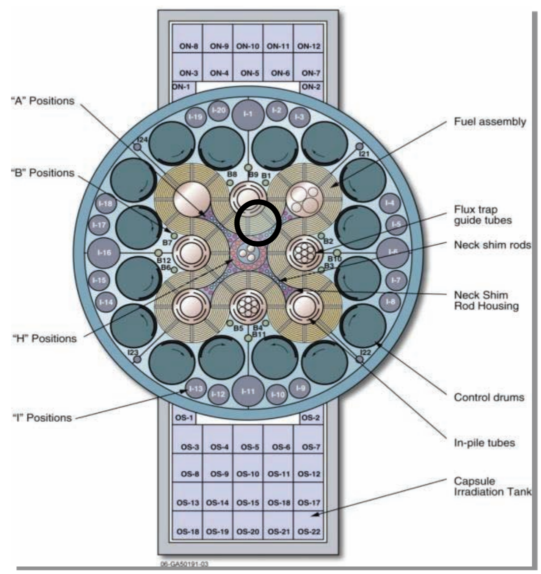

Fig. 1. Core and reflector geometry of the Advanced Test Reactor. References to core lobes and in-pile tubes are with respect to reactor north, at the top of the figure. Fuel element number 1 is circled, and the element numbering scheme proceeds clockwise around the serpentine core.

The ATRC is a nearly-identical open-pool nuclear mockup of the ATR that typically operates at powers in the range of several hundred watts. It is most often used with prototype experiments to characterize the expected changes in core reactivity and power distribution for the same experiments in the ATR itself. Useful physics data can also be obtained for evaluating the worth and calibration of control elements as well as thermal and fast neutron distributions.

VALIDATION EXPERIMENT DESCRIPTION

In the new validation experiment of interest here, activation measurements that can be related to the total fission power of each of the 40 ATRC fuel elements were made with fission wires composed of 10% by weight uranium (93% enriched in ^{235}U) in aluminum. The wires were 1 mm in diameter and approximately 0.635 cm (0.25") in length and were placed in various locations within the cooling channels of each fuel element at the core axial midplane as shown in Fig. 2. The total measured fission powers for the fuel elements are estimated using appropriately-weighted sums of the measured fission rates in the U/Al wires located in each element.

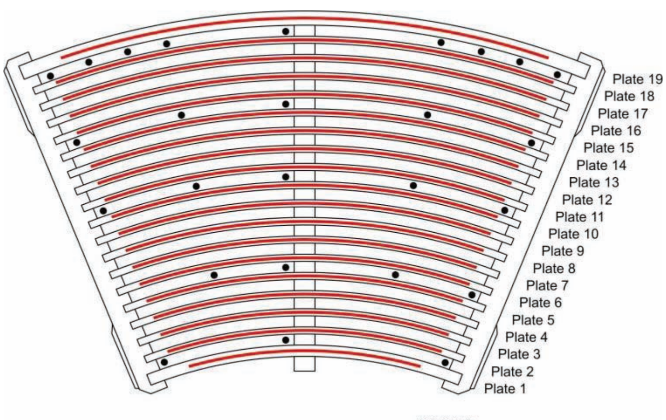

Fig. 2. ATR Fuel element geometry, showing standard fission wire positions used for intra-element power distribution measurements.

RESULTS

Fig. 3 shows the fission powers for the 40 ATRC fuel elements computed using MCNP5, along with the measured element powers based on the fission wire measurements. The top number (black) in the center of each element is the *a priori* element power (W) and the bottom number (red) is the measurement. Total measured power was 875.5 W. Uncertainties (1σ) associated with the measured element powers are estimated to be approximately 5% including correlated components arising primarily from the common calibration of the detector used to measure the fission rates in the wires plus uncorrelated components largely due to counting statistics.

HELIOS, NEWT and Serpent were also used to model this configuration of the ATRC in addition to MCNP5. All four sets of *a-priori* fuel element powers were then adjusted against the measurements using a covariance-weighted least-squares technique specifically adapted for this purpose [10]. The covariance matrix for the measured powers was constructed using the measurement uncertainty quoted above. The required covariance matrix for the *a priori* fuel element powers produced by each modeling code was constructed by normalizing the element-to-element fission correlation matrix to an estimated value of 10% (1σ) absolute *a priori* uncertainty for all fuel elements in each case, based on practical experience with ATRC modeling. The fission correlation matrix was in turn constructed using a novel method [10] based on the element-to-element fission matrix computed from an MCNP5 model for the ATRC configuration of interest.

Figures 4-7 show the results of the adjustments for each code. It can be seen in all cases that 68% of the adjustments fell well within the assumed *a priori* uncertainty (1σ=10%) and the maximum adjustments in both directions fell well within two *a priori* standard deviations, indicating possible degree of conservatism in the *a priori* uncertainties assumed for the computed element powers. The reduced uncertainties for the adjusted element powers were approximately 3.5% (1σ), consistent with the estimated 5% (1σ) uncertainties of the measured powers used for the adjustment. As expected, the adjustments associated with the 3D codes (MCNP5, Serpent) were generally smaller than was the case for the 2D (NEWT, HELIOS) codes, reflecting somewhat greater *a priori* consistency of the 3D results with the measurements.

DISCUSSION AND CONCLUSIONS

A relatively simple but effective fission-matrix-based method [10] for detailed statistical validation and best-estimate adjustment analysis of fission power distributions produced by computational reactor physics models of the ATR has been successfully applied to four different, independent models of the same high-fidelity ATRC benchmark experiment. Analyses of this type are useful not only for quantifying the bias and uncertainty of computational models for a specific measured reactor configuration of interest, but they also can serve as guides for model improvement and for estimation of *a priori* modeling uncertainties for related reactor configurations for which no measurements are available.

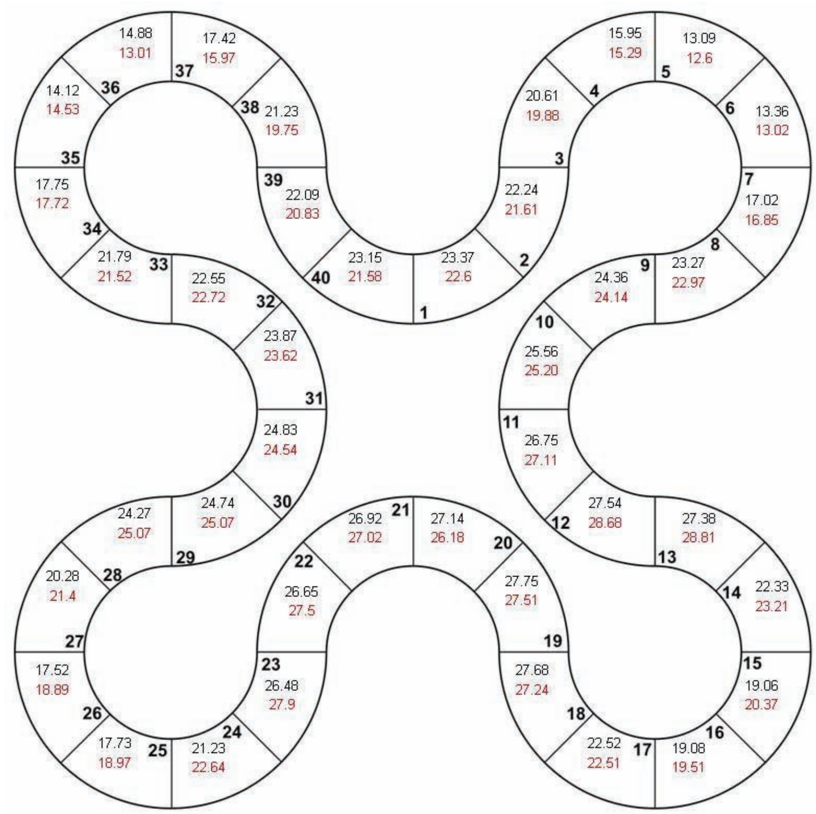

Fig. 3. Calculated (black) and measured (red) fuel element powers (W). The fuel element numbers are in bold type.

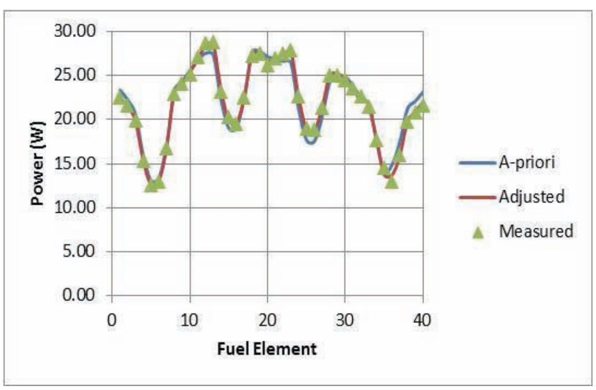

Fig. 4. Fuel element fission rate distribution for ATRC Depressurized Run Support Test 12-5 (MCNP *a priori).* The *a priori* uncertainty was assumed to be 10% (1σ). The adjustments ranged from -9.8% in Element 37 to +6.8% in Element 25 and 68% of the adjustments were within ±4%. Reduced uncertainties for the adjusted powers are in the range of 3.1% to 3.7% for all elements.

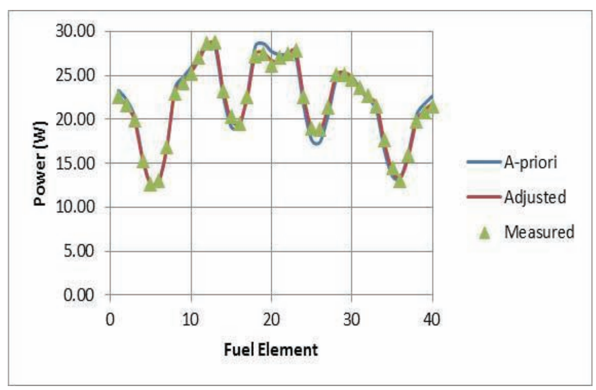

Fig. 5. Fuel element fission rate distribution for ATRC Depressurized Run Support Test 12-5 Serpent *a priori).* The *a priori* uncertainty was assumed to be 10% (1σ). The adjustments ranged from -3.7% in Element 20 to +7.1% in Element 26 and 68% of the adjustments were within ±3.2%. Reduced uncertainties for the adjusted powers are in the range of 3.2% to 3.6% for all elements.

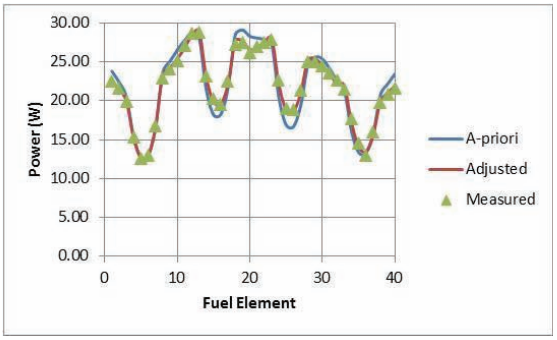

Fig. 6. Fuel element fission rate distribution for ATRC Depressurized Run Support Test 12-5 NEWT *a priori).* The *a priori* uncertainty was assumed to be 10% (1σ). The adjustments ranged from -6.4% in Element 40 to +11.8% in Element 26 and 68% of the adjustments were within ±5.1%. Reduced uncertainties for the adjusted powers are in the range of 3.2% to 3.6% for all elements.

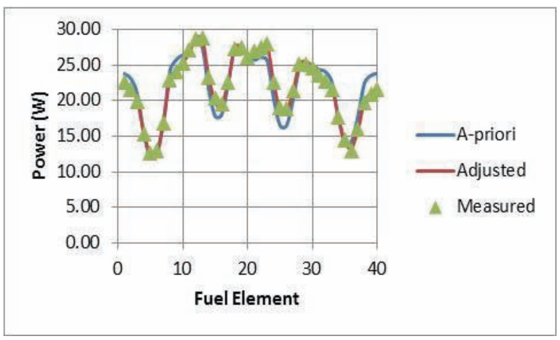

Fig. 7. Fuel element fission rate distribution for ATRC Depressurized Run Support Test 12-5 (HELIOS *a priori).* The *a priori* uncertainty was assumed to be 10% (1σ). The adjustments ranged from -11.5% in Element 38 to +13.2% in Element 25 and 68% of the adjustments were within ±6.3%. Reduced uncertainties for the adjusted powers are in the range of 3.1% to 3.7% for all elements.

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