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DEVELOPMENT OF THE COMPUTER PROGRAMS CORD AND DMR IN ZEL-EN PE GEN ENERGIJA – A STATUS REVIEW

RAZVOJ RAČUNALNIŠKIH PROGRAMOV CORD IN DMR V ZEL-EN PE GEN ENERGIJA – PREGLED STATUSA

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Keywords: nuclear reactor core design, digital reactivity meter, lattice code, nuclear data library, reactivity

Abstract

The planned activities and results achieved in the ZEL-EN's CORD and DMR reactor physics projects are reviewed. In the CORD project, new software tools for the numerical modelling of nuclear reactor cores in pressurized water reactors are being developed. The WIMS-D library update with nuclear data from the ENDF/B-VII.1 library was the first task in that project. The performed procedures and results of the WIMS-D benchmark test calculations with the WIMS-D libraries updated with the data from the ENDF/B-VII.0 and ENDF/B-VII.1 libraries are briefly presented. In the DMR project, a new digital reactivity meter for measuring the time-dependence of the reactivity in the reactor core is being developed. The new DMR software will run on current MS Windows systems and will be compatible with current electronic measuring equipment. Tests of the beta version of the new DMR software, presented in this paper, showed good agreement of the results for the measured reactivity with those obtained with the old DMR software, which is a validated MS-DOS application program and is considered as a reference tool.

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Povzetek

Predstavljen je pregled načrtovanih dejavnosti in doseženih rezultatov v ZEL-EN-ovih projektih na področju reaktorske fizike CORD in DMR. V projektu CORD bo razvita nova računalniška programska oprema za numerično modeliranje reaktorske sredice v tlačnovodnih jedrskih reaktorjih. Prva naloga v tem projektu je bila posodobitev knjižnice jedrskih podatkov programa WIMS-D z jedrskimi podatki iz knjižnice ENDF/B-VII.1. Na kratko so predstavljeni izvedeni postopki in rezultati referenčnih testnih izračunov s programom WIMS-D, ki so uporabljali knjižnice jedrskih podatkov posodobljene s podatki iz knjižnic ENDF/B-VII.0 in ENDF/B-VII.1. V projektu DMR bo razvit novi digitalni merilnik reaktivnosti za merjenje časovne odvisnosti reaktivnosti v reaktorski sredici. Programska oprema novega DMR bo delovala na sodobnih računalniških sistemih z operacijskim sistemom MS Windows in bo kompatibilna s sodobno elektronsko merilno opremo. Testi beta verzije programske opreme novega DMR, predstavljeni v tem prispevku, so pokazali dobro ujemanje rezultatov za izmerjeno reaktivnost z rezultati pridobljenimi s programsko opremo starega DMR, ki deluje na MS-DOS-u in velja za referenčno orodje.

1 INTRODUCTION

In this paper, the planned activities and results achieved in a development of updated versions of the CORD and DMR computer programs are reviewed. The development of the program CORD is being performed by ZEL-EN team members Dušan Čalić and Mladen Stanojević, and of the program DMR by ZEL-EN team member Slavko Slavič. These activities are being performed in collaboration with the Reactor Physics Division of the Jožef Stefan Institute.

The development of CORD is a part of ZEL-EN's reactor physics project, in which state-of-the-art software tools for numerical modelling of pressurized water reactors (PWRs) will be developed. These tools will enable more accurate prediction of nuclear reactor behaviour and, consequently, economically more efficient usage of the reactor fuel and safer operation. This represents a new business opportunity on the global nuclear energy market for the existing and new western-type PWRs that are planned or already being built. The expected final product will be a set of computer programs for the numerical modelling of western-type PWRs. In addition, new knowledge necessary for performing related services on the nuclear power plants with these computer programs will be obtained.

The development will be based on existing computer programs, which offer only some partial solutions. They will be upgraded with more detailed and accurate theoretical and computational models, and their results will be cross-checked and verified with other available computer programs and measurement results. The final product will have the following new capabilities:

- more precise modelling of the isotopic composition of the burned-up fuel due to explicitly taking into account the fuel cooling time between particular burning cycles;
- determination of the increase of the necessary time for project calculations (i.e. the core design calculations) due to more accurate burn-up modelling, and other improvements and optimizations of the models;
- improved lattice model calculations with the WIMS-D code, incorporated into the new CORD package, in particular, significant improvements in modelling MOX fuel (i.e. mixed oxide nuclear fuel that contains more than one oxide of fissile material, usually consisting of plutonium

blended with natural uranium, reprocessed uranium or depleted uranium) and burnable absorbers (especially gadolinium) included within the mixed fuel;

- reliable project calculations for various western-type PWRs.

The new CORD will be based on an existing software package, CORD-2, [1], which is already used for modelling and designing the reactor core in the Krško Nuclear Power Plant (KNPP). The CORD-2 includes reactor physics programs, WIMS-D, [2], and GNOMER, [3], and numerous libraries with various data and utility programs for manipulating these libraries and data (see Figure 1). An important part of this project, which is presented in this paper, is an update of the WIMS-D nuclear data libraries with the new data included in the ENDF/B-VII.1 nuclear data library. After the testing and validation of these libraries with standard benchmark tests for the WIMS-D, they will be used for modelling selected previous reactor fuel cycles of KNPP with CORD-2. Their validity will be verified by comparisons with relevant measured data for these fuel cycles.

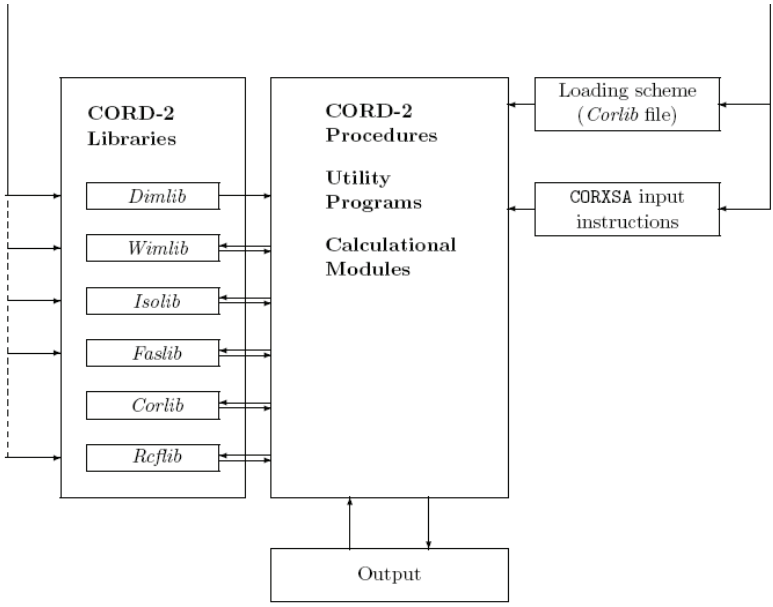

Figure 1: *Block diagram representing the CORD-2 system, [1]*

The development of the Digital Reactivity Meter (DMR) software [15] is a part of ZEL-EN's project in which a new generation of this measuring device will be developed. Digital reactivity meters are indispensable parts of the measuring equipment in nuclear energetics. However, the so-called rod-insertion method (or 'rod-in method') is mostly not allowed in the existing products available on the market, and additional functions for measuring and displaying measured data are not available. The new DMR will be applicable for physics tests in nuclear power plants, research reactors and for educational purposes, so it can be appealing for the nuclear energy market. It will be based on verified physical models, and numerical and experimental methods already applied into the existing DMR-043, [4] (see Figure 2), which is used in the start-up tests in Krško NPP and research on the TRIGA reactor at the Jožef Stefan Institute. The DMR-043 software and hardware use technologies developed in the 1980s, i.e. a computer program running on MS-DOS and analogue

electronic measuring equipment. From the measured neutron flux (proportional to the measured current (I) in Figure 2) and temperature (proportional to the voltage (V) in Figure 2, measured by the thermocouple) and with the point-kinetics equations, the time-dependence of the reactivity in various operating conditions of the reactor core, determined by a position of the control rods, the concentration of boric acid (i.e. neutron absorber) and temperature of the primary coolant is calculated. The purpose of these tests in the commercial reactors is to verify the control rod worth and reliability of the operation. The new DMR software will run on modern MS Windows systems (such as XP, Vista, 7, 8) and will have a graphical user interface (GUI), in addition to the old command line interface (CLI). It will be compatible with both previous analogue and modern digital electronic measuring equipment.

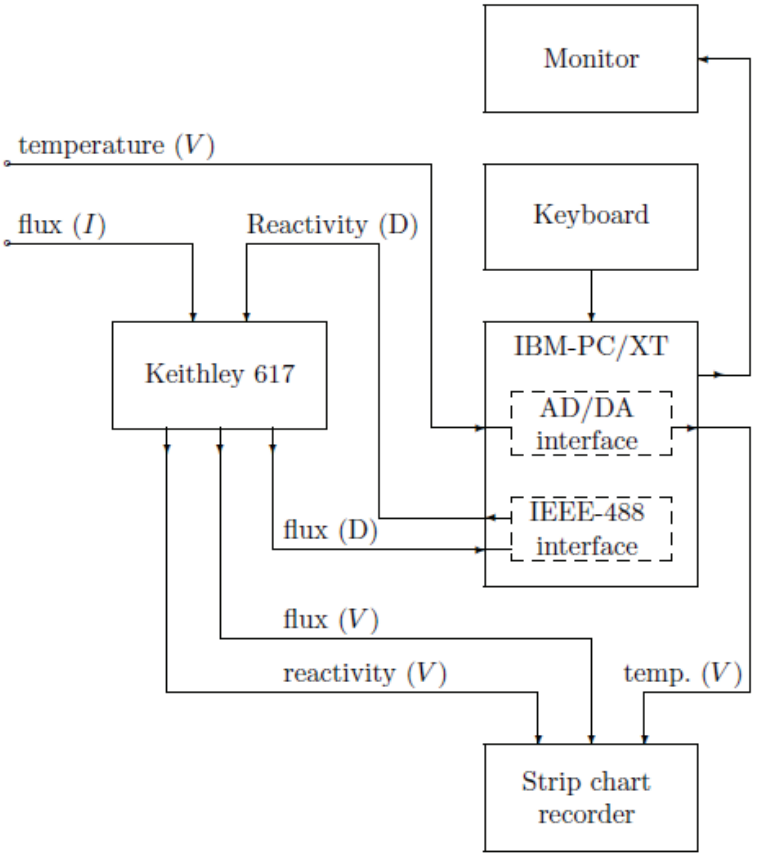

Figure 2: *Block diagram representing the digital reactivity meter DMR-043 hardware connections. (D) represents a digital signal, (V) represents an analogue voltage signal, and (I) represents an analogue current signal, [4]*

The present status of the new DMR and results of its first tests on the experimental TRIGA reactor at the Jožef Stefan Institute are presented in this paper.

2 METHODS

2.1 The WIMS-D library update with nuclear data from the ENDF/B-VII.1 library

The WIMS-D code is a reactor lattice code, [2], for solving the Boltzmann equation for neutron transport in simple geometries, such as a homogeneous medium, one-dimensional cylindrical, spherical and Cartesian geometry, and two-dimensional cylindrical geometry. It is used to calculate neutron flux distribution and neutron multiplication factor in an infinite medium, k_{inf}. Finite dimensions of the system and, consequently, a possibility that neutrons can escape from the system can be taken into account by specifying experimental or geometrical buckling as input parameter. In such cases, the effective multiplication factor for the finite system, k_{eff}, can be calculated. The WIMS-D takes as input the multi-group library of isotopic nuclear data and a description of the reactor lattice, and solves the neutron transport equation over a specified region of the reactor lattice. This region may be a unit cell or a macro-cell. Therefore, it includes methods for solving an appropriate set of equations for neutron flux and k_{inf} in a discrete energy and spatial mesh (i.e. energy groups and discrete spatial points). The calculated neutron flux may be used to calculate sets of neutron flux-averaged or flux-weighted macroscopic cross sections homogenized over chosen sub-regions and in a chosen broad energy group structure, [5]. These sets of macroscopic cross sections can be used as material data in the input for various codes solving the neutron transport equation or diffusion equation over the whole reactor or its fragment. In the CORD-2 package, this is done with the GNOMER code, which solves the neutron diffusion equation in three-dimensional Cartesian geometry with Green's function nodal method, [6]. The calculated neutron flux can also be used for reaction rate calculations or in fuel depletion calculations. A simplified flowchart of the WIMS-D calculations is presented in Figure 3.

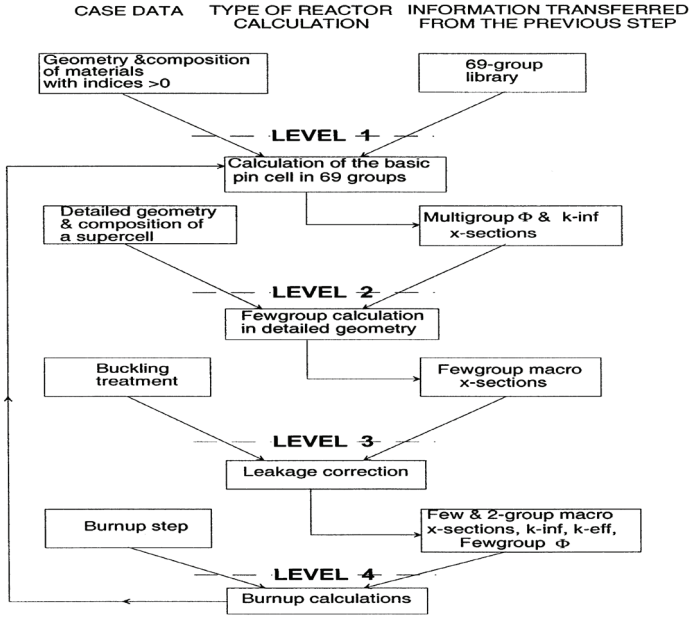

Figure 3: *Simplified flowchart of the WIMS-D calculations, [2]*

Updating the 69- and 172-group WIMS-D libraries with nuclear data from the ENDF/B-VII.1 library was the first task in the development of a new CORD package; this was accomplished by the author of this paper. Improvements of the fuel burn-up models are also being investigated, but this pending activity will not be presented in this paper. The WIMS-D code library update with nuclear data from the ENDF/B-VII.1 library could be considered as a continuation or supplement of the WIMS-D Library Update Project (WLUP), [7], which was supported by the International Atomic Energy Agency (IAEA). In that project, 69- and 172-energy group WIMS-D libraries based on nuclear data from the ENDF/B-VII.0 and other available nuclear data libraries were created and validated with the standard WIMS-D benchmark tests. Since a detailed presentation of this project and all relevant software is available on the WLUP web-page, [8], only changes of the input files and scripts that were used for processing input data based on the ENDF/B-VII.1 library are briefly described in this paper. The results of the benchmark calculations performed with the WIMSD-5B code and comparisons of the effective multiplication factors k_{eff} calculated with the WIMS-D libraries based on nuclear data from the ENDF/B-VII.0 and ENDF/B-VII.1 libraries are also briefly described.

The NJOY 99.364 code was used for processing ENDF files instead of the NJOY 99.65 code that was originally used in the WLUP. The NJOY 2012 is a new version of the NJOY Nuclear Data Processing System using Fortran-90/95 style, [9]. It includes all the capabilities of the NJOY 99 plus an ability to process evaluations, using the newer Reich-Moore Limited (RML) resonance format now allowed in ENDF files. This new version was not available for the calculations presented in this paper. The other codes, such as AVRFPY code for processing fission product yield and decay data, WILLIE code for managing the WIMS-D libraries, etc. were the same as those used in the WLUP (see page 63 in [7]).

The first step was to reproduce the WLUP results obtained with the ENDF/B-VII.0 library, as available on the WLUP web-page, and in that way to verify the reliability of the software package used in these calculations. In the next step, it was necessary to modify some input files and scripts, originally used in the WLUP, in order to process the ENDF/B-VII.1 library data. Then, after executing the benchmark calculations, the results of the benchmark tests were compared and validated.

The WIMS-D library updating under MS-DOS on PC is described in detail on the WLUP web-page. The input files for the NJOY and WILLIE codes are also available therein. Calculations of the 69- and 172-group WIMS-D libraries were performed by executing the batch script that runs all the calculations, including the benchmark tests. However, during the processing of the original nuclear data from the ENDF/B-VII.0 library with the NJOY 99.364, several errors were reported by the latter. Therefore, the following changes of the original input files had to be made in order to avoid these errors:

- minor changes of the co-variances of resonance parameters for thorium isotope Th-232 (ENDF material 9040);
- missing data in the ENDF/B-VII.0 library for energy distributions of delayed neutrons for U-232 (ENDF material 9219), Am-242m (ENDF material 9547), Cm-243 (ENDF material 9634) and Cm-244 (ENDF material 9637) had to be replaced with the corresponding data for U-234 (ENDF material 9225), Am-243 (ENDF material 9549), Cm-242 (ENDF material 9631) and Cm-245 (ENDF material 9640).

For the latter task, several ENDF pre-processing codes, which assure correct replacement of the data and correct format of the created ENDF data files, were used in a rather tedious procedure. After these changes had been implemented, the calculations of the WIMS-D libraries and benchmark

test calculations were successfully completed with the set of WLUP programs and scripts.

To calculate the WIMS-D libraries with the data from the ENDF/B-VII.1 library, appropriate input data files that include all WIMS-D materials should be prepared. In addition, the following changes of the NJOY and WILLIE input files and the batch script that runs all the calculations had to be made.

The NJOY and WILLIE input files with names 'endfb7...' were renamed to 'endfb71...'. In the renamed NJOY input files 'endfb71.nji' (for calculating the 69-group library) and 'endfb71gx.nji' (for calculating the 172-group library), the input data for natural vanadium were replaced by the input data for vanadium isotopes V-50 (ENDF material 2325) and V-51 (ENDF material 2328). In the renamed WILLIE input files 'endfb71.wli' and 'endfb71gx.wli', the natural vanadium had to be specified as a mixture of these two isotopes. Because of these changes, the batch script that runs all the calculations was also changed in the part relevant to processing of vanadium isotopes.

Examination of the original ENDF/B-VII.1 input data showed that the data for energy distributions of delayed neutrons were missing for the isotopes listed above (Am-242m, Cm-243 and Cm-244), so they were replaced with the corresponding data for Am-243, Cm-242 and Cm-245, using the same procedure as described above. However, the most serious problem was found while processing Cl-35 (ENDF material 1725) with the NJOY 99.364, because this material was written in the new ENDF format that can be processed by the NJOY 2012. Therefore, Cl-35 was first pre-processed with LINEAR and RECENT codes, which are available at the ENDF web-page, [10], and then with the 2013-versions of ENDF pre-processing codes STANEF, CHECKR, FIZCON and PSYCHE codes. Since the NJOY 99.364 could not process the resonance parameters for the Reich-Moore theory, they were replaced by a simple model without resonance parameters except for the scattering radius, as used in the ENDF/B-VII.0 library. The co-variances of resonance parameters were also deleted. After all these changes had been implemented, the calculations of the WIMS-D libraries and benchmark test calculations were successfully completed.

Each set of the calculations for the 69- and 172-group libraries, including almost 300 benchmark test cases, took about three days on Intel Core i7-3630QM CPU 2.40 GHz system with 8 GB RAM. The necessary changes of the input files due to the processing errors mentioned above were mostly not known in advance, so several test runs had to be executed before discovering all these errors and finding appropriate solutions.

2.2 Development of the new DMR ('DMReS')

The theoretical basis for the design of the DMR is the point kinetics equation, [4], [11]:

$$\rho = \frac{\Lambda}{T(t)}\frac{dT}{dt} + \sum_{i=1}^{I}\frac{\beta_i}{T(t)}e^{-\lambda_i t}\int_0^t \frac{dT}{dt'}e^{\lambda_i t'}dt' - \Lambda\frac{Q(t)}{T(t)} \tag{2.1}$$

where ρ = (k-1)/k is the reactivity (*k* is the neutron multiplication factor), Q is the neutron source strength, β_i is the effective delayed neutron fraction of group *i* of the delayed neutrons (i.e. the ratio of the number of delayed neutrons from precursor group *i* to the total number of fission neutrons ν, usually 6 groups are used), λ_i is the decay constant of group *i* of the delayed neutrons, Λ is the prompt neutron lifetime and T is the strongly time-dependent neutron flux amplitude function, defined with equation:

$$\Phi(\vec{r},t) = T(t)S(\vec{r},t); \qquad \int_V S(\vec{r},t)dV = 1 \tag{2.2}$$

that splits the total neutron flux $\Phi(\vec{r},t)$ at position $\vec{r}$ and in time t into the strongly time-dependent neutron flux amplitude function $T(t)$ and the shape function $S(\vec{r},t)$, which varies slowly with time and is normalized such that the integral over the core is constant. It is assumed that Q, β_i, λ_i and Λ are averaged over the core and constant. The source strength Q is the only non-homogeneous term in the point-kinetics equation (2.1). If the neutron source strength Q contribution to the signal is assumed constant, it can be measured quite easily while approaching criticality, [12]. Note that on the right side of equation (2.1) the average neutron flux and its derivative appear as a ratio, except in the neutron source contribution. For this reason, we do not need to know the neutron flux in absolute terms, so the measured signal that is proportional to the neutron flux can be used, except for the neutron source contribution term, where the ratio of the neutron source signal and the fission neutron flux signal are required.

A recurrent numerical algorithm was derived to solve equation (2.1) and was implemented in DMR-043. The point kinetics equation (2.1) is valid exactly if either the reactor core is homogeneous, so that the delayed neutron data are the same for the whole core, or if the neutron flux distribution shape function $S(\vec{r},t)$ does not change significantly during the transient. However, experience and numerous measurements have shown that these restrictions alone do not limit the use of this equation for practical applications, [11].

The principle of the rod-insertion method is to start from a critical reactor operating at low power and to measure the time-dependent reactivity change while a control rod is inserted into the core. Unlike in the rod-drop method, the measured control rod is inserted with the drive mechanism at normal speed. By analysing the flux trace using the point-kinetics, not only the total rod worth but also the differential and the integral control rod worth curves are obtained. During the rod-insertion measurement, the flux may drop by several orders of magnitude. Therefore, a high-quality electrometer is required for monitoring the neutron flux. The analysis is performed by transferring the data to a computer with the DMR software.

The new DMR software is being developed by S. Slavič [15] as a stand-alone 32-bit MS Windows application program running on MS Windows XP or newer operating systems (e.g., Vista, 7, 8, etc.). It was developed in the Xbase++ programming language, with the Alaska Xbase++ toolkit. It has a GUI (see *Figure 4*), with additional pre- and post-processing features that are not included in the DMR-043, such as saving all configuration files and results as text data files (by default), saving all projects or measurement scenarios in a way that they can easily be reproduced or re-run, exporting plots as JPG- or GIF-files, simple (auto) or advanced input options, etc. In addition, the programs INHOUR and BORDIL, [13], are included in the software package and can be run with the GUI. These two MS-DOS application programs (i.e., the '.exe'-binaries) have previously been developed, tested and verified. They have not been ported to MS Windows, so they are not included in the new DMR software source code. The INHOUR and BORDIL utilities are used to approximately relate the measured power doubling time (or reactor period) to the reactivity and to relate the boron concentration in the primary circuit to the added volumes of water or concentrated boron solution, respectively. The Inhour equation can be derived from the point kinetics equations, [14], if the neutron source term Q is neglected, the reactivity ρ is constant and asymptotic exponential solution for the neutron flux is assumed (i.e. $\Phi(t) \propto e^{\omega t}$). Assuming that there are six delayed-neutron precursor groups, the Inhour equation can be written as:

$$\rho = \omega\Lambda + \sum_{i=1}^{6} \frac{\omega\beta_i}{\omega + \lambda_i}, \tag{2.3}$$

where $\omega = \ln 2/t_d$ is the inverse reactor period and t_d is the reactor doubling time and other quantities in equation (2.3) are the same as defined above.

The beta version of the new DMR software, called 'DMReS', was initially tested with simulated measured signals for the neutron flux and temperature, which were generated by an electronic circuit made for that purpose, and then with the measured signals from the TRIGA experimental reactor at the Jožef Stefan Institute. The results of these tests are presented in the next section. It was also verified that DMReS is operational with both a Keithley 617 analogue electrometer (1980s technology) and a Keithley 6517 digital electrometer (late 1990s–early 2000s technology). The development of DMReS is still in progress, because some GUI errors must be corrected, and the usability of some of its functions must be improved.

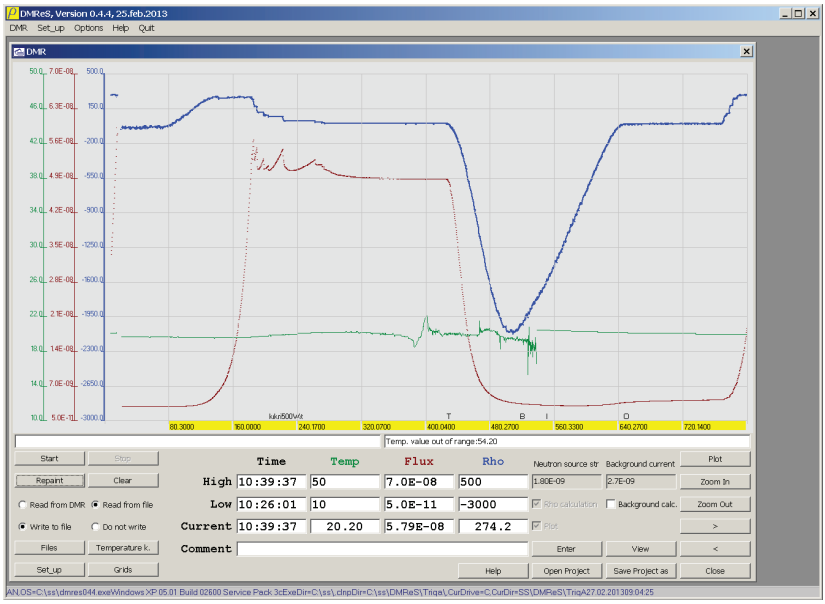

Figure 4: *Screen-shot of the DMReS GUI*

3 RESULTS

3.1 The WIMS-D benchmark test calculations

Detailed analyses of the results of the benchmark calculations with the updated WIMS-D libraries were performed in order to confirm the validity of these libraries for the planned calculations with the CORD-2 package. The benchmark calculations should reproduce the reference values of the effective multiplication factor k_{eff} for various experimental test cases that include different geometry parameters and combinations of nuclear fuel, cladding, moderator, neutron absorber and structural materials, which are relevant to commercial nuclear fuel systems. Considering the possible

differences between the input data and versions of the NJOY code used in the calculations with the ENDF/B-VII.0 library data presented in this paper and those available on the WLUP web-page, they provided almost the same results for the k_{eff}s. The differences between the k_{eff}s were mostly of the order of 10^{-5} (1 pcm) and only in a small number of test cases were they of the order of 10^{-4} (10 pcm). Thus, the reliability of the software package for updating the WIMS-D libraries was verified with these calculations.

Comparisons of the k_{eff}s calculated with the data from the ENDF/B-VII.0 and ENDF/B-VII.1 libraries showed the following trends and characteristics:

in the benchmark tests with uranium fuel, which do not include some structural materials, in particular natural chromium (Cr-nat), manganese (Mn-nat) and nickel (Ni-nat), and natural zirconium (Zr-nat) as the fuel cladding material, where the differences between the nuclear data in the ENDF/B-VII.0 and ENDF/B-VII.1 libraries are significant, the k_{eff}s calculated with the ENDF/B-VII.1 data are mostly smaller than the k_{eff}s obtained in the analogous calculations with the ENDF/B-VII.0 data; the differences between these k_{eff}s are from the order of 10^{-5} (1 pcm) to the order of 10^{-4} (10 pcm);

in the benchmark tests with uranium fuel, which include the aforementioned structural materials and natural zirconium, the k_{eff}s calculated with the ENDF/B-VII.1 data can be smaller or bigger than the k_{eff}s obtained in the analogous calculations with the ENDF/B-VII.0 data, depending on the content of the structural materials and zirconium, the differences between the k_{eff}s are from the order of 10^{-4} (10 pcm) to the order of 10^{-3} (100 pcm);

in the benchmark tests with a fuel containing a mixture of uranium and plutonium isotopes (MOX), the k_{eff}s calculated with the ENDF/B-VII.1 data are mostly smaller than the k_{eff}s obtained in the analogous calculations with the ENDF/B-VII.0 data; the differences between the k_{eff}s are from the order of 10^{-4} (10 pcm) to the order of 10^{-3} (100 pcm), and because they are also strongly dependent on the content of the structural materials and zirconium, qualitatively different results for the k_{eff}s are obtained in some cases, e.g. in Rowlands Pu Pin-cell benchmarks; [7]

in the benchmark tests with thorium fuel, the k_{eff}s calculated with the ENDF/B-VII.1 data are smaller than the k_{eff}s obtained in the analogous calculations with the ENDF/B-VII.0 data; the differences between the k_{eff}s are from the order of 10^{-4} (10 pcm) to the order of 10^{-3} (100 pcm) and they are also dependent on the content of the structural materials, zirconium (Zr-nat) and cadmium (Cd-nat);

changes of the dimensions and buckling of the system can significantly reduce the influence of the differences between the ENDF/B-VII.0 and ENDF/B-VII.1 data on the results for the k_{eff}s while keeping the same content of the materials used in the benchmark tests.

Therefore, it can be summarized that in about 90% of the test cases the k_{eff}s calculated with the ENDF/B-VII.1 data are smaller than the k_{eff}s obtained in the analogous calculations with the ENDF/B-VII.0 data. However, the important exceptions from this 'rule' are found in some MOX-fuel benchmark calculations (for example, in Rowland's Pu Pin-cell benchmarks), which are relevant to the reactor fuel burn-up calculations.

3.2 Tests of DMReS with simulated and measured signals from the TRIGA reactor

Comparisons of the time-variations of the neutron flux and reactivity obtained with DMReS and DMR-043 using simulated measured signals are presented in Figures 5 and 6, respectively. The 'DMR-043+'-label indicates the measurement with the DMR-043 using higher sampling rate.

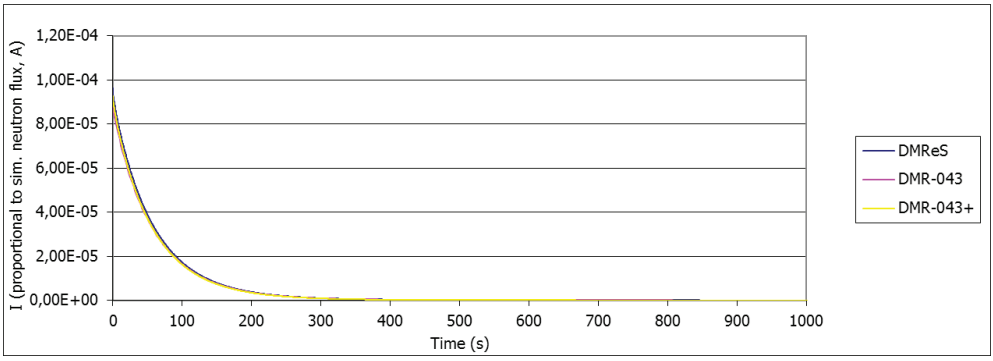

Figure 5: *Time-variation of the measured simulated neutron flux*

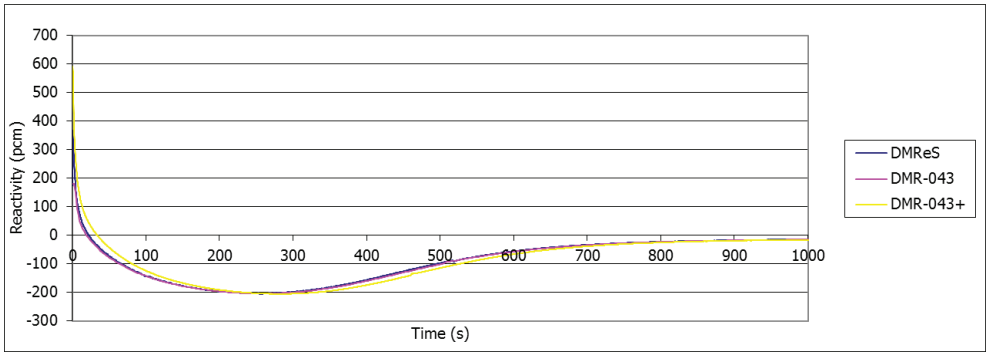

Figure 6: *Time-variation of the reactivity calculated with the measured simulated neutron flux*

It can be concluded from these results that the relative differences between the reactivities calculated with DMReS and DMR-043 are extremely small, i.e. up to a few percentage points or within the experimental error. However, the sampling rate or the density of the measured data can be extremely important, especially when the time-derivative of the neutron flux is large. In such cases, the relative differences between the reactivities calculated with either DMReS or DMR-043 using significantly different number of measured data can be from about 10 to 50%. Arbitrary changes of the sampling rate in the measurements in the real reactors are not possible, so the dependence of the numerical algorithm implemented in DMReS on the sampling rate must be minimized as much as possible, at least for processing realistic measured signals. This 'fine-tuning' of the free parameters of the numerical algorithm remains a pending task in the further development of DMReS.

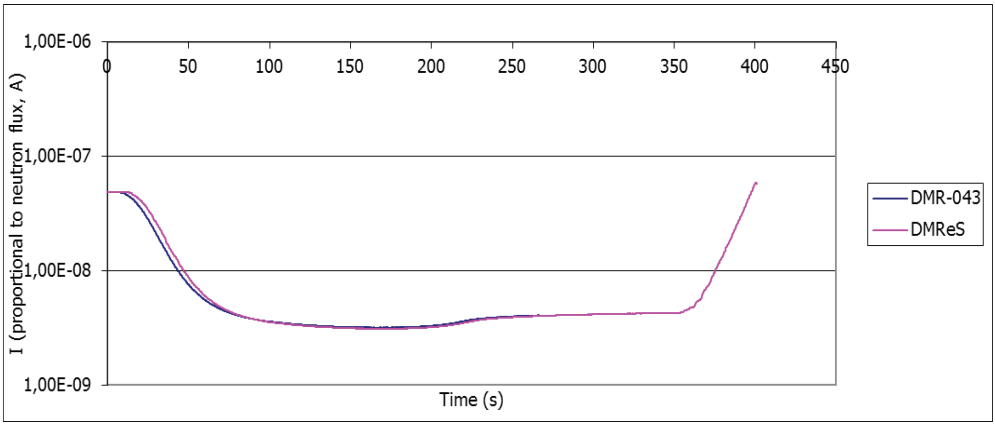

Figure 7: *Time-variation of the neutron flux measured in the TRIGA reactor*

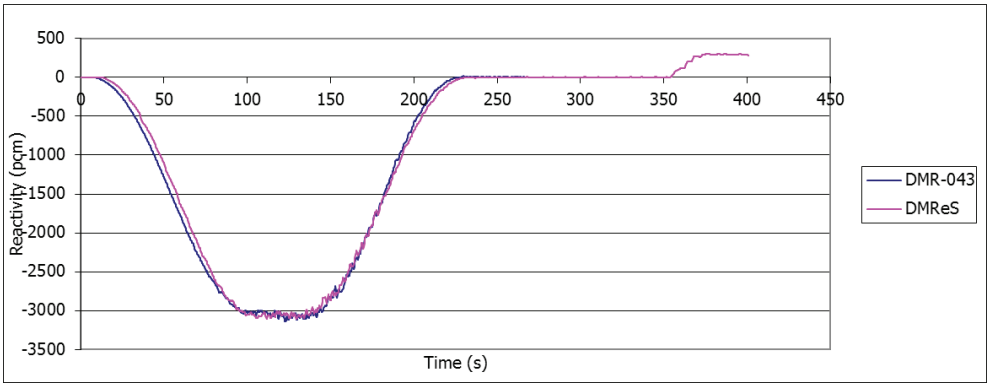

Figure 8: *Time-variation of the reactivity calculated with the neutron flux measured in the TRIGA reactor*

The rod-in method was used in the reactivity measurements on the TRIGA reactor with DMReS and DMR-043. Comparisons of the time-variations of the neutron flux and reactivity obtained with DMReS and DMR-043 are presented in Figures 7 and 8, respectively.

It is obvious from Figures 7 and 8 that the small differences between the measured neutron fluxes at the beginning of the experiment (i.e. in the first 100 s) have a strong influence on the results for the reactivity, because it is a time-interval in which the time-derivative of the neutron flux is large. These initial differences between the measured neutron fluxes occurred because simultaneous measurements of the neutron fluxes under identical initial conditions were not possible due to technical reasons (measured current from the ionization detector could not be simultaneously transferred to both DMR systems) and limitations of the experimental set-up. Later, the variations of the neutron flux were much slower, the differences between the measured neutron fluxes were extremely small; consequently, the relative differences between the calculated reactivities were a few percentage points within the experimental error.

4 DISCUSSION

The 69-group WIMS-D library based on the nuclear data from the ENDF/B-VII.1 library will be used in the KNPP core design calculations with the CORD-2 package, from the first to the 25th fuel cycle. The results of these calculations will be compared with the corresponding results of the calculations using the '1986' WIMS Nuclear Data Library, which is included in the WIMS-D5 code package and available experimental results. If better agreement with the experimental results is obtained with the new WIMS-D library, it will be used in the future KNPP core design calculations. This will also give additional motive to repeat the calculations of the new WIMS-D libraries with the NJOY 2012 code. Following the results of the WIMS-D benchmark calculations presented in this paper and considering the significant differences between the old and new nuclear data for zirconium and some structural materials, it may be expected that certain differences between the results obtained with the '1986' and new WIMS-D library will appear. However, because of the importance of the buckling parameter, it is not possible to predict how large they will be, as has been shown in some test cases in which changes of the dimensions and buckling of the system significantly reduced the differences between the calculated k_{eff}s. When the improved fuel burn-up model is developed and implemented in the CORD-2 package, the influence of the differences between the data in various WIMS-D libraries on the results of the burn-up calculations will also be investigated.

The first tests of DMReS, presented in this paper, showed good agreement of the results for the reactivity with those obtained with the DMR-043, which is considered to be a reference tool. However, additional development and tests of the updated software releases are necessary, in order to improve its precision, usability and quality of the GUI. These activities are already in progress. A patent application for the new DMR system has also been planned.

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Nomenclature

k	neutron multiplication factor
k_{inf}	neutron multiplication factor in an infinite medium
k_{eff}	effective neutron multiplication factor for a finite system
$\rho = (k-1)/k$	reactivity
t	time
$\vec{r}$	spatial position vector
Q	neutron source strength
β_i	effective delayed neutron fraction of group *i* of the delayed neutrons (i.e. the ratio of the number of delayed neutrons from precursor group *i* to the total number of fission neutrons, usually 6 groups are used)
λ_i	decay constant of group *i* of the delayed neutrons
Λ	prompt neutron lifetime
$\Phi(\vec{r},t)$	total neutron flux at position $\vec{r}$ and in time t
$T(t)$	strongly time-dependent neutron flux amplitude function
$S(\vec{r},t)$	neutron flux shape function, which varies slowly with time and is normalized such that the integral over the core is constant
t_d	reactor doubling time
$\omega = \ln 2/t_d$	inverse reactor period