Nuclear Engineering and Design 295 (2015) 615–624

Contents lists available at ScienceDirect

Nuclear Engineering and Design

journal homepage: www.elsevier.com/locate/nucengdes

A fission matrix based validation protocol for computed power distributions in the advanced test reactor

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ARTICLE INFO

Article history:
Received 1 July 2015
Accepted 30 July 2015
Available online 30 October 2015

ABSTRACT

The Idaho National Laboratory (INL) has been engaged in a significant multiyear effort to modernize the computational reactor physics tools and validation procedures used to support operations of the Advanced Test Reactor (ATR) and its companion critical facility (ATRC). Several new protocols for validation of computed neutron flux distributions and spectra as well as for validation of computed fission power distributions, based on new experiments and well-recognized least-squares statistical analysis techniques, have been under development. In the case of power distributions, estimates of the *a priori* ATR-specific fuel element-to-element fission power correlation and covariance matrices are required for validation analysis. A practical method for generating these matrices using the element-to-element fission matrix is presented, along with a high-order scheme for estimating the underlying fission matrix itself. The proposed methodology is illustrated using the MCNP5 neutron transport code for the required neutronics calculations. The general approach is readily adaptable for implementation using any multi-dimensional stochastic or deterministic transport code that offers the required level of spatial, angular, and energy resolution in the computed solution for the neutron flux and fission source.

1. 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 Advanced Test Reactor (ATR) core fuel management, experiment management, and safety analysis. This is being accomplished through the introduction of modern high-fidelity computational software and protocols, with appropriate verification and validation (V&V) according to applicable national standards. A suite of well-recognized stochastic and deterministic transport theory based reactor physics codes and their supporting nuclear data libraries (HELIOS (Studsvik Scandpower, 2008), NEWT (DeHart, 2006), ATTILA (McGhee et al., 2006), KENO6 (Hollenbach et al., 1996) and MCNP5 (Goorley et al., 2004)) is in place at the INL for this purpose, and corresponding baseline models of the ATR and its companion critical facility (ATRC) are operational. Furthermore, a capability for rigorous sensitivity analysis and uncertainty quantification based on the TSUNAMI (Broadhead et al., 2004) system has been implemented and initial computational results have been obtained. Finally, we are also incorporating the MC21 (Sutton et al., 2007) and SERPENT (Leppänen, 2012) stochastic simulation and depletion codes into the new suite as additional tools for V&V in the near term and possibly as advanced platforms for full 3-dimensional Monte Carlo based fuel cycle analysis and fuel management in the longer term.

On the experimental side of the effort, several new benchmark-quality code validation measurements based on neutron activation spectrometry have been conducted at the ATRC. Results for the first three experiments, focused on detailed neutron spectrum measurements within the Northwest Large In-Pile Tube (NW LIPT) were recently reported (Nigg et al., 2012a) as were some selected results for the fourth experiment, featuring neutron flux spectra within the core fuel elements surrounding the NW LIPT and the diametrically opposite Southeast IPT (Nigg et al., 2012b). In the current paper we focus on computation and validation of the fuel element-to-element power distribution in the ATRC (and by extension the ATR) using data from an additional, recently completed, ATRC experiment. In particular we present a method developed for estimating the covariance matrix for the fission power distribution using the corresponding fission matrix computed for the experimental configuration of interest. This covariance matrix is a key input parameter that is required for the least-squares adjustment validation methodology employed for assessment of the bias and uncertainty of the various modeling codes and techniques.

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http://dx.doi.org/10.1016/j.nucengdes.2015.07.049

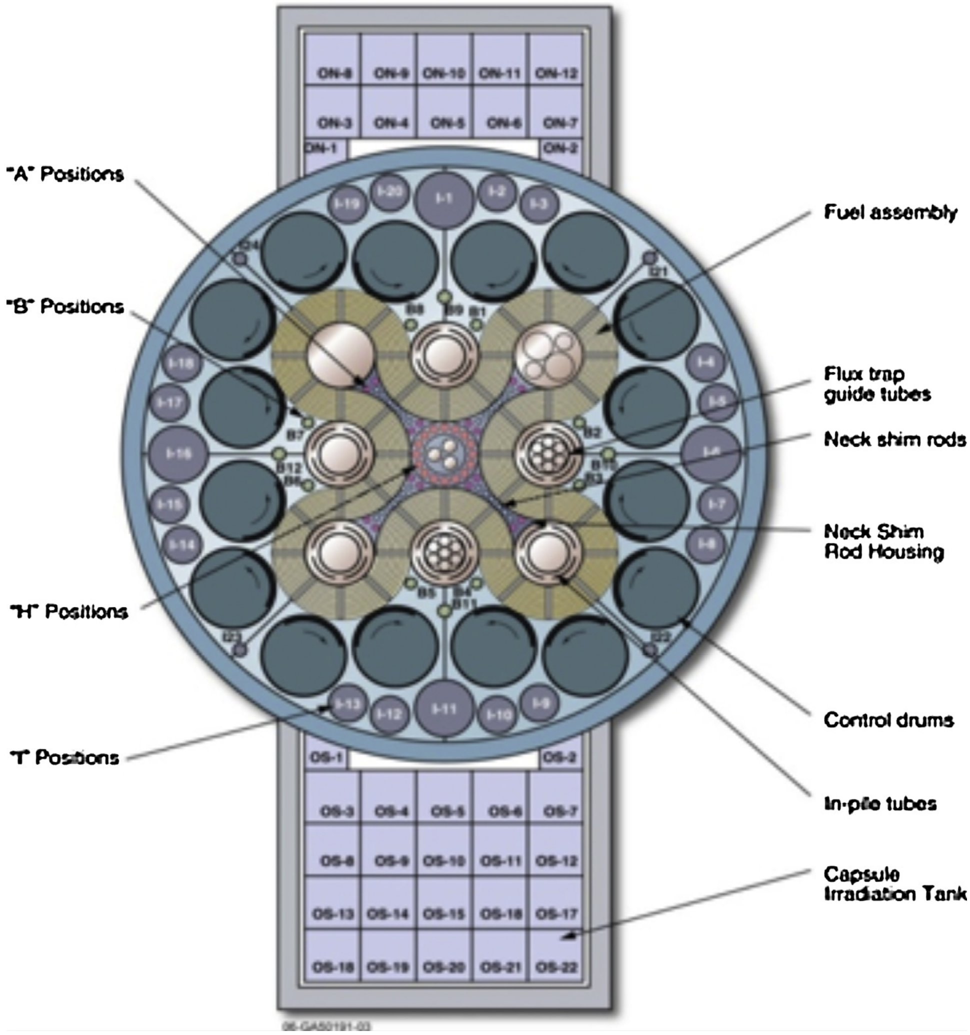

Fig. 1. Core and reflector geometry of the Advanced Test Reactor. References to core lobes and in-pile tubes are with respect to reactor north, at the top of the figure.

2. 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 five 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 5.0×10^{14} N/cm^2 s. Typical operating cycle lengths are in the range of 45–60 days.

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.

3. Computational methods and models

Computational reactor physics modeling is used extensively to support ATR experiment design, operations and fuel cycle management, core and experiment safety analysis, and many other applications. Experiment design and analysis for the ATR has been supported for a number of years by very detailed and sophisticated three-dimensional Monte Carlo analysis, typically using the MCNP5 code, coupled to extensive fuel isotope buildup and depletion analysis where appropriate. On the other hand, the computational reactor physics software tools and protocols currently used for ATR core fuel cycle analysis and operational support are largely based on four-group diffusion theory in Cartesian geometry (Pfeifer, 1971) with heavy reliance on "tuned" nuclear parameter input data. The latter approach is no longer consistent with the state of modern nuclear engineering practice, having been superseded in the general reactor physics community by high-fidelity multidimensional transport-theory-based methods. Furthermore, some aspects of the legacy ATR core analysis process are highly empirical in nature, with many "correction factors" and approximations that require very specialized experience to apply. But the staff knowledge from the 1960s and 1970s that is essential for the successful application of these various approximations and outdated computational processes is rapidly being depleted due to personnel turnover and retirements.

Fig. 2 shows the suite of new tools mentioned earlier, how they generally relate to one another, and how they will be applied to ATR. This illustration is not a computational flow chart or procedure *per se*. Specific computational protocols using the tools shown in Fig. 2 for routine ATR support applications will be promulgated