SUCCESSFUL VERIFICATION OF INNOVATIVE CHALLENGES IN RESEARCH REACTOR DESIGN

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Abstract

Nowadays, the design of modern Multipurpose Research Reactors must comply with the fulfilment of very different and stringent user requirements such as large volume Cold Neutron Source, several neutron beams, large volume Neutron Transmutation Doping devices, higher amounts of radioisotope production and more demanding material testing irradiation facilities, among other capabilities. The design of this complex layout of facilities requires a high number of parameters to be consider: nuclear heat load deposition, neutron flux uniformity, thermal to fast neutron flux ratio, neutron spectrums, radioisotope productions and minimal operational perturbations such as control rod movement, fuel burnup, and movement of the irradiation facilities, only to mention a few. The well-known MCNP code is the appropriate tool used to simulate the transport of radiation in real core conditions to the different facility locations. The Monte Carlo calculation code was integrated into the INVAP core calculation line, allowing the feedback of MCNP in all operation conditions, core burnup, Xenon build-up and temperature distributions, for example. The increase in the calculation capability of modern computers redounds in the possibility to include the utilisation of the Monte Carlo method in routine design calculations. These innovative challenges in nuclear design were successfully verified in the OPAL reactor during the commissioning stage and in normal operation. This paper presents how modelling and design verifications were carried out for many systems.

1. INTRODUCTION

INVAP utilizes a proprietary software package to design nuclear installations such as Research Reactors, Radio-Isotopes Production Reactors and Nuclear Power Plants. Through the years, the codes became more flexible and user-friendly to the point that the two graphic post-processors provide intuitive and self-explanatory 1D, 2D and 3D representations of the variables involved in calculations. This software package includes:

— CONDOR, a *cell code* which performs Fuel Assembly calculations and produces sets of data, like the homogenized and condensed cross sections to be used in the reactor (or core) code;
— CITVAP, a *reactor code* which performs burnup-dependent calculations of reactor cores, calculating the nuclear parameters associated with several states of the reactor taking into account the feedback of many thermal-hydraulic parameters. There is also a nodal code (under development) which uses automatic pin-power reconstruction, among other features;
— ESINLM, a *nuclear data library manager* which updates and upgrades the nuclear data required by CONDOR;
— HXS, a *macroscopic library manager* which provides the link between the cell and the reactor codes;
— POSCON and FLUX, two *graphic post-processors* which enable the user to view in a fast and easy way the calculated parameters of both the cell code and the reactor code and export information and graphics from the respective databases;
— Several *utilitarian programs* that serve specific functions to ease the interface with the user by graphically pre-processing the geometry input data, exporting data in MCNP format (through NDDUMP, see Figure 1), and many other functions required.

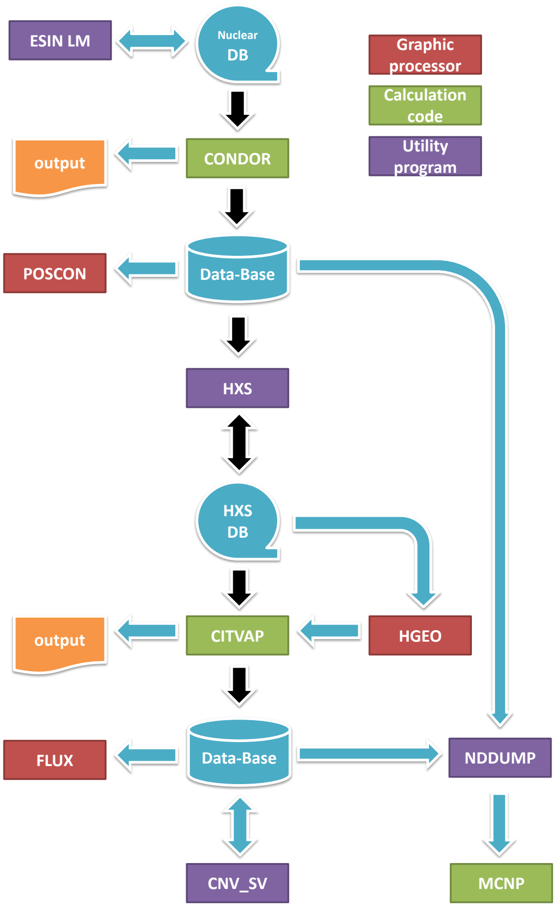

Fig. 1. INVAP's nuclear calculation system description.

The nuclear calculation system (the most relevant programs are shown in Figure 1) includes the calculation codes, utilitarian programs, post-processors and, as an external calculation package, the MCNP Monte Carlo code. This code is used as an alternative method in the cases where the diffusion approximation used by the reactor code is not accurate enough or cannot provide the desired parameters.

2. APPLICATION TO A HIGH VOLUME NTD FACILITY

The commercial specifications of neutron transmutation doping (NTD) services with very high uniformity, in a very large irradiation volume, lead to the design of a high performance NTD Facilities.

A high performance NTD facility should fulfil the following requirements:

— High axial and radial uniformity in large Silicon Target volumes;
— Low Thermal Neutron flux perturbation during the operation cycle;
— Low Thermal Neutron flux perturbation due to the operation of facilities placed close to the NTD facility and,
— High thermal to fast neutron flux ratio.

The conception of a high axial uniformity NTD facility is focused on the design of the flux-flattener device and the top and bottom plugs.

Radial uniformity is obtained placing the NTD facility in a low thermal neutron flux gradient region and it is improved with the rotational movement of the Silicon ingot.

The utilization of MCNP (ref. 0) allows a full description of the operational conditions and geometrical characteristics of the reactor core. The model includes full details of the core such as typical burnup distribution (radial and axial), burnable poisons localization, enrichment distribution and critical control rod positions. The full detail description extends to the surroundings of the core where the irradiation facilities, cold neutron source, neutron beams and NTD facilities are located. The NTD description includes the Silicon single crystal target, its can, the rotator device, the flux-flattener device and the top and the bottom plugs.

Criticality calculations are used by the MCNP to generate neutrons from fission in the critical core. A surface source is created to include the interest region, i.e., the NTD facility and moderator region close to the facility.

A high number of neutrons are created by the KCODE source and only those that arrive at the region involved are recorded in a surface source. The design is carried out using the surface source as a neutron source. Each particle coming from the surface source is initialized several times (typically 10 to 20 times) with its appropriate weight normalization. The source particles have collisions in the moderator region close to the NTD facility and quickly each particle that was born from a same neutron recorded in the surface source goes over a different path, becoming in a different history. The advantage of this technique is that it is possible to obtain neutron flux information with a very low statistical error (lower than 1%) in a short calculation time.

After several modifications of the NTD design parameters a new surface source is recorded to obtain a feedback on the neutron population of the surface source due to the NTD changes. During this feedback process, differences of about 1 to 5% in the axial profile were found, especially when large regions of neutron absorber are introduced or removed. Fig. 2 shows a typical NTD irradiation Facility.

A high axial uniformity of the thermal neutron flux in a large volume NTD facility is obtained with a flux-flattener device. The flux-flattener is a cylinder made of Aluminium, Stainless Steel and Nitrogen or void regions.

Additionally, a high axial uniformity of the thermal neutron flux is achieved using reflector plugs. At both target extremes (top and bottom) a reflector plug is placed in order to improve the thermal neutron flux and the radial uniformity in the extremes of the Silicon target. Along the target volume, the radial uniformity is obtained with the non-stop rotation of the Silicon target.

The low thermal flux perturbation during the operation cycle is obtained with a compact core design, an adequate fuel management strategy and a defined control rod movement during the operation cycle.

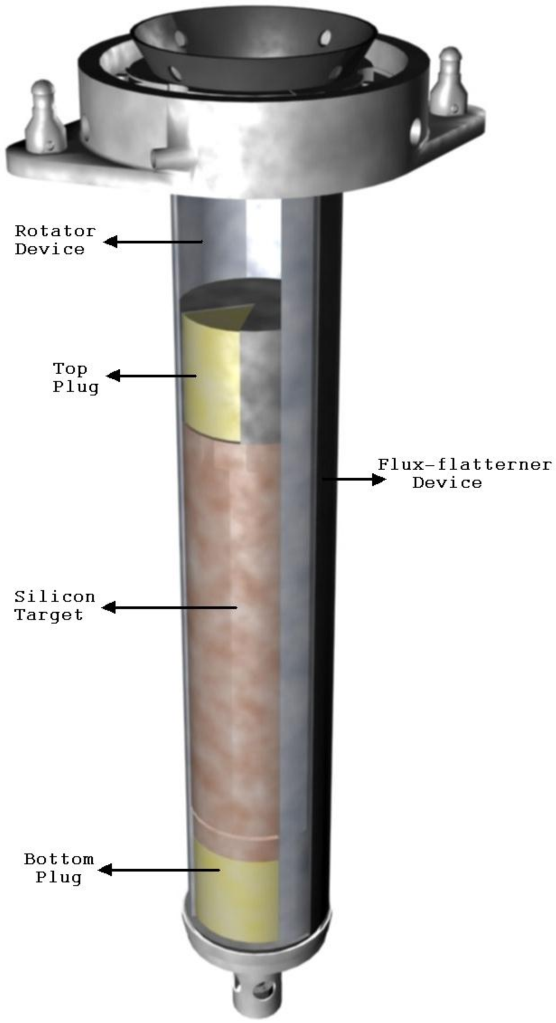

Fig. 2. Schematic view of a NTD irradiation facility.

Figure 3 shows the thermal neutron flux in a NTD facility with a silicon target of about 8 inches diameter and 60 cm length.

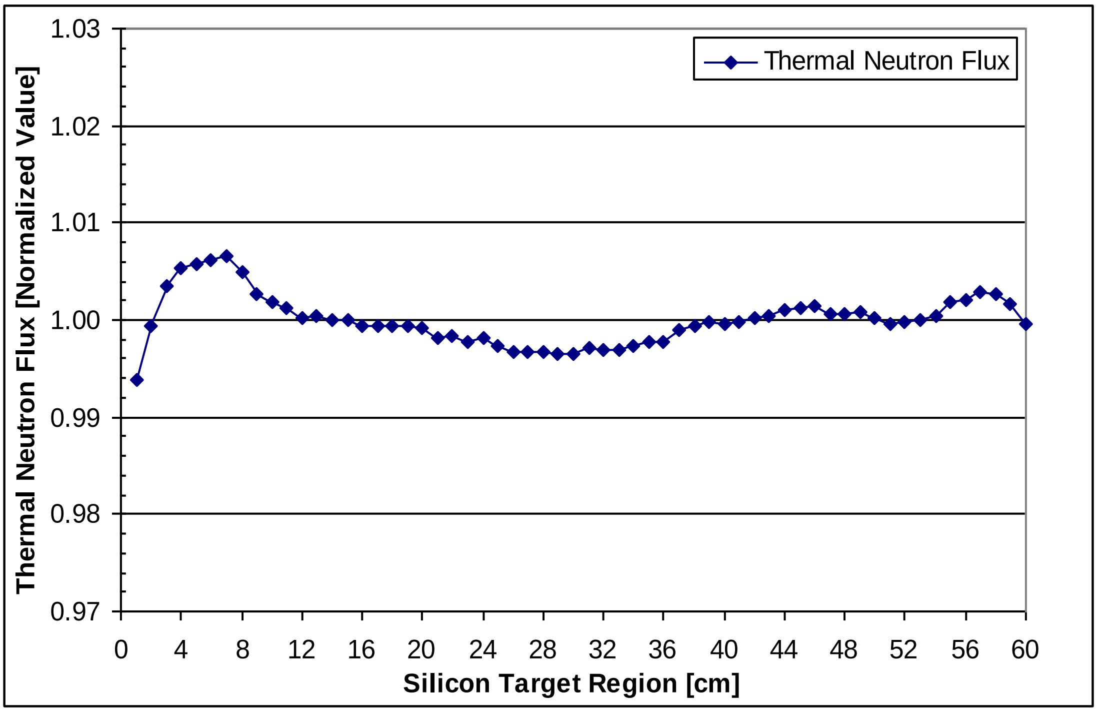

Fig. 3. Thermal neutron flux in a NTD facility with a Silicon target of about 8 inches diameter and 60 cm of length.

2.1. Verification

Design requirements for the OPAL NTD facilities were:

— Axial uniformity: +/- 5%,
— Thermal-to-Fast flux ratio: >200

Table 1 shows the thermal neutron flux measured in the NTD facilities, together with the measured values.

TABLE 1: NEUTRON FLUX VALUES IN THE NTD FACILITIES 0

Facility	Requirement	Measurement	99.7%(1)	Thermal-to-fast ratio (>200)	Axial non-uniformity [+/- 5%]
NTD-1	$1.0\ 10^{13}$ (+/- 20%)	$8.93\ 10^{12}$	$9.07\ 10^{12}$	812	4.1
NTD-2	$3.2\ 10^{12}$ (+/- 20%)	$3.02\ 10^{12}$	$3.15\ 10^{12}$	2290	3.45
NTD-3	$1.9\ 10^{13}$ (+/- 30%)	$1.62\ 10^{13}$	$1.65\ 10^{13}$	1285	3.1
NTD-4	$1.0\ 10^{13}$ (+/- 30%)	$8.03\ 10^{12}$	$8.47\ 10^{12}$	593	3.4
NTD-5	$1.1\ 10^{13}$ (+/- 30%)	$9.61\ 10^{12}$	$1.05\ 10^{13}$	384	4.55
NTD-6	$3.5\ 10^{12}$ (+/- 20%)	$2.77\ 10^{12}$	$3.06\ 10^{12}$	4800	2.0

(1) Measured values normalized to 99.7% D_2O purity.

3. APPLICATION TO THE DESIGN OF THE CNS AND RELATED NEUTRON BEAMS

The goal of the OPAL CNS design was to find the optimal combination between positioning and geometry of the moderator chamber and composition of the moderator material to produce the maximum cold neutron flux at experimental locations (downstream in the cold neutron guides). Close to the optimum balance, the influence of each of these parameters on the cold flux can be expected to be about 1 to 5%. These small effects must be discriminated from statistical errors without a strong increase of the calculation time.

The design of a CNS is affected by several parameters. They are often competitive and their behaviour is not always linear or monotonic. A typical example of this is the introduction of a cavity (or a displacer) in the CNS Moderator Cell, just in front of the Neutron Beam entrance, in order to increase the cold neutron current that leaves the source in the beam tube direction. These neutrons coming from the CNS centre are cooler and the neutron guide will transport them more efficiently. On the other hand, a large cavity volume reduces the volume of the CNS moderator, i.e., reduces the CNS capability to moderate neutrons and, hence, reduces the average cold neutron flux inside the moderator cell.

Similar challenges are faced during the design of the cold and thermal neutron beams. The goal of the design is the selection of the size and position of the beam entrance in order to have a good illumination of the neutron guides.

The most relevant magnitudes to control during the design of a CNS and neutron beams are:

- Average cold neutron flux inside the CNS Moderator Cell;
- Thermal and Cold Neutron flux spectrums that enter into the Beam Tubes;
- Cold and Thermal neutron flux at experimental locations downstream in the beams.

In terms of CNS utilization, it is clear that the cold neutron flux at experimental locations is the most significant magnitude. Ref. 0 focused on evaluating the effects that several design parameters had on this flux and the way the optimization process was carried out. Same approach is also used for the optimization of the thermal neutron beam.

To design purposes, Monte Carlo numeric techniques used in a bootstrap scheme was the appropriate solution for this type of analysis. The MCNP code, ref. [1], is the tool used to simulate the neutron production under operating reactor core conditions, moderate them in the cold moderator and/or reflector and to transport them to the neutron guide entrance position, located into the beam tube. From the guide entrance to the experimental location, the neutrons are transported through the neutron guide (typically supermirrors). The transport of neutrons along the neutron guides is simulated using alternative Monte Carlo techniques, which takes into account the reflective properties of the neutron guide as a function of neutron energy, and a comprehensive description of the geometrical conditions such as dimensions, curvature radii and misalignment effects. A complete description of the calculation line can be found in ref. 0.

Several parametric analyses of the design variables were evaluated in order to show how the calculation methodology worked and how consistent their results were. In particular, for the CNS design the calculation methodology and its application to the assessment of several parameters are described in several references 0–0.

As a conclusion the possibility to apply Monte Carlo techniques in a design project framework to obtain an optimised CNS and cold and thermal neutron beams neutronic design was demonstrated through the performance evaluation.

Table II shows the measured neutron flux values in several locations along the cold neutron guides. At the same time, measurements are shown in some other locations along the thermal neutron guides and in the reactor face of the beams HB1 and HB2.

3.1 Verification

Design requirements were established for the various beam tubes in different locations downstream the neutron guides (TG and CG lines) and for the beam HB. Table 2 shows the measured performance for the different beam tubes and neutron guide lines.

TABLE 2: MEASURED COLD AND THERMAL NEUTRON FLUX IN DIFFERENT LOCATIONS 0.

Performance Acceptance Criteria	Guaranteed Level of Performance ($cm^{-2}s^{-1}$)	Flux Energy Range	Measured Performance ($cm^{-2}s^{-1}$)
Thermal neutron flux at the reactor face for TG3	$1.6\ 10^{10}$		$\sim 2.3 \times 10^{10}$
Thermal neutron flux at the reactor face for TG4	$1.6\ 10^{10}$		3.99×10^{10}
Thermal neutron flux in the NGH for TG3	$1.6\ 10^{9}$		2.82×10^{9}
Peak in thermal neutron spectrum for TG3	< 1.4 Å		1.33 Å
Peak in thermal neutron spectrum for TG4	< 1.4 Å		1.20 Å
Thermal neutron flux at the reactor face for HB2	Not specified		$\sim 3.8 \times 10^{10}$
Peak in thermal neutron spectrum for HB1	Not specified		~ 1.20 Å
Peak in thermal neutron spectrum for HB2	Not specified		~ 1.20 Å
Cold neutron flux at the reactor face for CG1	$1.4\ 10^{10}$	< 10 meV	$\sim 1.7 \times 10^{10}$
Cold neutron flux at the reactor face for CG3	$1.4\ 10^{10}$	< 10 meV	$\sim 1.3 \times 10^{10}$
Cold neutron flux at the reactor face for CG4	$1.4\ 10^{10}$	< 10 meV	2.49×10^{10}
Cold neutron flux in the NGH for CG3	$3.2\ 10^{9}$	< 10 meV	6.38×10^{9}
Peak in cold neutron spectrum for CG3	< 4.2 meV		3.05 meV
Peak in cold neutron spectrum for CG4	<4.6 meV		2.35 meV

4. FINAL REMARKS

This paper presents the performance of some facilities of the OPAL reactor, measured during the Commissioning and Contract Performance Demonstration Tests stages as a verification of the design carried out using a Monte Carlo based methodology. The measured performance proved to be adequate in every case as neutron flux values are quite within the specified guaranteed levels. The utilization of Monte Carlo techniques during the design stage in a project framework was demonstrated to be appropriate to fulfill the client´s requirements. The conclusion is that the use of the MCNP code as a complement to the INVAP's calculation line was fruitful.

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