# Fission and capture rate measurements in a SVEA-96 Optima2 BWR assembly compared with MCNPX predictions

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## Abstract

As part of a joint research program between the Paul Scherrer Institute (PSI) and swiss*nuclear*, measurements and calculations have been made of the fission and neutron capture distributions in the fuel rods of a Westinghouse SVEA-96 Optima2 boiling water reactor assembly. The measurements were made in the zero-energy research reactor PROTEUS at PSI and the calculations reported in this paper were made at PSI. The results reported here are for the region near the ends of the part-length fuel rods used in SVEA-96 Optima2 assemblies.

***KEYWORDS: research reactor, BWR, part-length fuel rods, fresh fuel, fission rate, $^{238}$U capture rate, MCNP***

## 1. Introduction

The joint research program LWR-PROTEUS [1] was set up in 1997 between the Paul Scherrer Institute (PSI) in Switzerland and the Swiss nuclear operators (swiss*nuclear*) [2] to consolidate the understanding of the performance of advanced LWR fuel designs. Both reactor physics measurements and calculations have been carried out in the execution of the program, with the measurements being made in the PROTEUS zero-energy research reactor at PSI. Phase I of LWR-PROTEUS [3,4,5] was concerned with the detailed mapping of fission and $^{238}$U capture rates in Westinghouse SVEA96+ boiling water reactor (BWR) fuel assemblies. In Phase II [6], the reactivity worths of burnt pressurized water reactor (PWR) fuel samples were measured as they were inserted into a PWR lattice. Phase III, which is currently in progress, furthers the studies of modern BWR fuel, in particular Westinghouse SVEA-96 Optima2 assemblies. It should be noted that in all three phases, full-length authentic fuel has been used, rather than experimental mockups.

In the search for greater cost effectiveness in the operation of BWRs, fuel designs have become increasingly more complicated in terms of fuel enrichment variations, the inclusion of burnable poisons, and the use of novel geometrical features. For example, in the case of Westinghouse SVEA-96 Optima2 assemblies [7], besides the heterogeneous fuel enrichment differences, each corner rod is only one third the normal length, and the rods surrounding the central water canal are only two thirds normal length. The part-length rods present particular challenges to calculational methods because of the step changes in fuel and moderator configurations.

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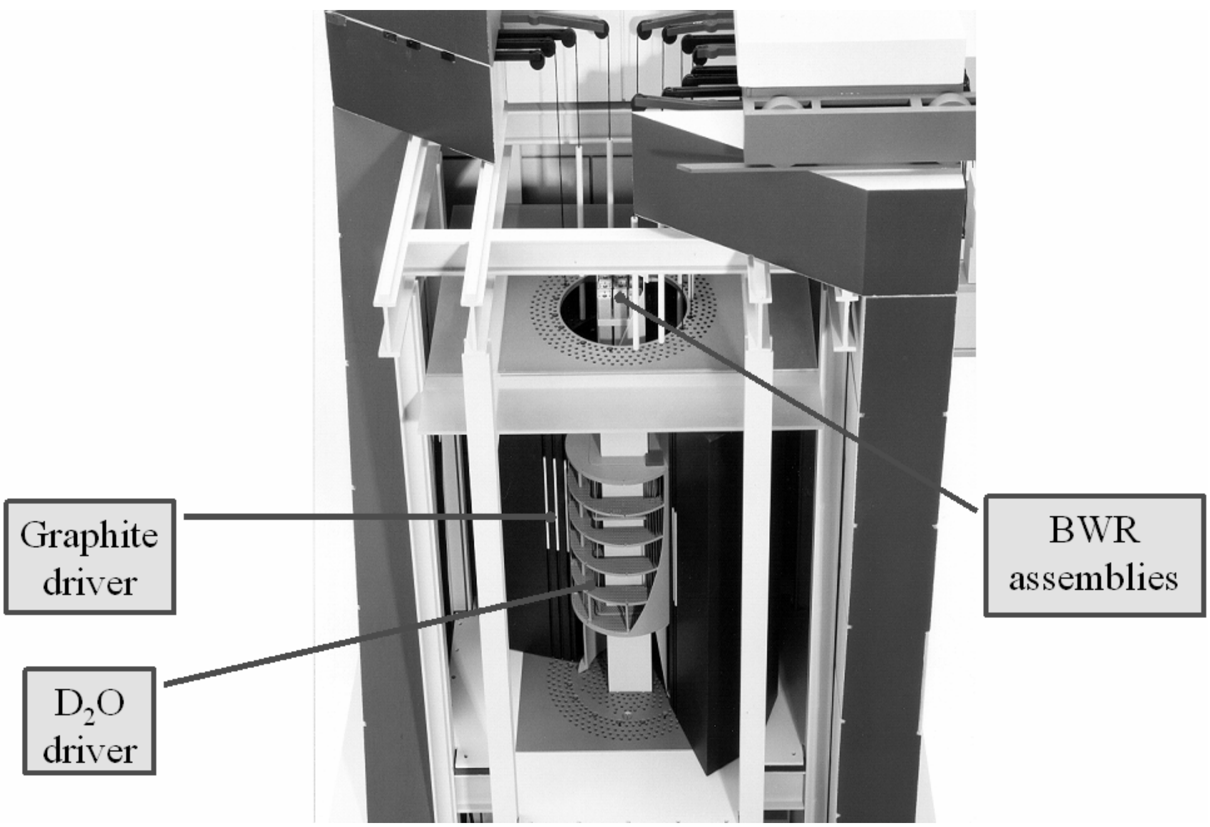
The first part of Phase III of the LWR-PROTEUS reactor research program (Cores 1 and 2) has been devoted to measuring the distribution of fission and $^{238}U$ capture rate distributions in and around the ends of the part length fuel rods in a SVEA-96 Optima2 BWR assembly. The aim is to produce a measurement database to prove the capabilities of reactor codes in these difficult situations. In Core 3 of Phase III, the effects of moderator voids have been measured, using various mixtures of light and heavy water, in special tanks, to simulate the hydrogen densities of different coolant voidage. The present paper describes the measurements and calculations at the ends of the one third length rods made in Phase III Core 1.

## 2. Experimental Method

### 2.1 Core Configuration in PROTEUS

The LWR-PROTEUS experiments are conducted in the PROTEUS research reactor (Fig. 1), which consists of a central test region surrounded by annular driver zones to achieve criticality.

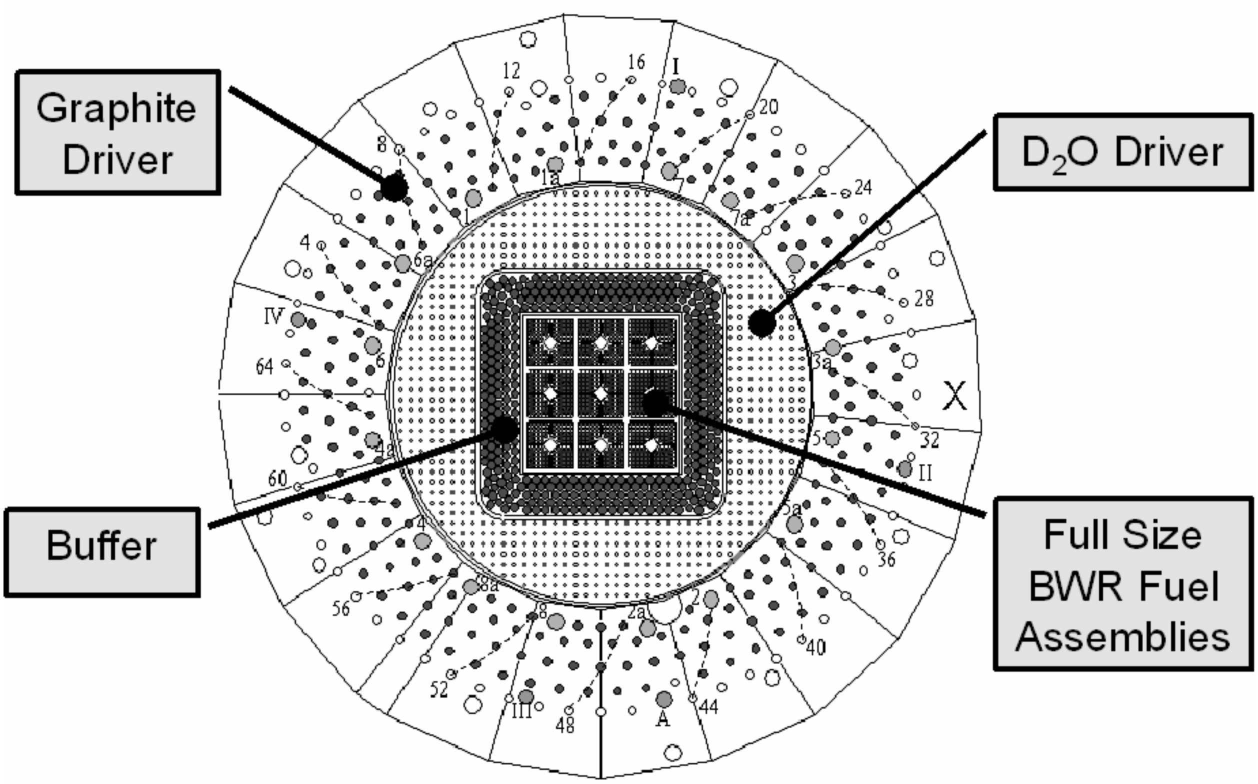
**Figure 1:** The general layout of the PROTEUS reactor.




These zones are, working from the inside out; the test zone consisting of a 4.6 meter long aluminum tank containing nine BWR assemblies, the natural uranium fueled buffer zone, the 5% $UO_2$ fueled heavy water zone, the 5% $UO_2$ fueled graphite driver zone, and finally, the graphite reflector zone. All the control, shut-down, and safety rods are situated in the graphite driver zone, as are the neutron and gamma detectors, so that the experiments in the test zone can be performed under unperturbed conditions. The graphite driver zone has remained virtually unchanged since its inception in 1968. The 1.2 m diameter cylindrical cavity within the graphite has contained numerous reactor concepts over the decades, including a gas cooled fast reactor, a close pitch high conversion reactor, and a particulate fuel, pebble bed, reactor.

For the present measurements, a SVEA-96 Optima2 BWR test assembly was placed at the center of the reactor and surrounded by eight other similar assemblies, the 3x3 arrangement being located inside the aluminum test tank (Fig. 2).

**Figure 2:** Fuel arrangement for a typical LWR-PROTEUS configuration.

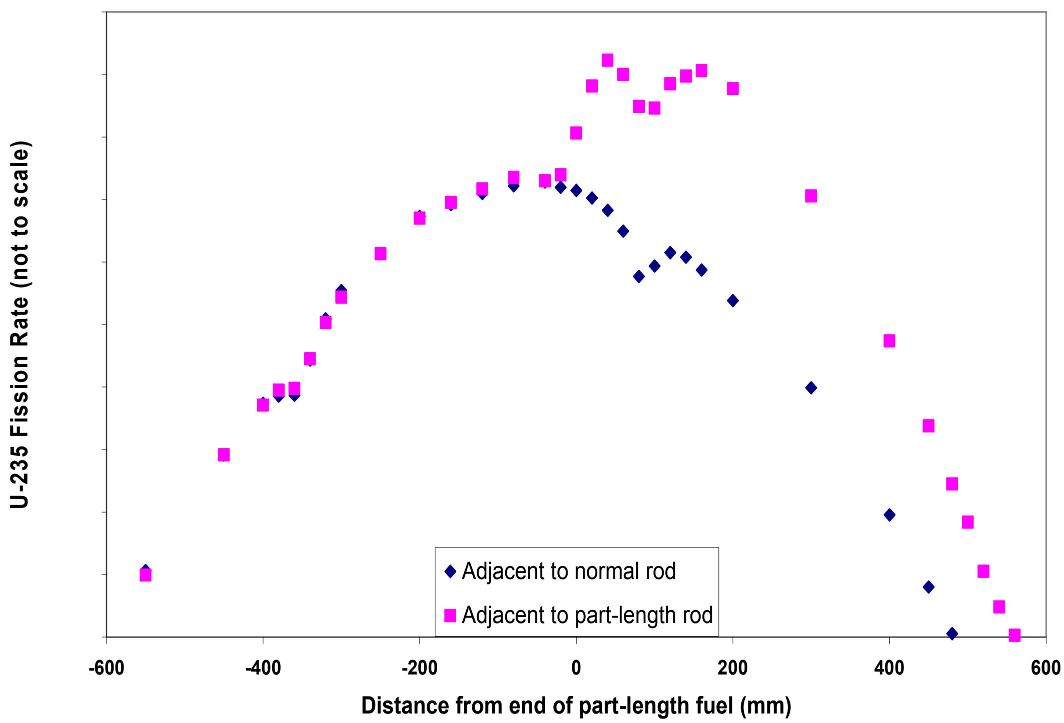



The central assembly in the test tank had specially treated spacers with reduced spring tensions, this meant that fuel rods in this assembly could be removed and replaced without damage. At the end of the experiments, these spacers were replaced by new ones before the assembly was returned to the power plant. Since the BWR assemblies are 4.5 m in length and the active height of the PROTEUS driver regions is about 1 m, the test tank can be driven axially to enable step-wise investigations along the whole length of the test assemblies. A moderator consisting of a mixture of light and heavy water was used to simulate the hydrogen content of hot full-density water in an operating power plant. In Core 1 of LWR-PROTEUS Phase III the test tank was positioned so that the ends of the one-third length fuel rods were at the core mid-plane.

## 2.2 Fission chamber scans

Scans were made with a miniature $^{235}U$ fission chamber adjacent to the part length rods and in unperturbed parts of the assembly. The scans were achieved by replacing a chosen fuel rod by a thin walled stainless steel guide tube. This situation is somewhat artificial, of course, but it does give a good indication of the change of thermal flux adjacent to a part length rod (Fig. 3). In observing the scans, it should be noted that the flux distribution appropriate to the 1 m high PROTEUS driver regions are superimposed on what would normally be expected for a BWR power reactor. The slight depression in the fission rate above the step change at the rod end is due to the absorbing effect of the spacers at that height.

**Figure 3:** Fission rate scans adjacent to normal and part-length fuel rods.




## 2.3 Fuel rod gamma-scanning procedure

Fuel rods were activated in preparation for a γ-scanning measurement by running the reactor at a power of about 40 W (thermal flux $2x10^8$ $ncm^{-2}s^{-1}$) for one hour. Up to ten fuel rods were then removed from the test assembly in preparation for the γ-scan. Ten main γ-scanning measurements were made, six radial scans at three vertical positions with ten fuel rods in each, and four detailed axial scans, with four rods in each, near to the part-length rods. A final normalizing irradiation was made with a selected rod from each of the previous ten scans being measured together on the γ-scanner.

The γ-scanning machine [8] is a purpose-built and fully automatic device, which enables the cyclic γ-scanning of activated fuel rods up to 4m in length. At the measurement position, there are two horizontally opposed germanium γ-ray detectors installed behind γ-ray collimators and shielding. The detectors are connected to EG&G Ortec DSPec devices, which comprise, in one unit; a high voltage supply, a spectrometry amplifier and a multi-channel analyzer. Digital signal processing is employed to produce an optimum virtual pulse shape that gives good energy resolution and peak position stability over a wide range of count-rates.

Each fuel rod was measured six times for the radial scans and three times for the axial scans. The collected spectra were analyzed using the Ortec GammaVision software, which subtracts the background continuum, derives the peak energies and areas, and identifies the nuclides. Decay corrections were made using a separate program, which took account of the complicated irradiation histories and could also correct for complex decay chains. The statistical uncertainties of the measured fission and capture rate distribution were within the range 0.4-1.0%.

The γ-ray transmission corrections from rod to rod were significant, particularly for the γ-ray from the $^{238}$U capture product $^{239}$Np at 277 keV, because the within-rod distributions were different in the various rods, due to the heterogeneous nature of the assembly. The corrections were derived in a two step process [9]. Firstly, a reflected assembly MCNPX model was used to obtain rod-wise reaction and fission rates, with the rods divided into 10 equal-volume annuli. Then the within-rod activity distributions so obtained were used as input to an MCNPX γ-ray transport model to derive the appropriate transmission corrections for each rod.

## 3. MCNPX Calculations

A whole reactor 3D MCNPX-2.5 [10] model in conjunction with the American nuclear data library ENDF/B-VI (release 2) was used to predict the fission and capture rate distributions. The model was developed for this present reactor configuration from a previously validated MCNP4B model [11], but various improvements have been incorporated [12] to optimize the modeling to reproduce the Phase III configurations. The four different zones of the PROTEUS facility and the positions of the 16 shutdown rods (borated rods) and the 4 control rods (elements comprising Cd-segments) are explicitly modeled.

The MCNPX2.5 predictions have been performed using a parallel processing system, called MERLIN, comprising 16 CPUs. This significant improvement in computer capability at PSI has facilitated the modeling of whole-reactor 3D models of the highly heterogeneous PROTEUS research reactor. A typical run of $2 \times 10^9$ histories, giving uncertainties at 1σ of ~0.3% for total fission and ~0.6% for $^{238}$U capture rate took about 1 week.

Typical predictions using MCNPX are shown in Fig. 4, where the results for the axial total fission rate distributions in a fuel rod adjacent to a part length rod are featured. The good agreement with the γ-scan and the similarity to the fission chamber scans given in Fig. 3 goes so far as to include the depressions caused by the fuel rod spacers. Agreement within statistical uncertainties was found between γ-scan experiments and MCNPX predictions for the axial scans, indicating both exact modeling and the capabilities of the code.

Tab. 1 demonstrates the generally good agreement between prediction and experiment for one of the radial total fission rate scans, bearing in mind that these results were produced using the nominal dimensions of the assemblies. Sensitivity studies have shown that a movement of 1.5 mm in the assembly casing position causes a 3% change in fission rate for the outer row of rods, and that an increase of the rod pitch by 0.1 mm causes a 2.5 % increase of fission rate at the sub-bundle center. Further analysis will use measured rod positions to attain a better reproduction of the real-life situation. In Tab.2 is shown the analogous information for the $^{238}$U capture rate.

Figure 4. Axial total fission distribution (not to scale) predicted with MCNPX and measured by gamma-scanning in a rod adjacent to a part-length rod.

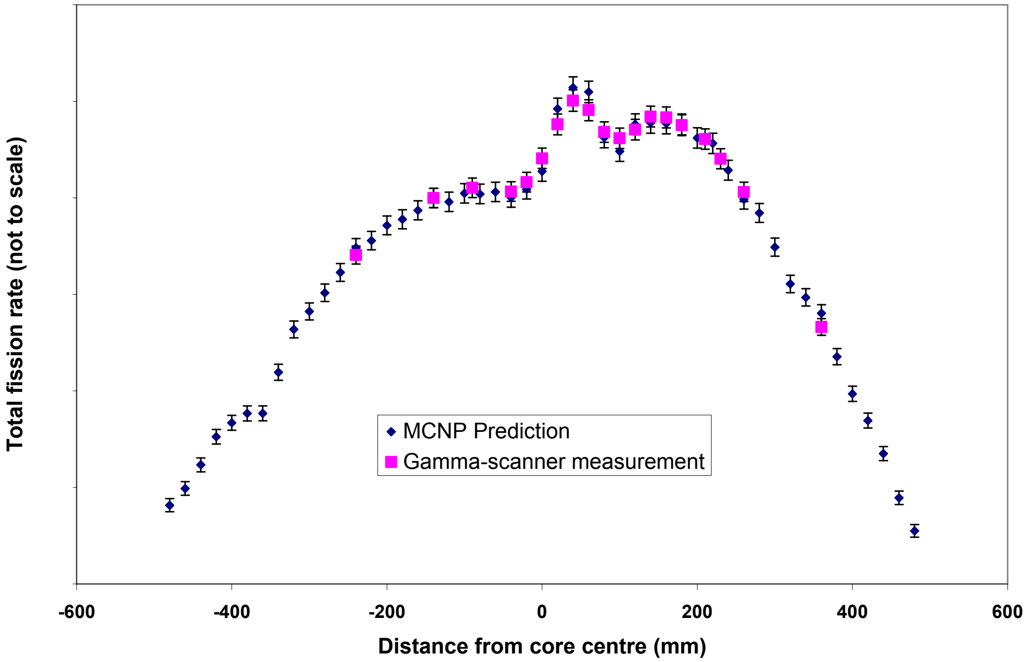



Table 1. Calculation/Experiment ratios for a radial scan of total fission rate.

| | 10 | 9 | 8 | 7 | 6 | 5 | 4 | 3 | 2 | 1 |
|---|---|---|---|---|---|---|---|---|---|---|
| J | 1.037 | 1.021 | 1.021 | 1.018 | 1.041 | 1.050 | 1.006 | 1.037 | 1.054 | 1.015 |
| I | 1.037 | 0.949 | 0.976 | | 1.010 | 0.994 | | 0.962 | 0.947 | 1.019 |
| H | | 0.937 | | | 0.988 | 1.000 | | | 0.968 | 1.003 |
| G | 1.016 | | 0.965 | 0.952 | | 1.002 | 0.943 | 0.980 | | 0.991 |
| F | 1.028 | 1.014 | 1.006 | 1.017 | | | | 0.990 | | 1.016 |
| E | 1.032 | 1.001 | 1.000 | 0.982 | | | 1.004 | 0.992 | 1.006 | |
| D | | | 0.980 | 0.965 | 1.022 | 0.978 | 0.960 | 0.954 | | 1.002 |
| C | 1.018 | 0.981 | | 0.971 | 0.981 | 0.984 | 0.976 | | 0.981 | 1.012 |
| B | 1.018 | 0.993 | 0.962 | | 0.988 | 0.980 | | 0.974 | 0.988 | 1.030 |
| A | 1.021 | 0.999 | 0.995 | 0.987 | 0.992 | 0.986 | 0.987 | 0.990 | 0.989 | 1.008 |

Table 2. Calculation/Experiment ratios for a radial scan of $^{238}U$ capture rate

| 10 | 9 | 8 | 7 | 6 | 5 | 4 | 3 | 2 | 1 | |
|---|---|---|---|---|---|---|---|---|---|---|
| 1.018 | 0.996 | 0.996 | 0.994 | 1.017 | 1.020 | 0.991 | 1.023 | 1.037 | 1.001 | J |
| 1.017 | 0.975 | 1.002 | | 1.006 | 1.003 | | 0.971 | 0.969 | 1.000 | I |
| | 0.929 | | | 0.997 | 1.007 | | | 0.992 | 0.993 | H |
| 1.016 | | 0.989 | 0.994 | | 1.012 | 0.989 | 0.992 | | 0.999 | G |
| 1.035 | 1.003 | 0.999 | 0.993 | | | | 0.998 | | 0.988 | F |
| 1.011 | 1.005 | 0.979 | 1.004 | | | 1.009 | 0.997 | 1.002 | | E |
| | | 0.984 | 0.993 | 1.011 | 0.985 | 1.005 | 0.985 | | 1.015 | D |
| 0.998 | 0.976 | | 0.978 | 1.013 | 0.994 | 1.001 | | 0.996 | 1.004 | C |
| 0.989 | 1.010 | 0.976 | | 1.013 | 0.988 | | 1.001 | 1.020 | 1.012 | B |
| 1.030 | 0.979 | 1.002 | 1.001 | 1.003 | 1.001 | 0.991 | 1.003 | 0.988 | 1.024 | A |

## 4. Conclusions

Total fission and $^{238}U$ capture rate distributions have been measured in a fresh Westinghouse SVEA-96 Optima2 BWR assembly. The moderator used was a mixture of light and heavy water simulating the hydrogen content of hot full density water. The measurements reported here were concentrated on the region around the ends of the one third length fuel rods at the corners of the assembly.

Excellent agreement has been found between the experimental results and the MCNPX predictions for axial fission rate distributions. In the case of radial reaction rate distributions, the comparisons made to date have indicated the need for taking into account the actual fuel rod positions in the experiments, rather than simply the nominal fuel assembly dimensions.

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