OPAL REACTOR: CALCULATION/EXPERIMENT COMPARISON OF NEUTRON FLUX MAPPING IN FUEL COOLANT CHANNELS

L. BARBOT*, C. DOMERGUE, J-F. VILLARD AND C. DESTOUCHES
CEA, DEN, DER, Instrumentation, Sensors and Dosimetry Laboratory, Cadarache, F-13108 St-Paul-Lez-Durance, France

G. BRAOUDAKIS, D. WASSINK, B. SINCLAIR, J. C. OSBORN AND H. WU
ANSTO, Locked Bag 2001, Kirrawee DC NSW 2232, Australia

*Corresponding author: loic.barbot@cea.fr

ABSTRACT

The measurement and calculation of the neutron flux mapping of the OPAL research reactor are presented. Following an investigation of fuel coolant channels using sub-miniature fission chambers to measure thermal neutron flux profiles, neutronic calculations were performed. Comparison between calculation and measurement shows very good agreement.

1. Introduction

Within the framework of a scientific and technological cooperation agreement between the French Alternative Energies and Atomic Energy Commission (CEA) and the Australian Nuclear Science and Technology Organisation (ANSTO), an on-line neutron flux profile in fuel coolant channels measurement campaign took place in the OPAL research reactor in December 2012.

The Open Pool Australian Light-water reactor (OPAL), which first achieved criticality in 2006, is a 20 MW thermal power research reactor using low enriched uranium fuel plates. The core contains 16 fuel elements and is located at the bottom of a 13 metre deep light water pool. Reactivity control is provided by five hafnium control rods. OPAL reactor activities are focused on neutron beams research thanks to the D_2O reflector surrounding the core, medical radioisotope production, silicon doping and neutron activation analysis.

The campaign main objective was the on-line relative thermal neutron flux mapping of OPAL fuel assemblies, using sub-miniature fission chambers. Neutron flux profile measurements within coolant channels were performed in December 2012 leading to an investigation of 5 of the 16 fuel assemblies.

The axial profiles obtained from measurement are then compared to neutronic calculation results from detailed models of the OPAL reactor. The comparison is used to assess the accuracy of the models and calculations and to identify the effect of various approximations adopted. Detailed results and comparisons are presented in this paper.

2. Experimental setup

2.1 Neutron detector

CEA proposed to use Ø1.5 mm sub-miniature fission chambers, manufactured by the PHOTONIS France Company under CEA patent [1]. They are of the gas ionizing detector type, using 10 µg ^{235}U enriched uranium fissile deposit for conversion of incident neutrons to charged particles (fission products). Due to the ^{235}U enrichment, these sub-miniature fission chambers are mainly sensitive to the thermal part of the neutron spectrum.

Horizontal neutron profile measurements are achieved by means of three identical Ø1.5 mm fission chambers (FC) placed on the irradiation rig at the same height. These FCs are referenced FC105, FC108 and FC107 and are placed respectively from bottom to the top in Fig. 1 below.

2.2 Irradiation rig

A dedicated experimental rig was designed jointly by CEA and ANSTO specifically for this measurement campaign to fit within the nominally 2.45 mm wide fuel coolant channels. The rig design is based on a 2 mm thick aluminium plate precisely machined to host the three FCs and their integrated mineral cable. Aluminium strips keep sensitive parts of the FCs and cable in position. Extreme care has been applied to smooth every potentially sharp edge to protect the fuel clad.

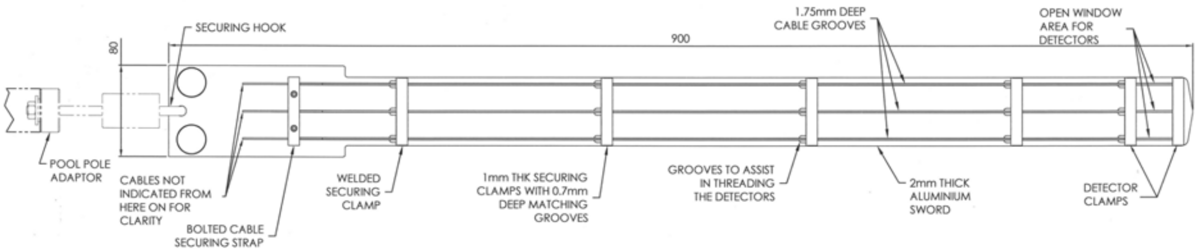

Fig. 1: Dedicated experimental rig overall sketch

The aluminium plate is attached to a 16 metre long pole made of carbon fibre for precise manoeuvring within the core. Axial profile measurements are achieved manually over the core height using a measuring tape attached to the pole.

2.3 Signal acquisition system

The OPAL reactor was stabilized at a power of 100 kW, corresponding to a total neutron flux of ~4.10^{12} $n.cm^{-2}.s^{-1}$. In these conditions, the three sub-miniature FC need to be operated in Campbell (or Fluctuation) mode [2]. A new acquisition system was specifically developed by CEA to perform real time synchronous and simultaneous signal acquisition from a maximum of four separate fission chambers, over a wide range of neutron flux by operating jointly the Pulse and Campbell modes. The new system is called MONACO, for ‘Multichannel Online Neutron Acquisition in Campbell mOde’ [3] [4] and is shown in Fig. 2.

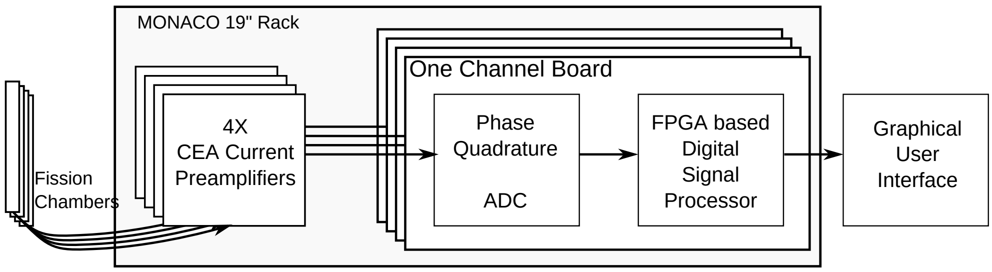

Fig. 2: Schematic diagram of the MONACO acquisition system.

3. Calculation model

3.1 Modelling procedure

Monte Carlo simulations of neutron transport in the reactor were conducted using the MCNP5 program, version 1.60 [5], to determine the flux distribution within the core. The model includes the reactor core, control rods and associated structures, the heavy water reflector and some of the surrounding light water. Within the reflector are the irradiation facilities, a neutron cold source and neutron beam tubes, which are included in the model and have a significant impact on the flux distribution within the core.

There are 16 fuel assemblies within the core each containing 21 fuel plates. The fuel meat in each plate was divided into 10 axial layers, each layer being 61.5 mm high. For each axial layer, the fuel composition is unique but the same for all fuel plates in one fuel assembly. Each fuel assembly contains burnable poison in the form of 20 vertical cadmium wires, extending from the bottom of axial layer 3 to the top of layer 7, where the layers are numbered from the bottom to the top of the core. The cadmium wires are 0.5 mm in diameter and for modelling purposes are divided into 5 axial layers (corresponding to fuel layers 3 to 7) and further into 4 annular layers of equal volume, to allow for differential depletion of ^{113}Cd. The fuel composition and cadmium depletion data are derived from a diffusion theory model which is described below.

The hafnium control plates were placed in the model at representative positions used during the measurements. The partial burnup of the hafnium during five years of reactor operation was not taken into account.

The aluminium plate on which the FCs were mounted was included in the model; however the FCs themselves were not. The plate was raised in 30 mm increments between each measurement. These movements were replicated in the model by performing 22 separate simulations. To speed up the calculations an aluminium plate was placed simultaneously in each of the five fuel assemblies in the model. Given the distance between aluminium plates the flux in one aluminium plate would not have been greatly perturbed by the presence of the other aluminium plates in the model.

In each measurement location, the neutron flux was tallied in the simulation over the 1.5 mm diameter, 10 mm high volume corresponding to the active zone of the FCs. Total flux over all neutron energies, thermal flux up to 0.625 eV and total fission rate over all neutron energies and for all uranium isotopes in the chamber were determined.

3.2 Diffusion theory model

The fuel composition (including the cadmium wires) used in these simulations, allowing for burnup, was obtained from a neutron diffusion theory model with three energy groups using the code CITVAP [6], an enhanced version of CITATION [7]. Homogenized cross-sections for the various regions in the model were obtained using the code CONDOR [6], using collision probability methods. The diffusion theory model includes the reactor core, the reflector and facilities contained therein except for the outer irradiation facilities, and some of the surrounding light water.

4. Results

Neutron flux profile measurements within coolant channels of 5 of the 16 fuel assemblies were performed with a total of 400 measurement points [8]. A manual irradiation rig moving system was used for axial neutron flux profiles with measurement every 30 mm. All measurement locations are represented in Fig. 3.

4.1 Measurement data processing

The reactor conditions were monitored throughout the measurements using the reactor ex-core instrumentation to normalize every coolant channel measurement to a reactor power of 100 kW. Sub-miniature FC absolute fissile deposit masses have been evaluated by the PHOTONIS France Company using an alpha particle counter. These absolute masses are used to normalize each FC response to a 10 µg ^{235}U enriched uranium deposit.

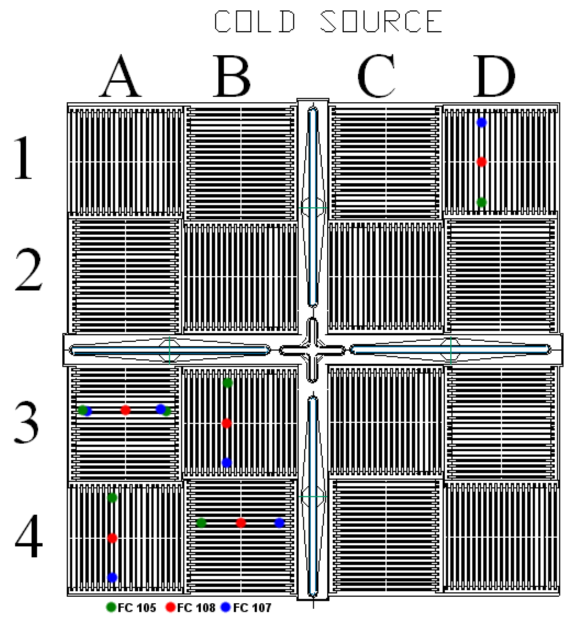

Fig. 3: Investigated fuel coolant channels in the OPAL reactor core.

4.2 Neutron flux spectra

Neutrons flux spectra have been calculated for each of the five investigated coolant channels over the 22 axial positions.

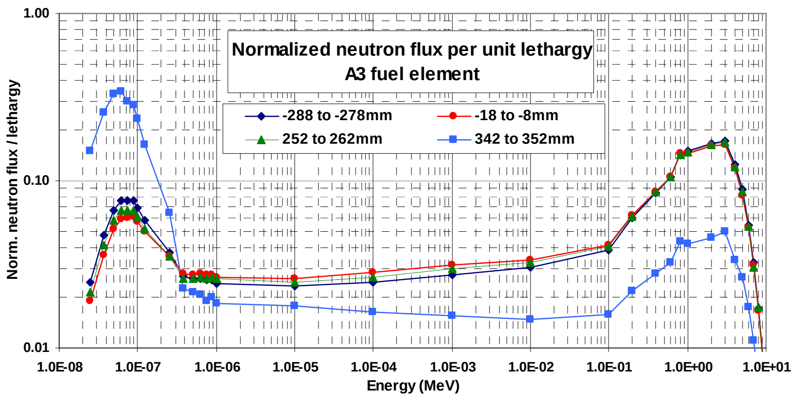

Fig. 4: Normalized neutron flux per unit lethargy over the height of the A3 fuel element.

Results are presented for the A3 fuel element but are similar for the five investigated fuel elements. Neutron flux spectra are identical over the height of fuel meat but more thermalized, as expected, in the top section of the fuel plate where there is no fuel meat.

4.3 Neutron flux profiles

All measurement and calculation data are presented on the plots of Fig. 5. Measurements values are normalized FC variances and calculation values are normalized total fission rates in FCs. Variances and calculated total fission rates are directly comparable providing the FC fissile deposit isotopic composition is taken into account.

Generally the simulated and experimental profiles match very well. The various different shapes are reproduced in the simulations along with the peaks and troughs and even the sharp increase in flux at the top as the FCs are raised out of the fuel meat zone.

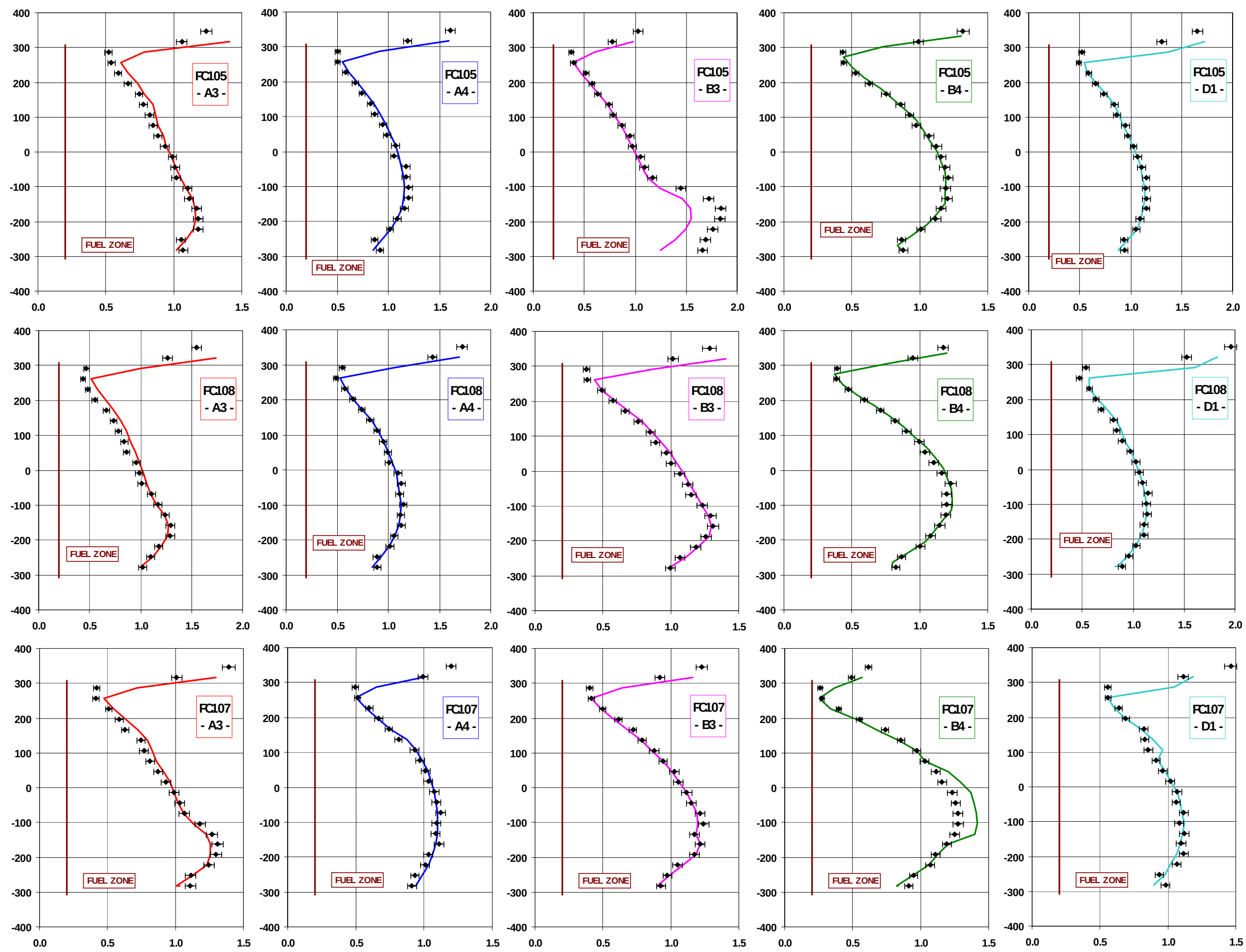

Fig. 5: Measured and calculated neutron profiles in the five investigated fuel elements for the three FCs (Dots are for total fission rates calculations [2σ]; lines for FC measurements)

However the measured data needed downward displacements relative to the simulated profiles. The probable explanation for this is that the plate holding the FCs was connected to the rig used to raise it between measurements in such a way that the first height increment in each fuel assembly was less than 30 mm, but all subsequent height increments were exactly 30 mm. The displacement is typically 15 to 30 mm, as estimated using the positions of the flux minimum at the top of the fuel and maximum just below the core centre plane. The proportional change between the central maximum and the minimum at the top of the core is greater in the simulation than as measured, except for two FCs in fuel assembly B4 and one in A4. Thus the simulated profile generally has a greater gradient than the measurements in the upper half of the core. The reason for this difference between simulated and experimental profiles is still unknown despite several investigations.

Flux profiles for FC positions in fuel assemblies A3 and B3 are influenced by the adjacent control rod which is 35% extracted, depressing the flux in the top 65% of the core. There are significant differences between the simulation and measurements for FC105 in B3. It is likely that these discrepancies are at least partly due to the neglect of depletion of the hafnium control rods in the model. Simulations to quantify this effect appear to indicate that allowance for depletion does improve the fit between simulation and experiment for this case. The control rod adjacent to B4 is 85% extracted and hence has little influence on the flux profile.

In fuel assembly D1, the experimental profile for FC107 shows a small peak at +140 mm which does not appear in the simulation. The peak is possibly due to an increase in flux just above the top of the cadmium wires, which are unburnt as this is a fresh fuel assembly. The simulation overestimates the central peak of the profiles for all FCs in A3, and underestimates the peak for FC107 in B4. The reasons for these local differences are not known.

4.4 Calculation / Experiment ratios

Calculation to measurement ratios (C/M) are calculated over the actual fuel meat range with normalized FC variances and simulations. Mean C/M ratios for each FC in each investigated coolant channels are presented in Table 1 with 1 σ standard deviation of C/M ratios over the fuel meat height.

	A3		A4		B3		B4		D1	
	Average	1σ (%)	Average	1σ (%)	Average	1σ (%)	Average	1σ (%)	Average	1σ (%)
FC105	**0.97**	4.2	**0.98**	4.1	**1.09**	8.8	**1.02**	3.7	**1.00**	3.1
FC108	**0.95**	6.2	**1.00**	2.6	**0.98**	3.0	**1.01**	4.5	**0.99**	4.7
FC107	**0.98**	5.7	**0.99**	2.6	**1.01**	2.4	**0.99**	8.0	**1.00**	4.4

Table 1: Different mean C/M ratios for all measurement locations (1σ standard deviation)

Mean C/M ratios of this neutron flux profile experiment are very good with 12 out of 15 values falling within 2% from the perfect C/M ratio (C/M=1). Only FC105 in B3 has an unsatisfactory C/M of 1.09 with ~9% standard deviation. This result has been explained in the previous section and arises from the neglect of depletion of the hafnium control rods in the model.

5. Discussion – Conclusion

Simulated fission rate profiles are in good agreement with the fission chamber measurements, confirming the validity of the model for use in reactor safety analysis and in planning reactor operation and utilization. At the same time the results indicate that the model could be improved by allowance for control rod depletion. There are several features which require further investigation. The flux profiles measured through this project form a valuable resource for use in validation of future neutronic models of OPAL, and thereby validation of the simulation software. This paper has discussed a Monte Carlo model of the reactor, but it is anticipated that these measurements will also be used in validation of the diffusion theory model of OPAL, which is used for safety analysis such as calculation of the power peaking factor (involving calculation of the core flux profile) as well as for fuel management.

Experimentally, on-line neutron flux profile measurement in fuel coolant channels is challenging due to the geometrical constraints of the irradiation locations and the need to strictly follow OPAL safety based operating conditions. Nevertheless the very good C/M comparison results demonstrate the adequacy of this experimental set-up and only more detailed measured neutron profile with less than a 30 mm increment would ease the C/M comparison and should be required for future measurement campaigns.

6. Acknowledgments

This work was supported by CEA,DEN,DER (reactor studies department) for the Jules Horowitz reactor under construction at CEA Cadarache site and by ANSTO. Authors are grateful to the OPAL Reactor operation teams for their involvement in this measurement campaign.

6. References

[1] Miniature insulated fission chamber, by G. Bignan and J-C. Guyard (1996, May 5). *Patent EP0715186* [Online]. Available https://data.epo.org/

[2] Wide range neutron detection system, by W. H. Todt (1978, Aprils 25) *US Patent 4086490 A* [Online]. Available http://www.google.com/patents/

[3] M. Thévenin et al (September 2013). ”Digital real-time multiple channel, multiple mode neutron flux estimation FPGA-based device”, to be presented at the ADVCOMP conference, Porto, Portugal. [Online]. Available http://www.iaria.org/conferences.html

[4] C. Carasco et al (August 2012), “Data acquisition and analysis of the uncoss underwater explosive neutron sensor”, *IEEE Transactions on Nuclear Science*, vol. 59, Issue 4, pp. 1438-1442.

[5] X-5 Monte Carlo Team (2003), MCNP – A General N-Particle Transport Code, Version 5. LA-UR-03-1987, Los Alamos National Laboratory (Los Alamos, NM, USA).

[6] I. Mochi (2011), INVAP's nuclear calculation system, Sci. Tech. Nuclear Installations 2011, article 215363.

[7] T.B. Fowler et al (1972), Nuclear reactor core analysis code: CITATION, ORNL-TM2496 Rev.2 Supplement 3, Oak Ridge National Laboratory (Oak Ridge, TN, USA).

[8] L. Barbot et al (June 2013). ”On line neutron flux mapping in fuel coolant channels of a research reactor”, presented at the ANIMMA conference, Marseille, France. [Online]. Available http://ieeexplore.ieee.org/