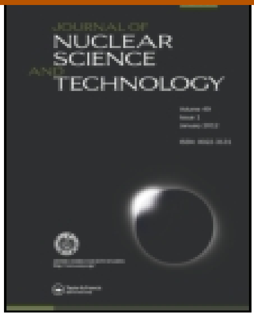

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TECHNICAL REPORT

The Effect of Nuclear Data on the MCNPX Modeling of Moderator Level Variations in the CROCUS Critical Facility

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A detailed MCNPX (version 2.4.0) model of the CROCUS zero-power critical facility was used to investigate the effect of nuclear data on the analysis of reactivity perturbations caused by variations in the moderator level. The analysis was performed using the ENDF/B-VI (Release 0), ENDF/B-V, JEF-2.2 and JENDL-3.2 cross-section libraries, and compared with experimental values derived from period measurements. The results of this study demonstrate that predictions of the reactivity effects of moderator-level variations are largely independent of the nuclear data evaluation even when the associated k_{eff} values vary significantly.

KEYWORDS: MCNPX, moderator level, reactivity perturbations, nuclear data

I. Introduction

The CROCUS reactor is a zero-power critical facility used mainly for educational purposes at the Swiss Federal Institute of Technology (EPFL) in Lausanne.[1)] A set of critical measurements was carried out at CROCUS in 1993, in which the level of the light-water moderator was varied and the associated reactivity changes were determined from the measured reactor periods using the Inhour equation.[1)]

The purpose of this paper is to report the results of MCNPX[2)] calculations performed at the Paul Scherrer Institute (PSI) with a detailed three-dimensional model of the CROCUS facility, in which the moderator-level variations were modeled using different cross-section libraries and the resulting reactivity changes compared with experimental values derived from the period measurements. These results highlight differences between the ENDF/B-VI (release 0),[3)] ENDF/B-V,[3)] JEF-2.2[4)] and JENDL-3.2[5)] nuclear data, and their effects on the prediction of k_{eff} and the reactivity perturbations considered.

II. The CROCUS Facility

The CROCUS research reactor consists of a two-zone core positioned in a light-water filled aluminum vessel (**Fig. 1**). A horizontal drawing of the facility appears in **Fig. 2**, where the following features are identified: ① metal structure; ② reactor vessel; ③ upper grid plate; ④ support frame of upper grid plate; ⑤ outlet pipe; ⑥ safety valve; and ⑦ water level measurement device. Also visible are the fuel rods arranged in square lattices, with a pitch of 1.837 cm for the inner uranium oxide zone and 2.917 cm for the outer metallic uranium zone. A small water gap is present between the two fuel zones. The inner core contains 336 UO_2 fuel rods enriched to 1.806 wt% ^{235}U, while the outer core comprises either 172 uranium metal fuel rods with a ^{235}U enrichment of 0.947 wt%. The specifications of the fuel rods appear in **Table 1**.

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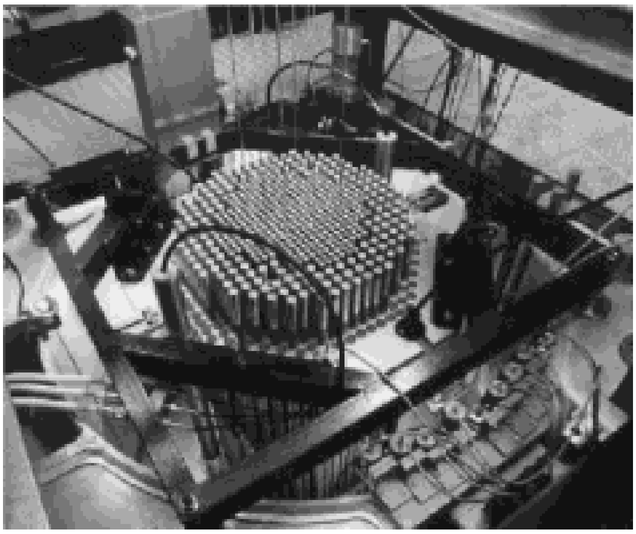

Fig. 1 The CROCUS zero-power facility at EPFL[1)]

The aluminum-clad fuel rods are supported by lower and upper aluminum grid plates, each incorporating a 0.5 cm-thick layer of natural cadmium. Axial dimensions relevant to the neutronic MCNPX model are given in **Fig. 3** (dimensions in millimetres). The reactivity of the core is regulated by varying the moderator level between 80 and 100 cm with an accuracy of ±0.01 cm. A description of non-fuel materials appears in **Table 2**. The facility operates at room temperature and is limited to a fission power of 100 W.

III. Modeling Details

Figures 4 and **5** show horizontal and vertical views of the detailed three-dimensional MCNPX model of the CROCUS facility. The details of the grid plates were modeled accurately, although some simplifications were introduced to the support structure below the core. The safety rods, which were fully withdrawn during the experiments, and their guide wires were not modeled. The model was terminated axially at the

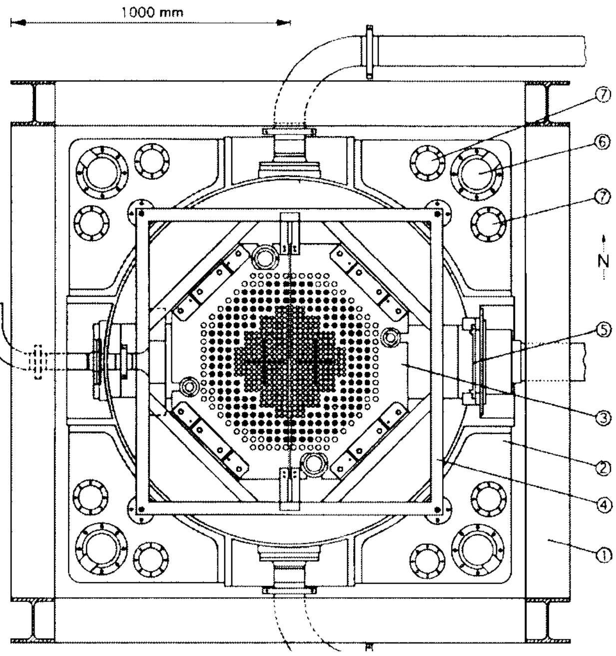

Fig. 2 Horizontal drawing of the CROCUS facility[1)]

Table 1 CROCUS fuel rod geometry specifications[1)]

	Inner fuel zone	Outer fuel zone
Fuel material	UO_2	U metal
Fuel density (g/cm^3)	10.556	18.677
Enrichment (wt% ^{235}U)	1.806	0.947
Fuel radius (cm)	0.526	0.85
Fuel length (cm)	100.0	100.0
Clad inner radius (cm)	0.545	0.8675
Clad thickness (cm)	0.085	0.1
Length of lower plug (cm)	2.7	2.7
Length of upper plug (cm)	17.3	17.3

Table 2 Description of CROCUS component materials

Component	Material (at%)	Density (g/cm^3)
Moderator	H_2O	0.9983
Base and grid plates End plugs Cladding	Aluminum 6060	2.702
Cadmium layers	Natural Cd	8.65
Filler gas	He	1.64×10^{-4}

top of the fuel rods.

The calculations were performed with MCNPX 2.4.0, a high-energy version of the code that incorporates MCNP4C3.[2,6)a] All cases were run to 100 million neutron histories using parallel processing in order to reduce the 1σ relative error in reactivity to less than $0.01\%\Delta k/k$. The calculations utilized stable fission source distributions and 1,000 active cycles of 100,000 neutron histories per cycle. The analysis was performed with four continuous-energy cross-section libraries at room temperature: ENDF/B-VI (release 0), ENDF/B-V, JEF-2.2 and JENDL-3.2. Thermal $S(\alpha, \beta)$ cross-sections were used for hydrogen in the light-water moderator.

IV. Results and Discussion

Direct calculations of the effective multiplication constant (k_{eff}) were performed for different moderator levels, in which the water level was raised from the measured critical height of 96.51 cm by 2, 2.49 and 3 cm. The resulting k_{eff} values are shown in **Fig. 6**.

The reactivity effects of the moderator height variations were calculated as:

$$\Delta\rho = \frac{1}{k_1} - \frac{1}{k_2}. \tag{1}$$

These static reactivities are compared with the experimental results in **Table 3** and **Fig. 7**. The experiments involved measurements of reactor period for reactivity perturbations from a critical state. The dynamic reactivity was calculated using the reactor period *via* the Inhour equation:

$$\rho = \omega\Lambda + \sum_{i=1}^{6} \frac{\beta_i \omega}{\omega + \lambda_i} \approx \sum_{i=1}^{6} \frac{\beta_i \omega_0}{\omega_0 + \lambda_i}, \tag{2}$$

where $1/\omega_0$ is the asymptotic reactor period.[7)] The kinetic parameters that appear in the above equation must be calculated. The experimental results given in Table 3 and Fig. 7 are based on kinetic parameters that were derived using delayed neutron data from JEF-1.[8)]

The results were found to be sensitive to the starting fission source distribution because of the small reactivity effects sought. (The perturbation option in MCNPX did not produce accurate results.) Therefore, each of the $\Delta\rho$ values given in Table 3 is an average of repeated k_{eff} calculations carried out with three different sources. A comparison of the k_{eff} predictions for the measured critical moderator height in the CROCUS facility shows a variation of approximately $0.7\%\Delta k/k$, which is attributable to differences between the nuclear data libraries used. The reactivity changes with moderator height are essentially independent of the cross-section libraries within the 1σ statistical error of the Monte Carlo calculation, although the values obtained are slightly higher than those derived from reactor period measurements. These over-predictions may have been caused by deviations of the calculated neutron flux from the critical fundamental mode, and by the methods and cross-section library dependence of the CROCUS kinetic parameters used for converting the period measurements to the "experimental" $\Delta\rho$ values. The latter is the subject of an OECD/NEA benchmark.[9)] The purpose of the current discussion is to inter-compare the various $\Delta\rho$ values calculated with MCNPX using different cross-section libraries.

Modeling approximations likely led to the general underprediction of k_{eff} by all libraries other than JENDL-3.2. However, it should be mentioned that the continuous-energy ENDF/B-VI, ENDF/B-V and JEF-2.2 neutron cross-sections

[a] MCNPX is used at PSI for both fission reactor and accelerator driven system applications.

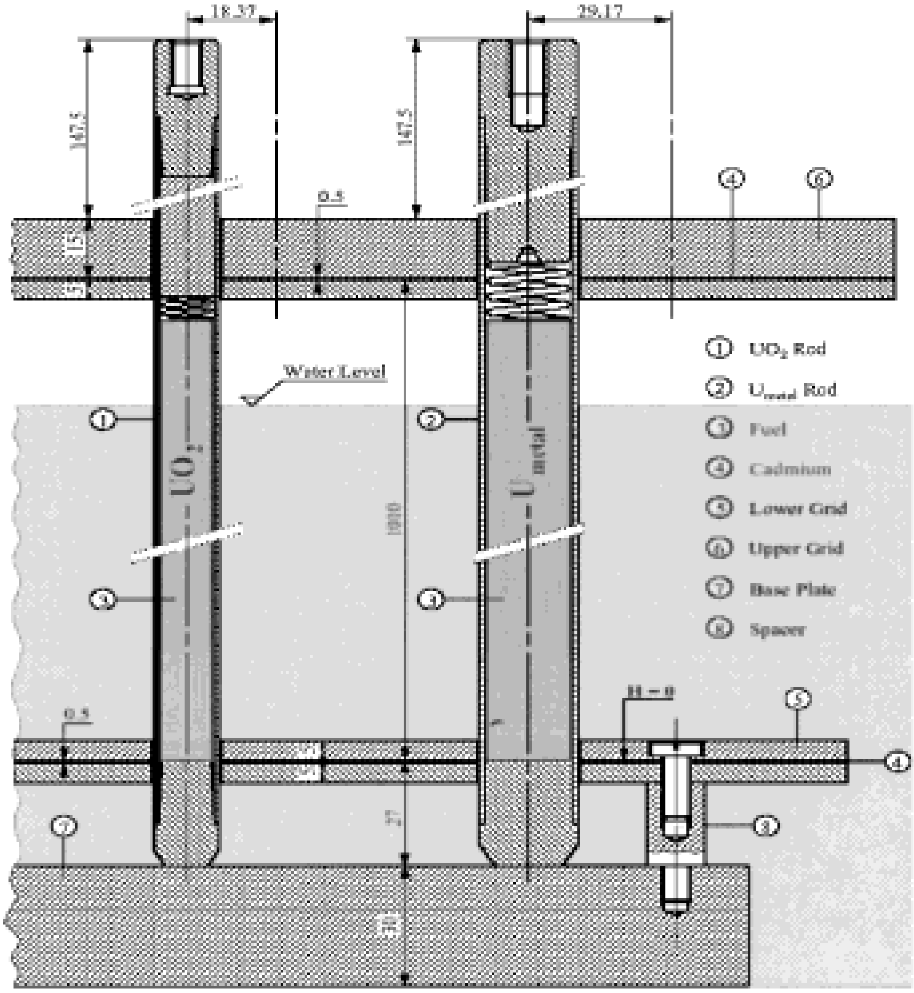

Fig. 3 Schematic view of the CROCUS fuel rods and grid plates[1)]

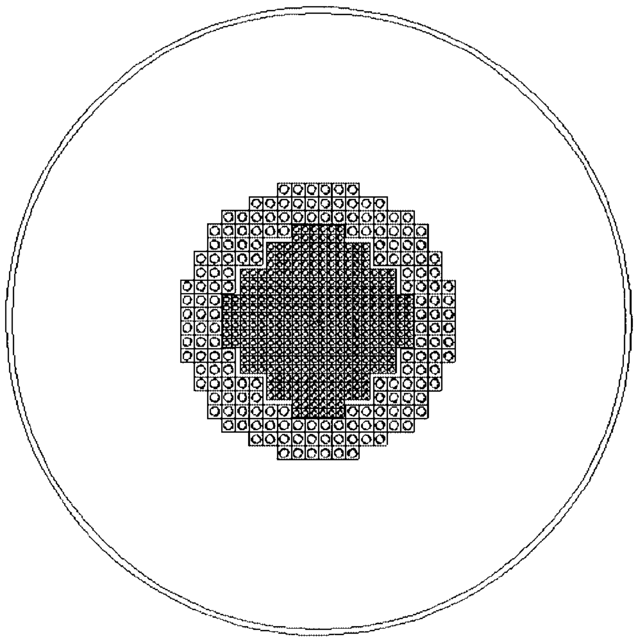

Fig. 4 Horizontal view of the CROCUS reactor model

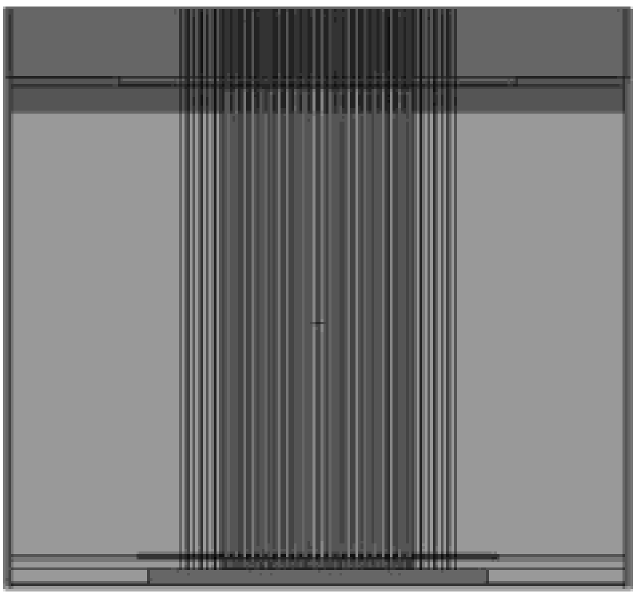

Fig. 5 Vertical view of CROCUS reactor model

Table 3 k_{eff} for the reference configuration and reactivity effects of moderator-level variations in CROCUS[a)]

Parameter	ENDF/B-VI	ENDF/B-V	JEF-2.2	JENDL-3.2	Average	Experiment
k_{eff}	0.99406±0.00002	0.99789±0.00003	0.99874±0.00003	1.00119±0.00004	0.99798±0.00002	1.00000
$\Delta\rho$ (2.00 cm)	96.6±2.4	90.6±3.0	97.5±2.9	96.7±3.1	95.8±1.4	89.2±0.6
$\Delta\rho$ (2.49 cm)	118.9±2.4	111.3±3.1	118.2±2.9	116.9±3.1	115.8±1.5	111.0±0.7
$\Delta\rho$ (3.00 cm)	138.0±2.4	133.7±3.1	138.8±3.1	135.8±3.1	137.3±1.5	131.7±0.8

[a)]Reactivities are given in units of pcm (0.001%$\Delta k/k$).

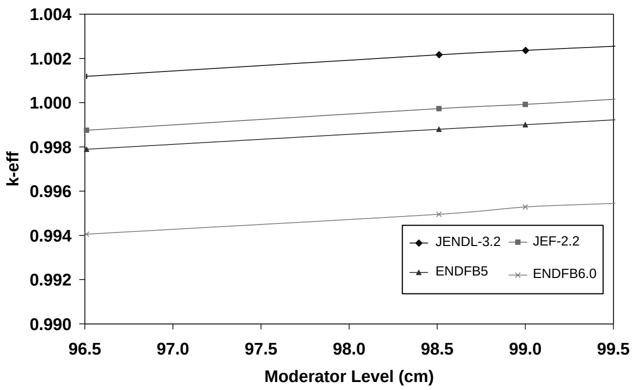

Fig. 6 Variation of k_{eff} with moderator level in CROCUS

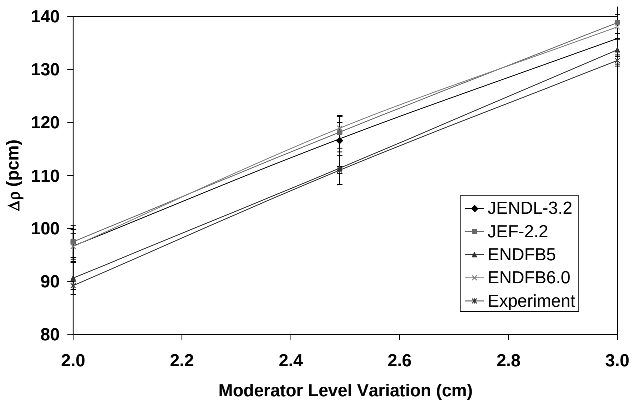

Fig. 7 Reactivity effect of moderator level variation in CROCUS

were generated at 300 K. The JENDL-3.2 library was prepared at 293 K, but used with the 300 K ENDF/B-VI thermal $S(\alpha, \beta)$ data. Since the experiments were carried out at an accurately regulated temperature of 293.0±0.1 K, the small temperature differences among these cross-section libraries also played a finite role. The effect was significant for the small reactivity changes resulting from moderator height variations. An average sensitivity of approximately 1.5 pcm/K was assessed for the $\Delta\rho$ values in Table 3. The higher k_{eff} obtained with JENDL-3.2 is related to the larger thermal-fission and smaller resonance-capture cross-sections for ^{235}U relative to other evaluations.[10)]

In conclusion, the results of this study demonstrate that the reactivity effects of moderator-level variations in a critical facility are largely independent of the nuclear data evaluation even when the associated k_{eff} values vary significantly.

Acknowledgments

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