# Numerical modelling of control rod calibrations and fuel depletion at the OPAL research reactor

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**Abstract.** The IAEA is currently administering an international Coordinated Research Project (CRP), the main purpose of which is to develop a set of research reactor benchmarks for the verification and validation of computational codes. The focus of the CRP in particular is the modelling of multi-cycle depletion. Necsa is developing a new calculational framework for performing nuclear reactor core calculations, which integrates both the stochastic and deterministic modeling methods in a consistent manner. In this work, the system is applied to the OPAL benchmark problem. The OPAL reactor is a modern research reactor with challenging aspects in neutronic design. In particular, the use of burnable poisons and a heavy water reflector pose modelling challenges. Analysis conducted on this benchmark includes control rod calibration experiments as well as the simulation of four actual operating cycles.

## 1. Introduction

Numerical modelling is often employed to support the safe and economic operation of nuclear reactors. Computer codes are used to calculate, amongst other things, the distribution of neutrons in space and time by solving the Boltzmann transport equation, or some approximation of it. Operational parameters such as power distribution, control rod position and fuel depletion can then be calculated using this information.

The International Atomic Energy Agency (IAEA) is conducting a CRP titled “Benchmarks of Computational Tools against Experimental Data on Fuel Burnup and Material Activation for Utilization, Operation and Safety Analysis of Research Reactors”. Various institutions have submitted specifications of the research reactors they operate, as well as descriptions of experiments that were performed with the reactors and data gathered in the course of normal operations, to provide benchmarks against which reactor computational codes may be verified and validated [1].

Necsa developed a tool for creating detailed heterogeneous three dimensional models, which can be deployed to generate input for various reactor calculation codes such as Serpent 2 [2, 3], MCNP [4] and the OSCAR-4 [5] nodal diffusion solver, while maintaining the consistency of the model across these codes. In this work, the tool was used to prepare detailed models for the OPAL benchmark problem and generate input for Serpent, a criticality and burn-up code that employs a Monte Carlo solution method [3]. In this way Serpent and the unified model were validated against experimental data before they can be used to support operations.

## 2. Problem Description

The Open Pool Australian Lightwater (OPAL) reactor is a 20 MW research reactor that is operated by ANSTO (Australian Nuclear Science and Technology Organisation). It is heavy water reflected and light water moderated. A schematic representation of the model of the reactor, including the facilities around the core, is shown in Figure 1. There are a number of irradiation facilities, located mainly to the east and south of the core, for the production of radio isotopes, neutron activation analysis and silicon transmutation doping. A cold neutron source, which uses liquid helium to moderate neutrons to very low energies, is located to the north. There are five beam tubes that provide neutrons to various instruments, of which two are for thermal neutrons, two for cold neutrons, and one for hot neutrons.

The reactor consists of a compact core with 16 fuel assemblies, four control rods and a central, cruciform, regulating rod, which is used for fine-grained control of the critical state of the reactor as shown in Figure 2. It is divided into four quadrants by the control rods, with the regulating rod in the centre. The fuel is MTR (Material Testing Reactor) type assemblies, with 21 plates each. Three types of assemblies are used for the first cycle and are designated as type 1 fuel, type 2 fuel and standard fuel with 212 g U-235, 383 g U-235 and 484 g U-235 respectively. Only the standard fuel is used for reloading [6]. There are also burnable poisons in the form of cadmium wires slotted into every second fuel plate of the type 2 and standard fuel. Burnable poisons remove neutrons to limit the fission rate when fuel is fresh, and are removed by transmutation over time.

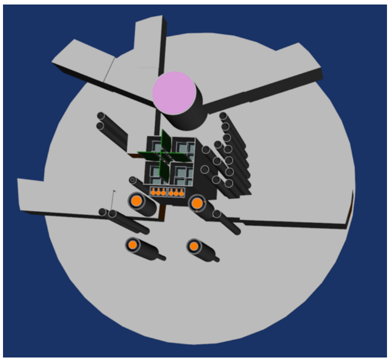

**Figure 1:** Schematic representation of the OPAL model and the reactor components

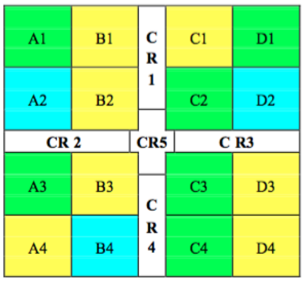


**Figure 2:** The numbering of control rods and fuel elements within the reactor core. Type 1 fuel is yellow, type 2 green and standard fuel is blue

The reactor operator submitted results from control rod calibration experiments that were performed at the start-up of the initial core, as well as data on the core state for the first 7 operating cycles [6].

Control rod calibrations start from a core that is critical, i.e. one in which the neutron production rate is balanced by the rate at which neutrons are lost due to capture or leakage from the system. They then proceed as follows: the core is made super-critical (neutron population is increasing) by withdrawing the control rod to be calibrated by a certain distance, and the resultant reactivity insertion is measured. One or more rods are then inserted to restore criticality. This process is repeated until the calibrated rod is fully extracted from the core. The reactivity change per unit movement of a rod is called the differential worth of the rod, and is dependent on insertion depth. The experiments for which data was made available are for control rod 1 compensated by rod 4, rod 2 compensated by rod 3, and rod 5 compensated by rod 2 and rod 3. The location of the rods within the core is demonstrated Figure 2.

The composition of fissionable materials within the core changes over time during operation, a process that is referred to as burn-up or depletion. No direct measurements of burn-up were provided, but data such as the reactor power and control rod positions at various points during reactor operation was made available. In this benchmark, cycles were between 30 and 40 days long and the core was reloaded with 3 fresh standard fuel assemblies after each cycle. Data was provided for the first 7 cycles of the reactor's operating life.

## 3. Calculational approach adopted and model description

The new pre and post processor tool was used to create heterogeneous model of the core and surrounding structures using the OPAL benchmark specification document, with as much detail as reasonably achievable. The specifications of certain structures were simplified or lacked full engineering detail. This was especially true for the cold neutron source and many of the irradiation facilities that surround the core.

The control rod calibration experiments were modelled using a modified version of Serpent 2.1.23 [3] and compared to the measured data. Additionally, core follow analysis was performed with Serpent for the first 4 reactor cycles to track the evolution of burnable elements in the core. It is known that a reactor is critical under normal operating conditions. Therefore, in the absence of depletion measurements, the deviation from criticality predicted by the model during operation may be used as a proxy for performance in tracking fuel depletion.

## 4. Results and Discussion

### *4.1. Control rod calibration*

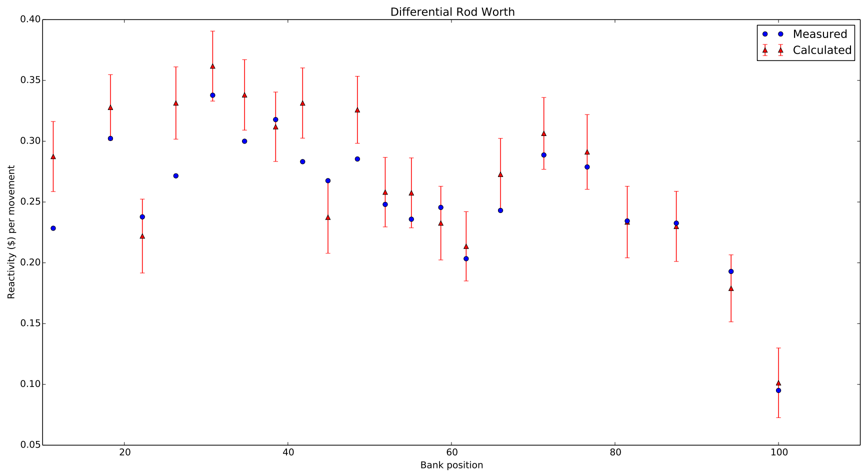


**Figure 3:** A comparison between the calculated and measured differential rod worth for control rod 1 compensated by rod 4

Figure 3 shows the calculated and measured differential rod worth curve for rod 1, which is measured in reactivity ($) inserted per unit bank movement (% of total rod travel distance), with rod 4 as the compensating rod. It is clear from Figure 3 that the model slightly overestimated the rod worth, especially in the lower and central parts of the core.

Figure 4 show the differential rod worth curves for the calibration of rod 2, which was compensated by rod 3. For these rods the model predicted the measured values very well.

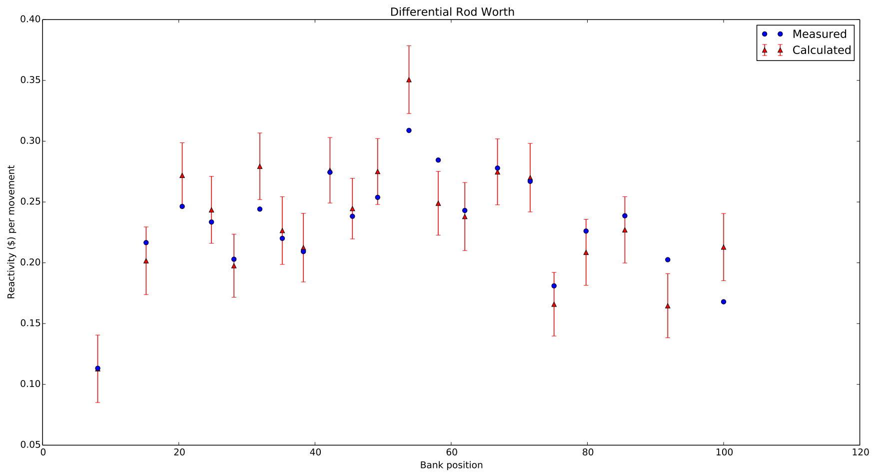


**Figure 4:** A comparison between the calculated and measured differential rod worth for control rod 2 compensated by rod 3

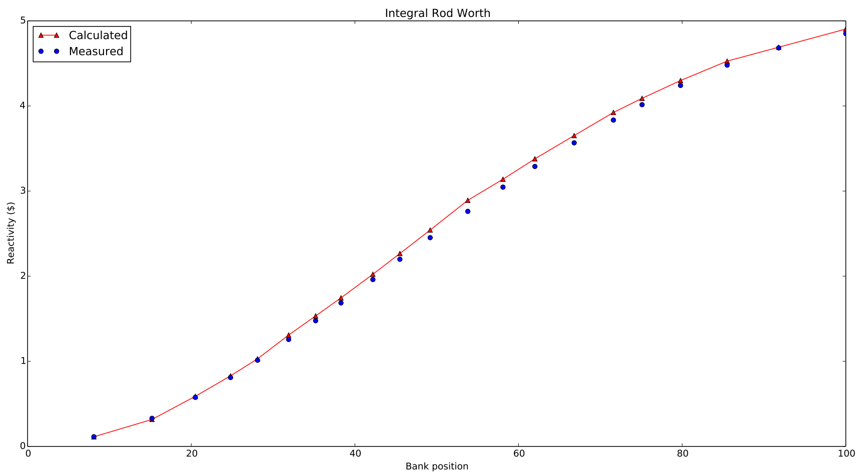


**Figure 5:** A comparison between the calculated and measured integral rod worth curves for control rod 2 compensated by rod 3

One can observe just how well the model performs by looking at the integral rod worth curve in Figure 5, which shows the cumulative worth of the rod over its total travel distance.

Figure 6 is the differential rod worth curve for rod 5, the regulating rod. It must be noted that the reactivity that was inserted by rod 5 was compensated for by both rod 2 and rod 3. The model again slightly overestimated the worth of this rod.

In general, good agreement between the measured control rod worth and the calculated control rod worth is observed. Since Serpent uses a Monte Carlo algorithm, it estimates the distribution of fissions in the system by continually re-sampling the distribution from an initial guess until balance is achieved. This creates a correlation between the critical estimates at

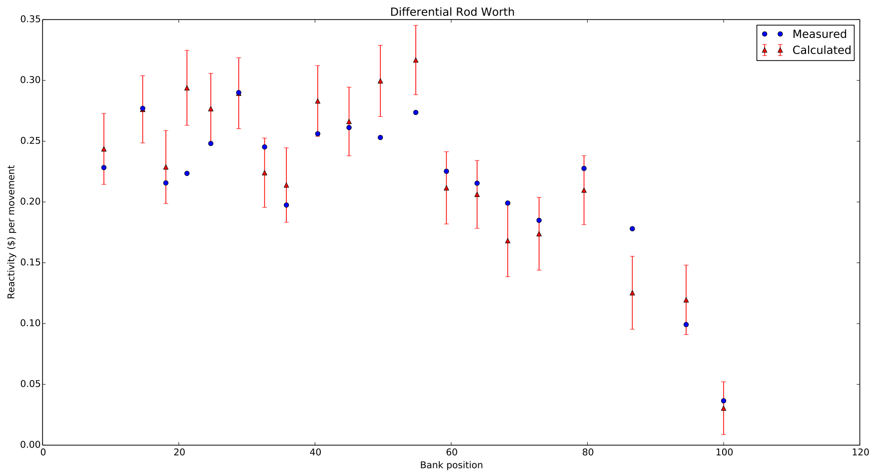


**Figure 6:** A comparison between the calculated and measured differential rod worth for control rod 5 compensated by rod 2 and rod 3

different rod positions, which will increase the variance, and hence the uncertainties on the estimates. That is, the displayed error bars should be larger. Currently, there is no built in measure of this additional source of uncertainty in Serpent. The effect would be particularly noticeable when a rod movement causes a large change in the fission distribution, either radially or axially. This, at least partly explains why the Serpent results seem more sensitive to large reactivity changes.

*4.2. Core follow analysis*

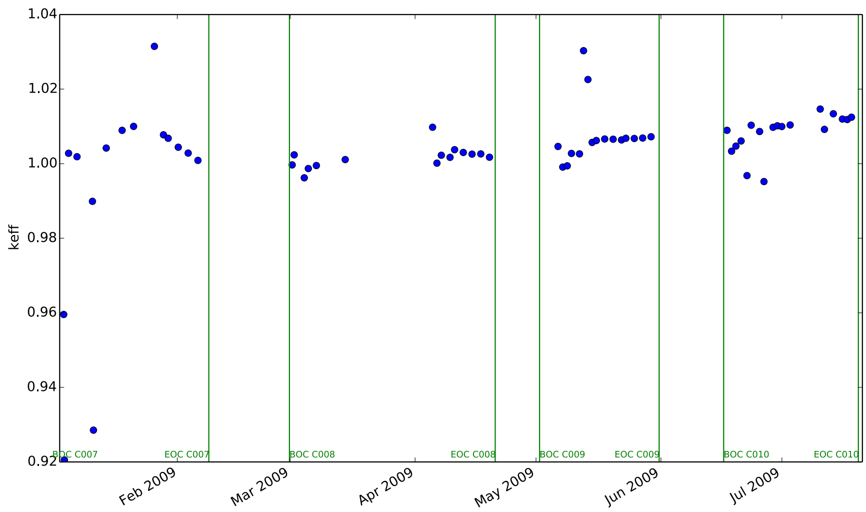


**Figure 7:** Calculated reactor $k_{\mathrm{eff}}$ for the first four cycles

Figure 7 shows the results of the core follow analysis for the first 4 of the 7 cycles for which data was provided. A reactor in steady state operation is critical, with an effective neutron multiplication factor of $k_{\text{eff}} = 1$, which is the ratio between the number of neutrons produced by fission and the number of neutrons lost through absorption and leakage. In the absence of direct fuel depletion measurements, calculating $k_{\text{eff}}$ with given operational data, such as control rod positions and core power, over the course of reactor operations gives a reasonable indication of the accuracy with which the depletion of fuel is modelled.

Overall the core follow calculation performed well, although there are some outliers in the data, which may be attributed to several shut-downs that the reactor experienced in the first few cycles for which data was not provided. Overall, however, the calculated $k_{\text{eff}}$ is close to 1, although it increases gradually from cycle to cycle. There are a number of factors that can contribute to the rise in reactivity: The first is in the modelling of the burnable absorbers (cadmium wires) in the fuel assemblies. Since both the volume and surface of absorbing materials are important, when depleting the wires, absorbing materials near the surface of the wires should not be removed too quickly (the so called rim effect). Constraints on computing resources limit the number of depletion zones (rings) that can be used in the wires, and for this study only one inner and outer ring were employed. This can be refined in order to check if reactivity estimates improve.

The second factor contributing to a systematic reactivity increase is uncertainty in the provided core power levels. Reactor facilities have indirect measures of estimating core power, which must be carefully calibrated, since inlet and outlet temperature measurements can not directly account for loss of heat to the reflector pool and other structures. If the measured power is too low, fuel will be under depleted, which will cause reactivity to steadily rise until a new equilibrium mass distribution is reached. This estimated distribution will differ slightly from the real mass distribution in the system. Such a power sensitivity study is planned in future work.

## 5. Conclusion

The calculated control rod calibration results compared well with the measured results although there were some slight discrepancies. The discrepancies could be attributed to the constraints within the structure of Serpent especially in areas where the rod movement would cause a large change in the fission distribution. In the corefollow calculation the criticality estimates increased gradually from cycle to cycle. The gradual increase could be as a result of the uncertainties in the provided core power levels or the over burning of the burnable poisons.

## Acknowledgements

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