A Sophisticated Computational Method for HEU-LEU Conversion of the German FRJ-2 Research Reactor Using MCNP

R. NABBI, G. Thamm, J. WOLTERS

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

Abstract

Due to a more than four-year shutdown of FRJ-2, for comprehensive inspection and refurbishment work, the HEU-LEU conversion plan was extended beyond the original time schedule. In the meantime a sophisticated computational method has been developed –as a requirement of the licensing authority– for reactor physics calculations, core conversion studies and fuel element performance analysis. For this purpose, an MCNP model in very high fidelity was developed and coupled with a depletion code. The fuel elements, absorber arms, and the moderator zones were highly segmented, and all core and reflector components were included.

The results of the calculation shows that the reactivity value of each core at any burnup step is predicted with a high accuracy (within the 2σ confidence interval). The flux and burnup distribution in all core configurations with HEU and individual LEU fuel elements, respectively, are also in very good agreement with the experimental results. In the case of a single LEU fuel element, the neutron flux is approx. 5 % lower than in an HEU fuel element resulting in an extended irradiation time in the reactor core.

1. Introduction

In view of the fuel supply for long-term operation and return of spent fuel, the operator of the German research reactor, FRJ-2, has participated from the beginning in the RERTR programme and made a comprehensive contribution to the test and use of the LEU fuel for the HEU-LEU-conversion measures. The development and application of the high-density uranium silicide fuel required a modification of the fuel element design of FRJ-2 resulting in performance tests under operating conditions. For the conversion of the whole reactor core with the new LEU silicide fuel element design a long-term programme was established in the mid-eighties in agreement with the German licensing authorities and representatives of the US RERTR programme. The programme consists of the following 4 steps/1/:

Step 1: Performance test of the new fuel element design with HEU fuel in the reactor core under operating conditions

Step 2: Performance test of the new fuel element design with LEU fuel in the reactor core under operating conditions

Step 3: Conversion of the whole core for the use of new fuel element design with HEU fuel

Step 4: Conversion of the whole core for the use of new fuel element design with LEU fuel

At present step 2 is being carried out and fuel elements have been ordered for step 3 from CERCA in France.

During the time originally planned for the implementation of the 4-step conversion programme, the reactor had to be shutdown for comprehensive inspection with regard to ageing and refurbishing for about 4.5 years resulting in a postponement of the conversion activities/2/. In parallel, a sophisticated computational method was developed which is capable of precisely simulating the physical behavior of mixed and LEU cores as well as LEU fuel element performance in the FRJ-2 core of high power density. The method was also considered to be a tool for the major replacement of comprehensive reactor physics measurements and criticality tests being performed at the beginning of each operating cycle (BOC). The extensive reactor physics measurements at BOC are required to demonstrate sufficient safety margins for fuel element power limits and sufficient shutdown reactivity to cope with a failure of the most effective absorber arm (so-called coarse control arms)/3/.

For the aim of reactor physics analysis and conversion studies, the Monte Carlo codeMCNP was chosen, coupled with a depletion program and applied to the FRJ-2 /4-6/. The MCNP code was used because of its comprehensive capability of modelling complex 3D geometries of the fuel element assembly, shutdown system and core structures of FRJ-2.

The present paper describes in detail the results of the new computational method for the simulation of the neutronic and criticality behavior of FRJ-2 for routine application in fuel management and HEU-LEU conversion analysis.

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 arranged in five rows of 4, 6, 5, 6 and 4 fuel elements (Fig. 1). It is accommodated within an aluminum tank 2 m in diameter and 3.2 m in height. The tank is surrounded by a graphite reflector 0.6 m thick 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 0.61 m. Each tube is formed by three material testing fuel plates containing fuel meat and aluminum cladding. The tubes are accommodated in a shroud tube of 103 mm diameter, to which they are attached by four combs at either end (electron-beam-welded tube for HEU fuel elements). Due to metallurgical interaction of the high-density LEU fuel with the cladding material, aluminum alloys such as AlMg or AlMgSi must be employed for the fabrication of LEU fuel elements. When using the current electron-beam welding technique (EB design with HEU fuel) to form fuel tubes from the pre-curved fuel plates, the vaporization of the alloy constituents (Mg, Si) results in extraordinarily high failure

rates so that a new element design was proposed by Nukem and tested by dummy element fabrication. According to the new design the pre-curved fuel plates are swaged into 3 plates provided with special grooves then forming the final fuel tube (roll-swaged element; RS design). Fig. 2 illustrates the differences between the EB- and RS-element design.

The fuel elements with HEU fuel contains UAlx in an aluminum matrix with a U-235 mass of 170 g and 150 g, respectively which are used in the outer and inner positions of the core. The meat of the fuel element with LEU fuel consists of U_3Si_2-Al with a uranium mass of 180 g and 225 g, respectively. The annular water gap between the tubes has in both cases 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 pneumatic actuators. The CCAs are lowered and raised manually around a pivot in order to control power levels during normal operation. 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 penetrating into the reactor tank.

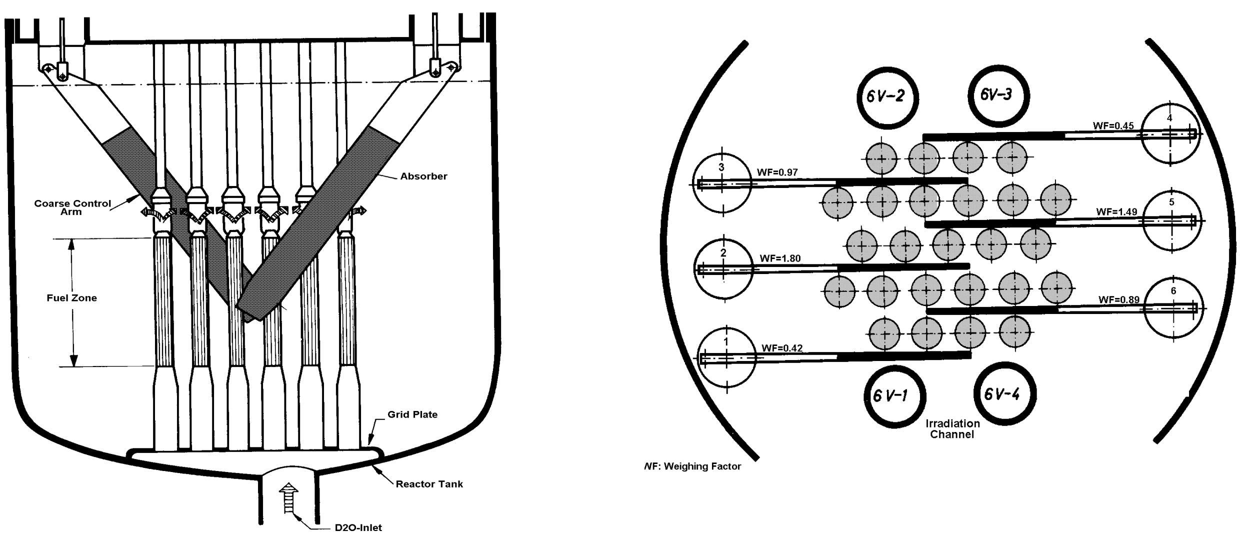


Fig. 1: Arrangement of fuel elements and coarse control arm inside the reactor tank

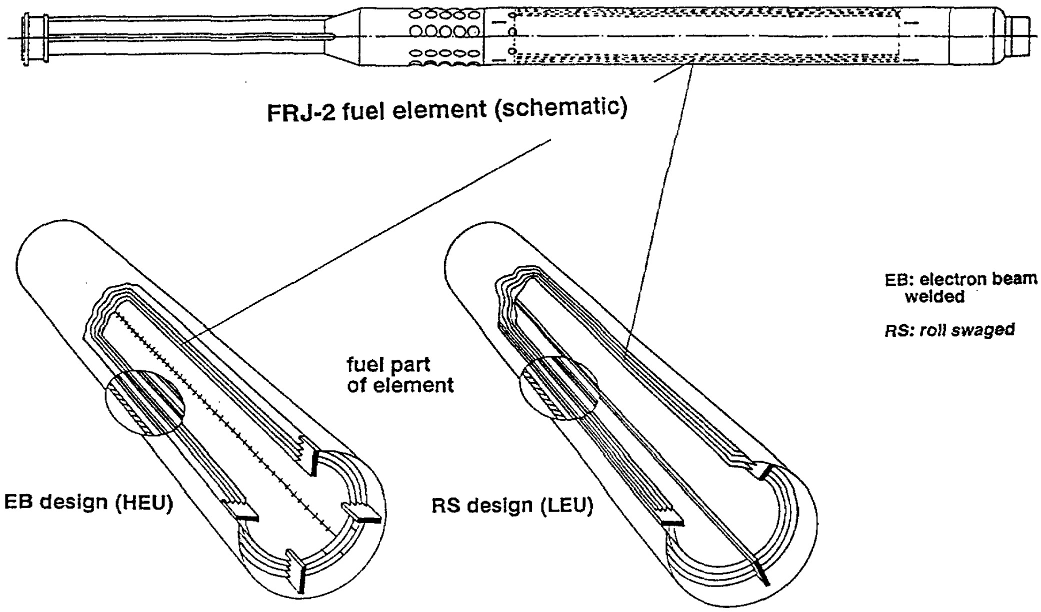


Fig. 2: Comparison of HEU and LEU fuel elements for FRJ-2

3. MCNP model of FRJ-2

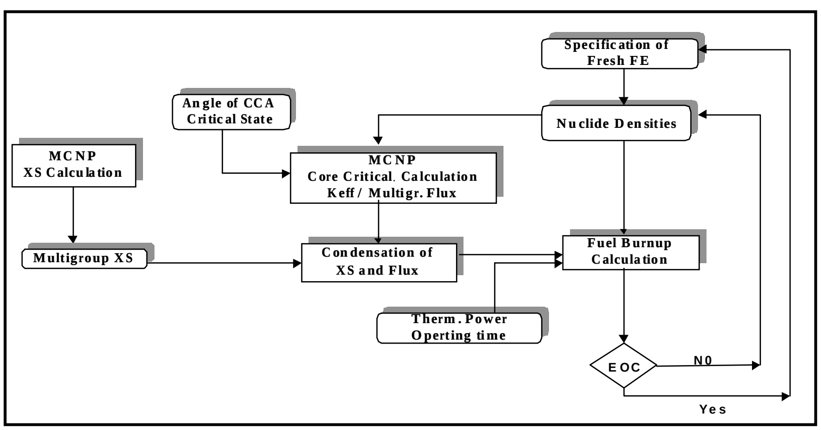
The MCNP model of FRJ-2 is a complete 3-dimensional full-scale model with a very high level of geometric fidelity. It comprises the reactor core, CCAs, core structures, beam tubes, the graphite reflector and the biological shield. The core region consisting of 25 fuel elements was modelled as a cylinder 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 fuel tubes and the borated outer shroud tube. Each individual cell is divided into 15 axial, 35 radial and azimuthal material zones.

The D2O reflector in the 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, unfuelled ends and nozzles of the fuel elements, the alumninum structures and corresponding D2O were modelled 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 rapid-shutdown rods were located in the periphery of the core and modelled in the form of a hollow cylinder with a wall thickness of 4 mm. The fine control rod was modelled in the same detail and placed in the D2O reflector region. The beam tubes of varying diameter and length were modelled in detail and integrated in the corresponding position of the entire model in accordance with the design and construction documents.

4. Burnup recycling

Due to the continuous change of the material composition in the fuel meat resulting from the 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 multigroups cross-section data and flux values from the MCNP run in one group with conservation of total reaction rate.



Flow diagram for coupled MCNP and burnup calculation of FRJ-2

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 under consideration of the extended length over the active core.

5. Precision of the MCNP runs

Due to the stochastic character of the MCNP code, the results of the calculations represent an average of the contributions from many individual neutron histories sampled during a simulation. The precision of such calculations is determined by modelling 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.

As a result of a coarse representation of the core geometry – for example by homogenization of the material zones - the physical events like scattering, absorption, fission take place in an extended material region resulting in a local disturbance of physical states. For this reason, in the MCNP model of FRJ-2 the FE, CCA and structures were segmented with very high fidelity. 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 reduce the estimated error of the physical values (keff and local neutron flux) all simulations were run with 900 cycles each with 1000 particle histories, of which 100 cycles were excluded for the convergency of the initial fission source. By the statistical evaluation of 800 000, 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.

The nuclear data used in the calculation are those included in the library of MCNP developed from the ENDF/B-V and VI by the NJOY data processing code. In some cases the NJOY code was also used to generate our own cross-section data for application in the MCNP runs/7/. The effect of the cross section has been studied in different investigations using ENDF/B-V and ENDF/B-VI for light water criticality analysis and criticality safety benchmarks /8,9/. Accordingly, a slight difference could be found by using the two libraries with MCNP.

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).

6. Results of core physics simulations
6.1 Whole core analysis

After completing the geometrical modeling of the reactor core and surrounding, the neutronic and criticality behavior of FRJ-2 was studied in detail by a simulation of past operating cycles with various core configurations and burnup distribution. In order to verify the whole model the criticality state (corresponding to the multiplication factor) of the reactor core for each step of the operating cycle was compared with those predicted by the calculation. Moreover, the total burnup values of the individual FE at the end of each cycle calculated by the coupled programs were compared with the experimental values obtained on the basis of the flux measurement. In addition, the distribution of the average fuel element flux for each operating cycle was taken for comparison to check the accuracy of the calculational method in the prediction of the local neutronic conditions and neutron flux. Fig. 3 shows the deviation of the multiplication factor for each critical core from a reference value corresponding to the criticality state of the reactor as a function of the burnup step for 4 operating cycles. Accordingly, the criticality state of the core at each time and burnup step is reproduced with a precision of less than 0.3 % dk/k representing the accuracy of the model in the simulation of the neutronic conditions of the core. Due to the high prediction quality the method was used to predict the reactivity value of the core of the actual operating cycle (BOC 3/99) after reloading 6 fresh fuel elements including 4 shuffling steps. 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. For this precision level one million neutron histories were simulated.

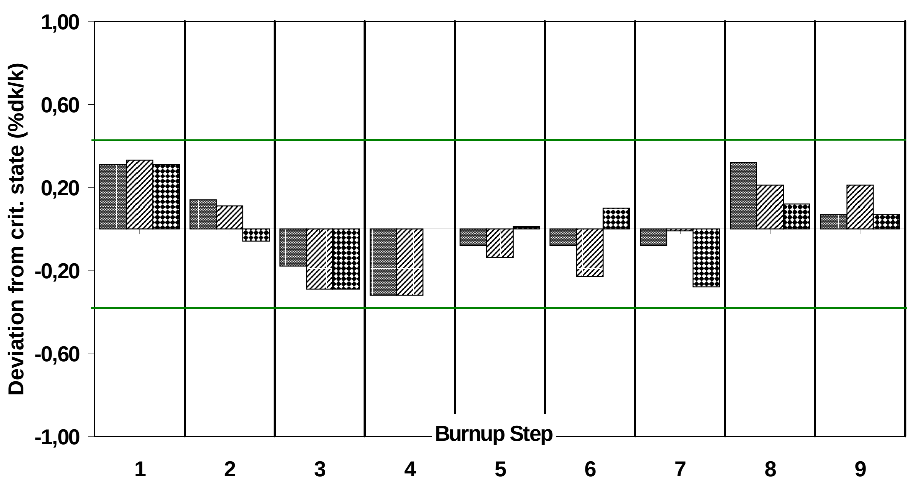


Fig. 3: Deviation of calculated multiplication factor from the reference value for the criticality state as a function of burnup step

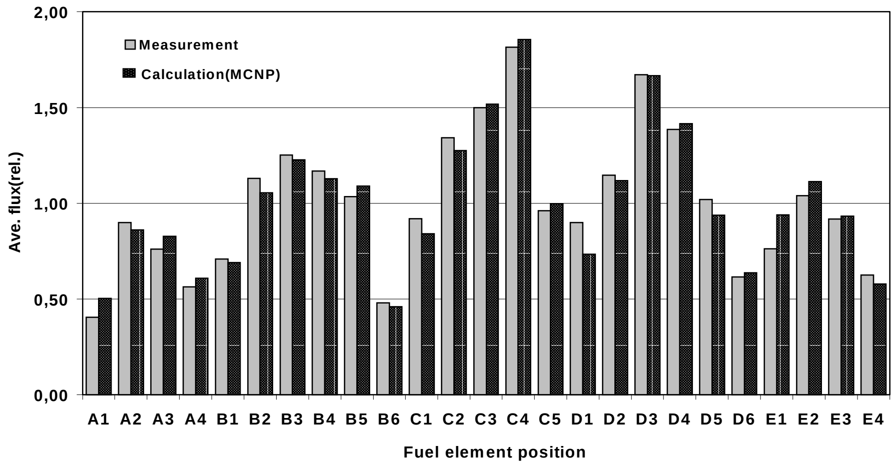


Fig. 4: Comparison of calculation and measurement for the average neutron flux at the inner channel of fuel elements for core configuration 3/1999

The distribution of the average neutron flux of the fuel element is given in Fig. 4 for comparison with the results of the measurement performed during the operating cycle 3/99 by the foil activation technique. (activity of Co-60). For this comparison, the reaction rates are generated for the measuring position using the corresponding flux and cross-section tallies. Under the consideration of the uncertainties of the measuring method resulting from the irradiation time and positioning of the foils, the calculated values are in good agreement with the experimental results in the whole core.

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 experimentally determined by using the average neutron fluxes are also in very good agreement with those calculated with the coupled MCNP and burnup codes.

6.2 Behavior of LEU Fuel Elements

In order to compare the operating behavior of the HEU and LEU fuel elements in the reactor core, burnup and flux calculations were performed using the existing MCNP and burnup model. For an HEU-170 fuel element(HEU: 170 g U-235, 79.85 % enriched) and LEU-180 (LEU type: 180 g U-235, 19.95 % enriched). The results of the calculations are given in Fig. 5.

Accordingly the neutron flux in an HEU fuel element is 4.8 % higher than in the case of the LEU fuel type. The difference becomes higher with irradiation time and approaches 14 % at a burnup level of 50 %. The difference results from the changes in the macroscopic cross sections which are determined by the density of U-235 and U-238 as well as by the changes in the neutron spectrum. Due to the constant local power, the increase in the uranium content results in an increase of the macroscopic cross section causing a reduction of the neutron flux. The change of the neutron spectrum (hardening) due to the use of the LEU fuel influences the neutron flux too. The effect of density remains dominant and determines the variation of the neutron flux during the whole irradiation cycle. The burnup of uranium 235 in the fuel meat

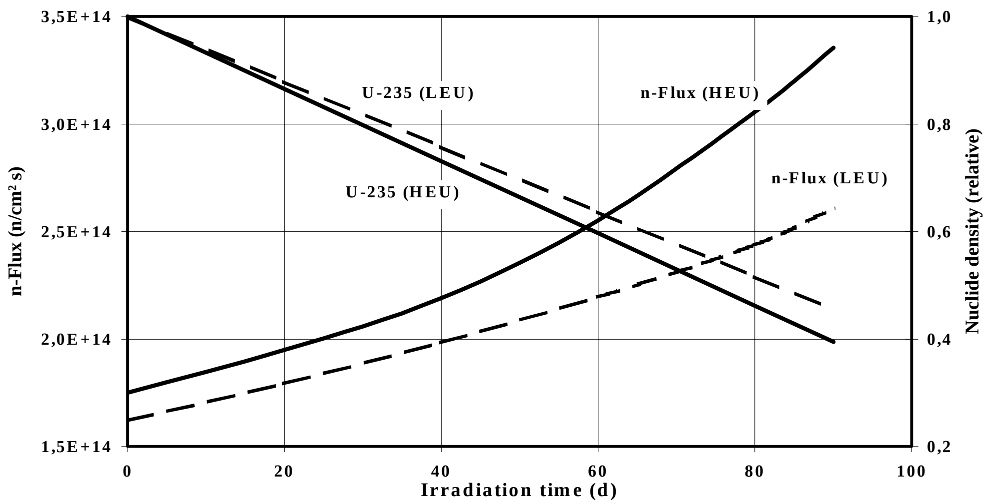


Fig. 5: Variation of neutron flux and U-235 density for HEU and LEU fuel element

for the HEU and LEU is shown in Fig. 5 for comparison. It shows that - due to the higher neutron flux - the burnup of U235 occurs more rapidly in the case of the HEU fuel element.
he consequence is a relatively rapid rise in the neutron flux as shown in Fig. 5. A slow burnup rate in the LEU fuel element results in an increase of the irradiation time of samples in the reactor core.
The use of high-uranium-density LEU fuel elements influences the reactivity state of the core. Further calculations with the existing code show that besides the high amount of U-235 content in the LEU case (10 g) the reactivity loss by an amount of 0.1 % dk/k takes place as a consequence of the higher contribution of neutron absorption in the LEU fuel. This value depends on the position of the fuel element in the core and reaches 0.5 % dk/k in the case of loading 3 additional 180 g LEU fuel elements in the reactor core. By partial loading of the core with LEU fuel elements containing 225 g U-235, the reactivity loss is compensated for the operation period in a cycle.

7. Conclusions

For optimum fuel management and HEU-LEU conversion of the reactor core, a sophisticated method was developed on the basis of the MCNP and a depletion code and verified by comparison with reactor physics measurements and experiments. The existing numerical tool is capable of predicting the neutronic state of each individual fuel element in an HEU, mixed and LEU core. The application of the code reduced the extent of comprehensive measurements and tests significantly, resulting in an optimization of operation management, core conversion measures and in a reduction of the fuel element performance tests. The research reactor division is able to perform detailed core physics analysis with respect to the power peaking profile, flux distribution and safety characteristics of the reactor core. Based on the existing method, core physics calculations are performed in advance to facilitate the realization of the conversion plan and meet the requirements of the licensing authority in due course.

The use of 180 g/225 g LEU fuel elements to replace the 150 g/170 g HEU type compensates the reactivity loss due to the high uranium density. The development of an optimum loading strategy using LEU fuel is the subject of further analysis.

8. References

/1/ R. Nabbi and G. Thamm"HEU-LEU conversion of FRJ-2 DIDO at KFA Jülich"
RERTR Meeting, 1990, Newport, Rhode Island, USA,

/2/ J. Wolters," Refurbishing of the German FRJ-2 Research Reactor"
RERTR Meeting, 1993, Tokamura, Japan

/3/ R. Nabbi and J. Wolters,"Fuel Management for a Safe and Optimum Operation of the German Research Reactor FRJ-2",
Nucl. Eng. and Design 182, pp 225-231 (1998)

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

/5/ Ronald C. Brockhoff, John S. Hendricks,"A New MCNP Test Set"
Los Alamos National Laboratory Report LA-12839, Sept. 1994

/6/ 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

/7/ MacFarlane et al., "The NJOY Nuclear Data Processing System",
LA-9303-M, 1982,Vol. I and II, Los Alamos National Laboratory

/8/ 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 (1997)

/9/ R. D. Mosteller et al.,"Data Testing of ENDF/B-VI with MCNP: Critical Experiments, Reactor Lattices and Time-of-Flight Measurements"
Advances in Nucl. Sci. And Techno., Vol. 24, (1998)