COUPLING MCNP AND A DEPLETION CODE FOR DETAILED NEUTRONIC ANALYSIS AND OPTIMUM CORE MANAGEMENT AT THE GERMAN FRJ-2 RESEARCH REACTOR

R. NABBI and J. WOLTERS

Research Center Juelich
Leo-Brandt-Straße
52425 Juelich, Germany

Keywords: MCNP, Burnup Recycling, Fuel Management, Flux Depression, Absorber Effectiveness

ABSTRACT

By coupling MCNP and a depletion code a sophisticated tool was developed for detailed neutronic analysis, optimum core management and reactor utilization. The MCNP model consists of a detailed geometrical description of the reactor core and surroundings with 11 250 material cells. The fuel elements, absorber arms and beam tubes were highly segmented. The accuracy of the model was verified by the simulation of criticality experiments and comparison with the measured distribution of average and local n-flux. The reactivity value of each individual core configuration at any burnup step could be predicted within 2σ standard deviation. The local flux was determined with a deviation of less than 4 %. In addition to the routine application for fuel management, the method was applied to determine the n-flux and fission power in the fuel tubes as well as to study the burnup behavior of the absorber blades with regard to the change of effectiveness and lifetime.

1. BACKGROUND

Fuel management at the German FRJ-2 research reactor is handled on the basis of the experimental method consisting of reactivity and n-flux measurement. Before and during refueling of the core, the estimated excess and shutdown reactivity is checked by the inverse kinetic method. Due to the complex character and uncertainty of the measurements, conservative values are applied for reactivity balances, fuel element burnups and power distribution in the reactor core. To overcome the shortcomings of the core management and to reduce the experimental demand, a sophisticated calculational method is required for an exact reactivity balance and optimization of fuel management at FRJ-2 (Nabbi,1998 and 1999).

The need for such a code also arises from the fact that the measurement of neutronic details is not possible in all cases due to the limitation of the measuring devices and the inaccessibility of the desired core positions. In addition the safety parameters and neutronic characteristics of the core configurations set up for the next operating cycle cannot be measured in advance. In this respect, the prediction of local flux and fission power in the individual fuel plate becomes significant with regard to the heat load limits. The determination of neutronic details and characteristics in the beam tubes for irradiation purposes and beam experiments also requires a reliable and sophisticated calculational method for optimum utilization.

With regard to the safety character of the absorber system the application of the intended model would allow an accurate determination of the effectiveness of the absorber plates by calculation of the n-flux profile and depletion rate in the absorber plate. In view of these objectives a computational method was developed on the basis of the MCNP Monte Carlo code and a depletion code at FRJ-2.

The MCNP Monte Carlo code has been widely used for many years now in nuclear engineering to perform complex criticality studies and calculations (Briemeister, 1997, Rahnema, 1997). It is capable of treating any 3-dimensional configuration of materials in geometric cells of complex form using pointwise continuous-energy cross sections existing for a variety of reactions (Mosteller, 1998). The numerical models and features of the code have been extensively validated on the basis of comprehensive benchmark tests and experiments (Brockhoff,1994). Due to its suitability for modeling the complex material geometry of the fuel elements and the absorber arms of the high-power research reactor FRJ-2, it has been chosen for core physics analysis and criticality calculations. For the simulation of changes in material composition taking place during operation, MCNP was coupled with a depletion code and applied for core calculations by recycling the burnup and flux calculations. The present paper describes the details and verification of the model and the simulation of the neutronic and criticality behavior of FRJ-2.

2. DESCRIPTION OF FRJ-2

The FRJ-2 is a DIDO-class tank-type research reactor cooled and moderated by heavy water. The core consists of 25 so-called tubular MTR fuel elements (FE) arranged in five rows of 4, 6, 5, 6 and 4 fuel elements (Fig. 1). lt is accommodated within an aluminum tank 2 m in diameter and 2 m in height. The tank is surrounded by a graphite reflector 0.6 m in thickness enclosed within a double-walled steel tank.

The active part of the tubular fuel elements is formed by four concentric tubes having a wall thickness of 1.5 mm and a length of 0.63 m. The tubes consist of fuel meat clad with pure aluminum and are accommodated in a shroud tube 103 mm in diameter. The fuel meat contains UAlx in an aluminum matrix with a U235 enrichment of 80 %. The annular water gap between the tubes has a width of about 3 mm leaving a central hole of 50 mm diameter filled with a thimble for irradiation purposes.

The reactor is equipped with two independent and diverse shutdown systems, the coarse control arms (CCAs) and the rapid shutdown rods (RSRs). In case of demand, the six CCAs are released from their electromagnets and drop into the shutdown position by gravity, whereas the three RSRs are shot in by their pneumatic actuators. The CCAs are lowered and raised manually around a pivot in order to control power levels during normal operation, whereas the RSRs are permanently in their upper position. The shutdown position of the CCAs is at an angle of 56 ° against the horizontal line. In this position, they have the highest worth. In the case of a failure of the holding mechanism of one CCA, the sword would swing out of the core causing a high amount of reactivity insertion. The absorber plate of the CCAs contains cadmium of natural isotopic composition.

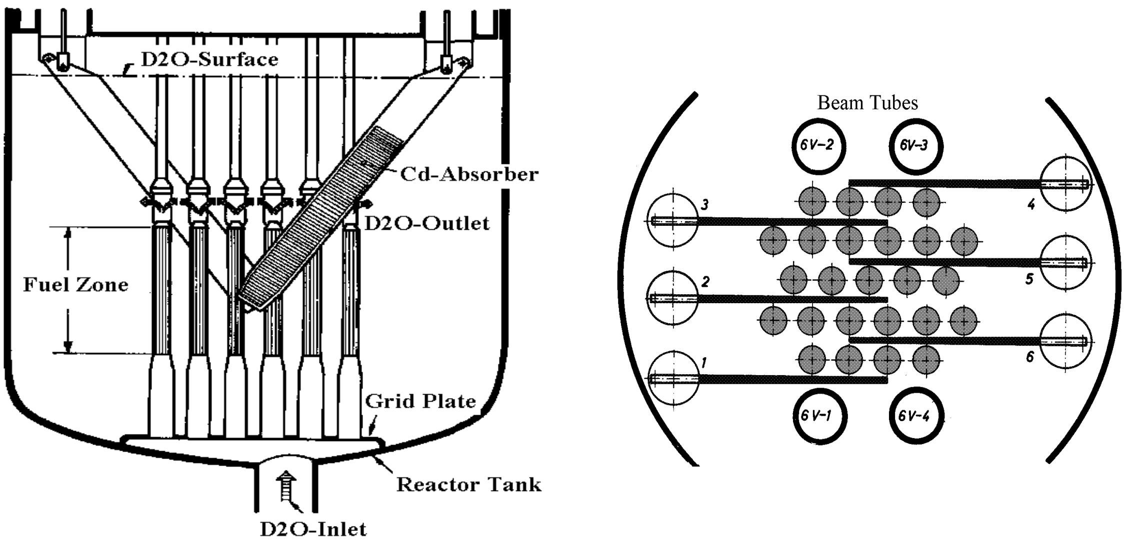


Fig. 1: Arrangement of fuel elements and coarse control arms inside the reactor tank

A large number of horizontal and vertical channels give access to the neutron field in the reactor. The horizontal channels (beam tubes) end either at the tank wall or at the periphery of the core by penetrating into the reactor tank.

3. MCNP MODEL OF FRJ-2

The MCNP model of FRJ-2 is a complete 3-dimensional full-scale model with a very high level of geometric fidelity. It comprises the reactor core, CCAs, core structures, beam tubes, the graphite reflector and the biological shield. The core region consisting of 25 fuel elements was modeled as a cylinder containing a square lattice with an array of cells representing the individual fuel elements. Each cell in the lattice contains a detailed model of each fuel element comprising the internal thimble, 4 circular

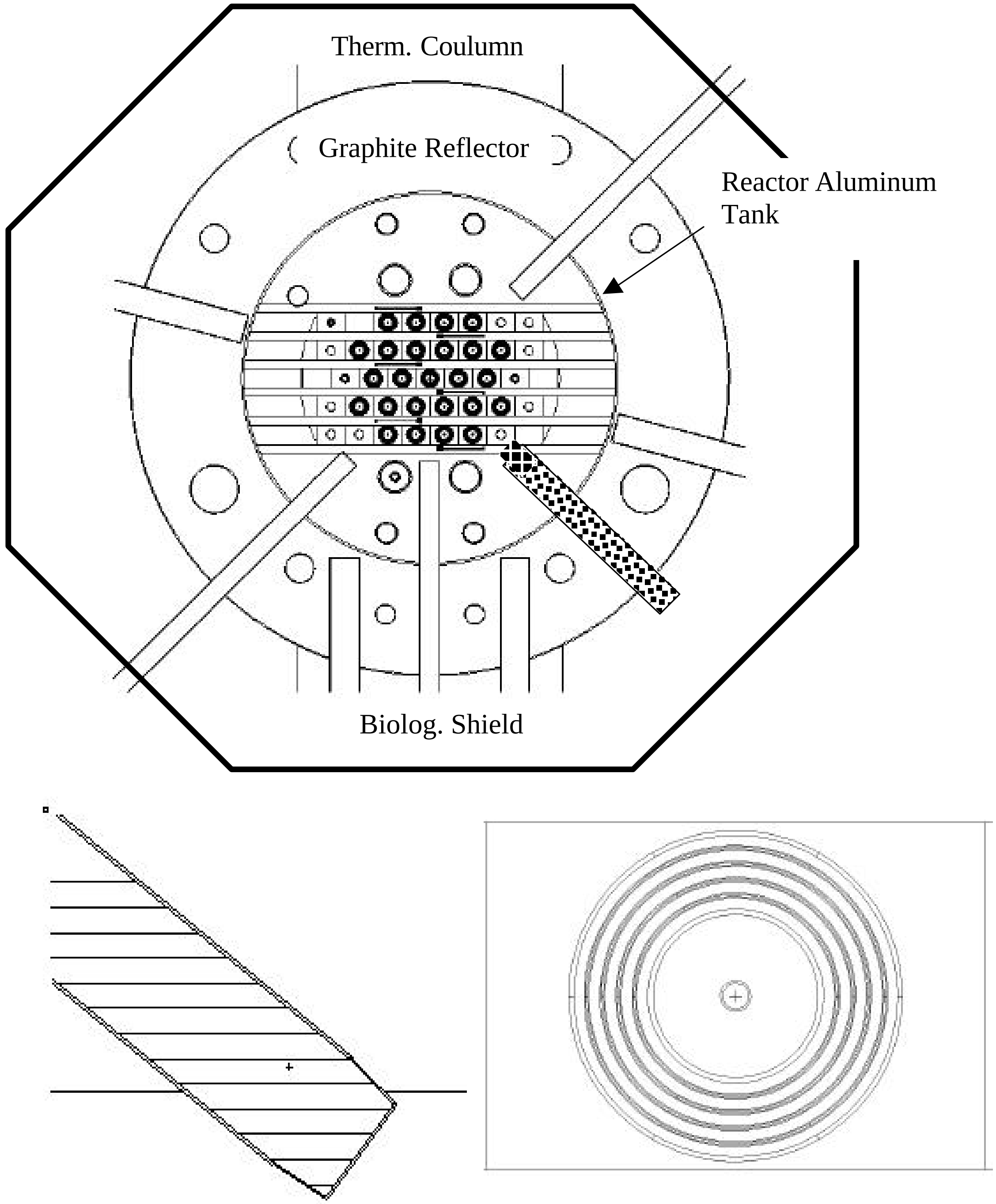


Fig. 2: Cross section of the MCNP model of FRJ-2 with the fuel element assembly and beam tube including:

- Fuel element model representing fuel tubes and outer borated tube
- Vertical orientation of a segmented arm (CCA)

fuel tubes and the borated outer shroud tube. Each individual cell is divided into 15 axial and 35 radial and azimuthal material zones.

The D2O reflector in the reactor aluminum tank (RAT) was represented by a cylinder with an outer diameter of 2.00 m. The lower region of the core down to the bottom of the RAT accommodating the grid plate, unfueled ends and nozzles of the fuel elements, the aluminum structures and corresponding D2O were modeled in detail. In the whole geometric model, the cell boundaries were specified by 1st and 2nd degree surfaces with appropriate transformation in accordance with the position of the cells in the model.

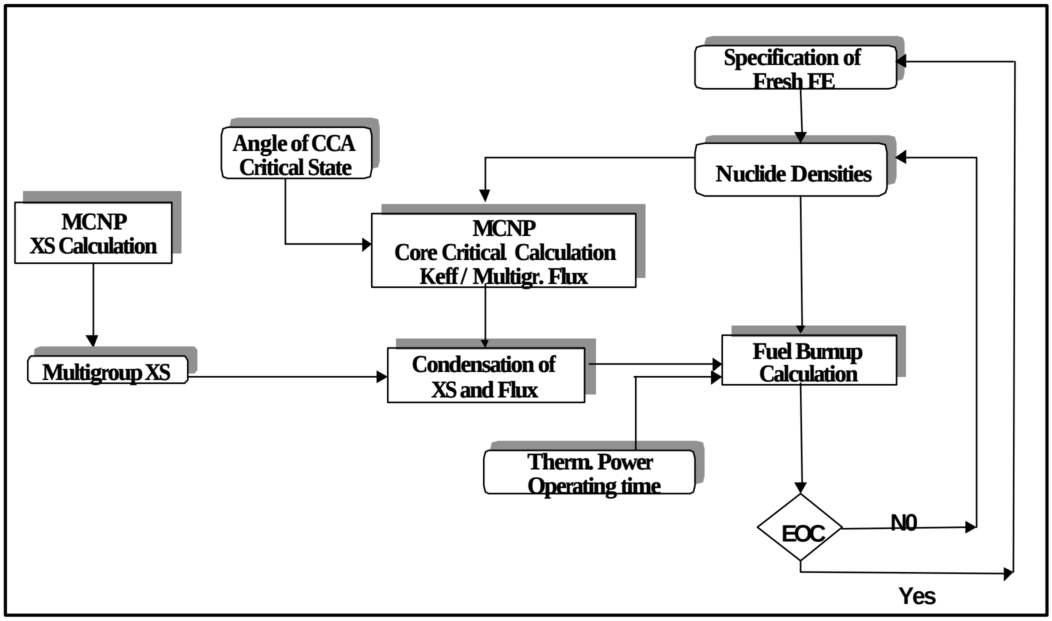
The details of the fuel elements model are given in Fig 2. The absorber arms which are moved between the fuel element rows in the aluminum tank are modeled by the application of material cells surrounded by boundary surfaces of 2nd degree. For the input, the coefficient of the each surface is determined in a input generation routine as a function of the angle. The orientation and the segmentation of an absorber arm in the surroundings of the fuel elements are given in Fig. 2.

The rapid shutdown rods are located in the periphery of the core and were modeled in the form of a hollow cylinder with a wall thickness of 4 mm. The fine control rod was modeled in the same detail and placed in the D2O reflector region. The beam tubes of varying diameter and length were modeled in detail and integrated in the corresponding positions of the entire model in accordance with the design and construction documents.

4. FLUX AND BURNUP RECYCLING

Due to the continuous change of the material composition in the fuel meat resulting from fuel consumption, it was necessary to couple the MCNP code with a depletion code. In this way the variation of the neutronic states of the core could be simulated by multiple linked burnup and MCNP calculations. In the next flow diagram, the scheme of data processing and data transfer between the two codes is illustrated. In each time step - representing a time interval in the operation history - fuel burnup is determined on the basis of neutron cross sections and local flux from the previous step of the MCNP run. The resulting local nuclide densities –zone-wise- are used in the next step of calculations with MCNP to generate the flux distribution and one-group cross-section data for the next burnup step.

This procedure is sequentially repeated for all material zones for a couple of time steps until the final time step representing the end of the operating cycle (EOC) has been reached. The one-group data for each burnup calculation are generated by using a separate routine which allows the condensation of the multigroup cross-section data and flux values from the MCNP run in one group with conservation of the total reaction rate.



Flow diagram of coupled MCNP and burnup calculation

In the course of coupled calculations, the variation of the nuclide densities is taken into account for all fission products and actinides which are produced to different extents with fuel consumption. In the case of fresh fuel elements, the material composition is specified by the initial nuclide densities in accordance with the fuel specification of the fuel element supplier. The burnup calculation of the outer shroud tube containing boron is handled in the same way taken into consideration the extended length over the active core.

The precision of MCNP and depletion calculations is determined by modeling the geometrical details of the FE, absorber rods and structures and by the number of histories as well as by the uncertainties of the physical data including the cross-section data sets. In the MCNP model of FRJ-2 the FE, CCAs and structures were segmented in detail. The material zones containing fuel were extensively segmented in the axial, radial and azimuthal direction. To consider the variation of the composition in the fuel meat zones, the detailed segmentation of the whole core resulted in a model with 11 250 material cells.

To achieve a sufficient number of neutron tracks and score in all cells and consequently to reduce the estimated error of the physical values (keff and local neutron flux) all simulations were run for 900 cycles each with 1000 particle histories, of which 100 cycles were excluded for the convergence of the initial fission source. By the statistical evaluation of 900 000, a standard deviation of 0.001 was achieved for the multiplication factor (i.e. 0.10 % dk/k) and 0.05 for the value of the local n-flux.

Due to the complexity of the geometrical model of the reactor and the large number of particle histories, the computing time was reduced significantly by the application of the PVM version of MCNP and utilization of a massively parallel computer system, CRAY-1200, with 32 processors (available 520).

5. RESULTS OF SIMULATIONS

5.1 Criticality States

The neutronic and criticality behavior of FRJ-2 was studied in detail by the simulation of 15 past operating cycles with various core configurations and burnup distribution. In order to verify the geometrical model, the criticality state (representing the multiplication factor) of the reactor core was considered for comparison. For this purpose the eigenvalue was calculated for different burnup states and angles of the CCAs and compared with the critical angle known from the critical experiments at the beginning of a cycle and during operation. According to the results, the criticality state of the core at each time and burnup step is reproduced with a maximum deviation of 0.3 % dk/k representing the accuracy of the model in the simulation of the neutronic conditions of the core. Due to the accurate calculation, the model was used to predict the reactivity value of the core of the actual operating cycle after refueling the core and shuffling some fuel elements. The criticality after reloading the core was achieved as predicted by the model with a deviation of less than 0.2 % dk/k amounting to the 2σ confidence interval.

5.2 Average and Local Distribution of n-Flux

The distribution of the average neutron flux of the fuel elements is given in Fig. 3 in comparison with the measurement performed during operating cycle 1/2000. For the measurement, cobalt foils were irradiated in the central channels of all fuel elements. The counting rate of the foils after activation was applied and compared with the reaction rates calculated by MCNP. The reaction rate was generated for the measuring position using the corresponding flux and cross-section data. To obtain the mean reaction rate for each individual FE, the axial distributions from the calculation and measurement were averaged. Taking the uncertainties of the measuring method resulting from the irradiation time and positioning of the foils into consideration, the calculated values are in good agreement with the experimental results in the whole core.

The maximum flux appears in the central fuel element and decreases towards the outer positions. In addition to the relative flux factors, the axial distribution of the n-flux was calculated in the central channel of the highest rated fuel element and compared with the results of the measurement. For this purpose the rate of the (n,γ) reaction for Co-59 was determined with MCNP by the application of a conversion factor for the mass of

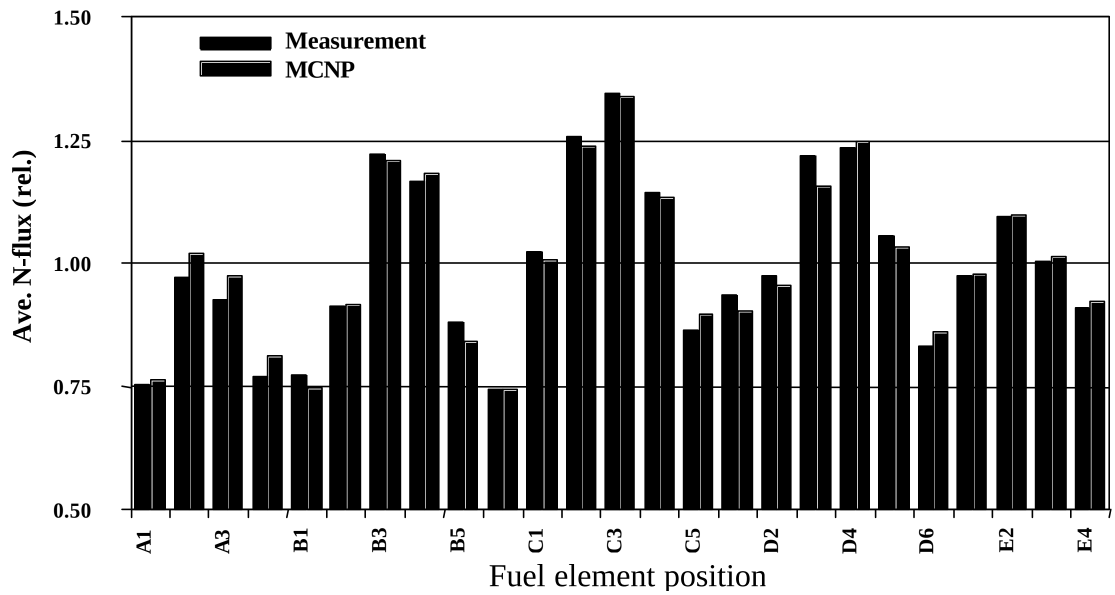


Fig. 3: Comparison of MCNP calculation and measurement of the average n-flux in the inner channel of the FE for the core configuration 1/2000

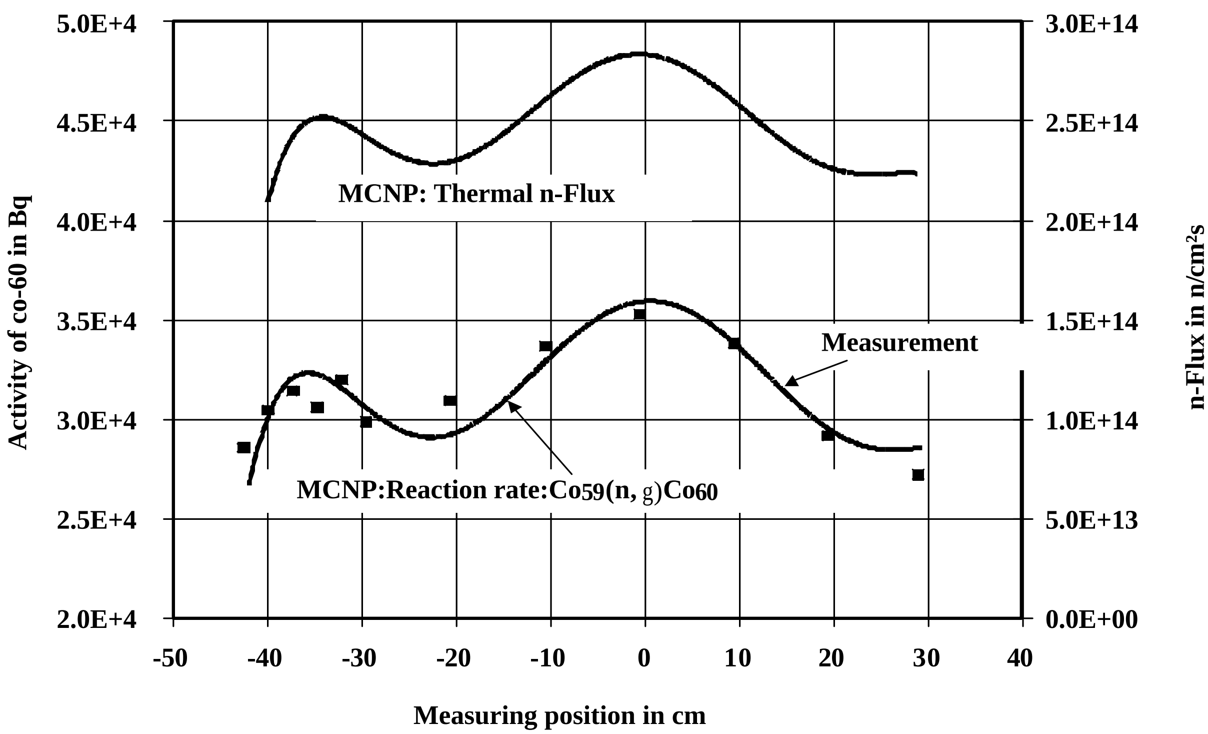


Fig. 4: Distribution of reaction rate (Co-59) in the central channel of the highest rated fuel element (C-3) calculated by MCNP in comparison to the measurement

cobalt foils. The number of the history was chosen as 1.4 million to achieve a standard deviation of 4 %. This result was obtained by 35 h CPU time for all processing nodes and 1 h by using 32 nodes. The results of the calculation are summarized in Fig. 4. The maximum deviation amounts to 5 %. The corresponding n-flux in the inner channel of the FE shown in Fig. 4 was calculated from the reaction rate by using one group cross-section data. The maximum n-flux in the inner channel of the highest rated FE amounts to 2.80E14 n/cm² s at 20 MW power level.

These results show that the model is capable of precisely predicting the flux values. The comparison of the burnup values of the individual fuel elements which are determined experimentally by using the average neutron fluxes also shows a very good agreement with those calculated with the coupled MCNP and burnup code.

5.3 Burnup and Flux Profile in Absorber Blades

Compensation of fuel burnup and shutdown of the reactor is done by raising and dropping of the absorber arms. Due to the inhomogeneous n-flux in the core, each individual CCA is differently burnt up resulting in different lifetimes. In particular, the lower ends of the absorber blades experience a high rate of burnup because of their position in the high flux zone. The consequence is a reduction of the effectiveness and a redistribution of the n-flux and fission power in the adjacent fuel elements.

To determine the burnup profile in the absorber blade of a CCA with a high worth (CCA-5), the absorber blade was subdivided into small segments as presented in Fig. 2. Using the MCNP input file for a typical operating core configuration, the absorption rate was calculated for each individual segment. Fig. 5 shows that the lower

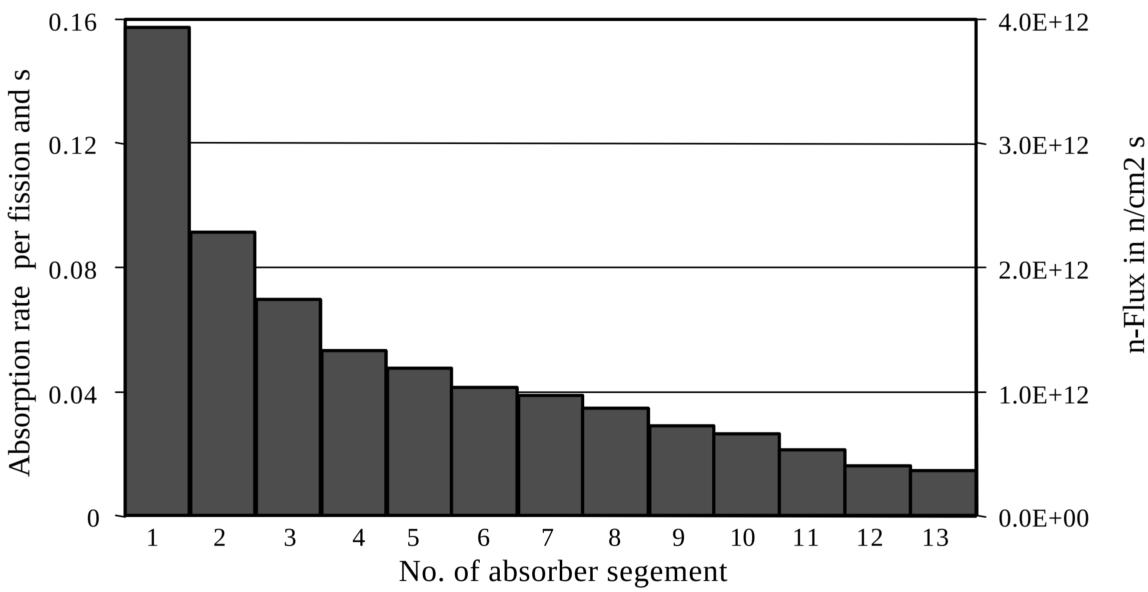


Fig. 5: Burnup and flux profile in a fresh absorber blade of the coarse control arm

ends of the absorber blade experience a rapid consumption of Cd-113 as the main absorbing isotop. The thermal n-flux in the absorber plate of the lowest segment amounts to 3.93E+12 n/cm²s. A comparison with the n-flux in the moderator zone surrounding the absorber segment shows that in the case of a fresh absorber blade the thermal n-flux is depressed in the absorber plate by a factor of 65 as a consequence of total absorption of thermal neutrons. The decrease of the burnup rate in the upper segments is caused by the axial shape of the thermal flux resulting in an inhomogeneous burnup profile in the absorber arm. The reduction of the n-flux in the axial direction is caused by the angular position of all absorber arms in the reactor core. Fig. 5 indicates that within the first 5 cm of the absorber blade the reaction rate is 5times higher than the average.

The decrease of the absorber concentration due to burnup causes an increase of n-flux in the absorber zone so that the burnup rate increases with irradiation time. After 375 days of operation, Cd-113 is completely consumed in the two lowest segments of CCA5 resulting in a reduction of the effectiveness by about 10 % compared with the worth of a fresh absorber blade. If one considers the total burnup of Cd-113 in the two lowest segments as the limit, then the maximum lifetime of this absorber blade in the reactor core is limited to 375 operating days. When the limit is reached the respective absorber blade is replaced by a fresh one.

The worth of the control arms depends on their position inside the reactor core. Due to an enhanced flux maximum in the inner area of the core, higher worths result for the two central CCAs between the D and B rows. Because of the difference in the peripheral structures and due to the inhomogeneous distribution of the fuel burnup in the core, the distribution of the worth is not symmetrical. Fig. 6 shows the relative values of 6 CCAs determined by MCNP for the core of operating cycle 01/2000. For this purpose, the multiplication factors of the core are calculated in different steps. In each step, one

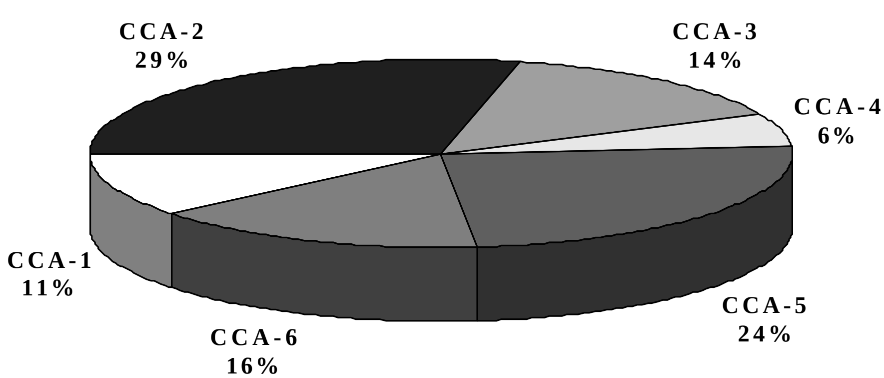


Fig. 6: Relative weights of the coarse control arms in the core of operating cycle 1/2000

CCA is kept in the shutdown position and the remaining five are raised to the critical angle previously determined. The difference in the individual eigenvalue from the critical value with 6 CCAs (keff=1,000) is considered as the individual worth. According to Fig. 6, the effectiveness of the central CCA is significantly higher than the those in the 2rd and 3rd rows. The relation in the effectiveness between CCA4(outer position) and CCA2(central position) amounts to 4.8 in the core configuration of the operating cycle 2/2000.

6. CONCLUSIONS

By combining MCNP and a depletion code a powerful numerical tool has been developed which is capable of precisely simulating the neutronic condition in the reactor core and surroundings. Due to a detailed geometrical MCNP model for FRJ-2, the code system allows the determination of the key physical parameters in fuel tubes, beam channels and structures. The coupled codes enable the users to obtain accurate neutronic details in locations, which are not easily accessible for measurements. The verified code system is an alternative to comprehensive measurements and experiments consisting of criticality tests and routine flux measurements which are performed in each operating cycle. For the aspect of refueling, it predicts the neutronic state and the safety characteristics for any possible configuration of fuel elements including shuffling. The routine application of the method resulted in optimum operation and utilization as well as in a considerable reduction of fuel consumption.

8. References

/1/ J. F. Briemeister Ed. MCNP- A General Monte Carlo N-Particle Transport Code LA-12625-M, Version 4B, Los Alamos National Laboratory, March 1997

/2/ Ronald C. Brockhoff, John S. Hendricks, A New MCNP Test Set, Report LA-12839, Los Alamos National Laboratory, Sept. 1994

/3/ R. D. Mosteller et al, Data Testing of ENDF/B-VI with MCNP: Critical Experiments, Reactor Lattices and Time-of-Flight Measurements,1998, Advances in Nucl. Sci. and Techn., Vol. 24

/4/ R. Nabbi and J. Wolters, Fuel Management for a Safe and Optimum Operation of the German Research Reactor FRJ-2, 1998, Nucl. Eng. and Design 182, pp. 225-231

/5/ R. Nabbi and J. Wolters, Analysis of the Criticality States of the German Research Reactor FRJ-2 Using MCNP4B, Proc. VI. Intl. Conf. on Nuclear Criticality Safety, Versailles, France, Sept. 1999

/6/ F. Rahnema et al., Comparison of ENDF/BV and VI.3 for water reactor calculations Proc. Ann. Mtg. of American Nuclear Society, Vol. 76, pp. 324-325, June 1997