Status of HEU-LEU Conversion of FRJ-2

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Abstract

The operator of the German FRJ-2 research reactor, "Research Center Jülich", has participated from the beginning in the RERTR programme and made comprehensive contributions to the test and use of LEU fuel for HEU-LEU-conversion measures. The originally planned time scale for the conversion of FRJ-2 was significantly delayed because of a change of the manufacturer of the LEU fuel elements and a 4 years shutdown of the reactor for refurbishment purposes. In the meantime the new LEU fuel elements are qualified and tested in the reactor. In the moment calculations for the safety report are made and it is planned to apply for the license of FRJ-2 operation with LEU fuel at the beginning of 2003. In order to get most reliable results a sophisticated computational method based on a MCNP model coupled with the depletion code BURN was developed for reactor physical calculations, core conversion studies and fuel element performance analysis and applied to the mixed and LEU core. The licensing schedule and results of latest calculations for the conversion study will be presented. The simulations shows that the thermal flux in the LEU core is about 19 % resulting in a lower burnup rate. But in the reflector area around the core and in the center of the cold n source the neutron flux reduction remains limited to 6 %. Due to a harder neutron spectrum in the LEU core the kinetic and safety related parameters are slightly reduced. Using the ORIGEN code it could be shown that the increase of the total fission products inventory amounts to about 6 % compared to a HEU core. As a consequence of the high amount of U-238, the amount of U 235 in the LEU core has to be about 27 % higher than in the HEU core but the U-235 burnup is approx. 5 % lower due to the contribution of fissile plutonium.

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. The core configuration could be left without any change. 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

Steps 1 and 2 are successfully carried out and step 3 has been started by loading the first 4 fuel elements from CERCA in France with the new design into the reactor core.

During the time originally planned for the implementation of the 4-step conversion programme, the originally chosen manufacturer for the LEU fuel elements gave up production so that a new company had to be found and qualified. In addition the reactor had to be shut down for comprehensive inspection with regard to aging and refurbishing. Those two measures resulted in a quite long 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 /3/. For the aim of reactor physics analysis and conversion studies, the Monte Carlo code MCNP was chosen, coupled with a depletion program and applied to the FRJ-2. The MCNP code was used because of its comprehensive capability of modeling complex 3D geometries of the fuel element assembly, shutdown system and core structures of FRJ-2 /4-6/.

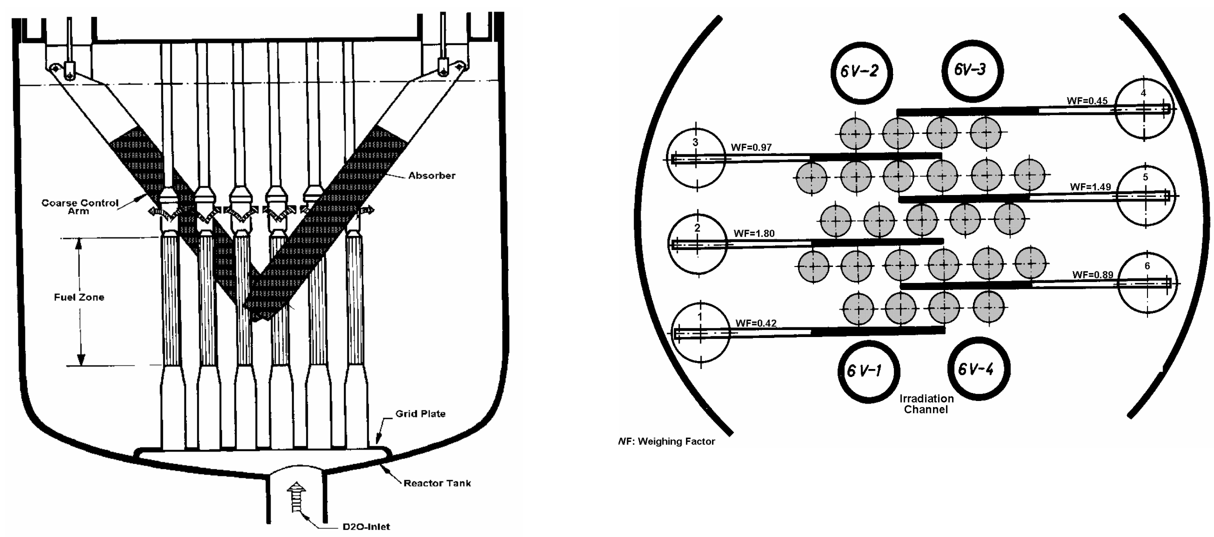
Having successfully finished the in core tests of the newly designed fuel elements with HEU (step 1) and LEU fuel (step 2) and having started step 3 it was decided to apply for the license of HEU-LEU conversion by the regulatory body in spring 2003. With help of the MCNP code many reactor physical calculations for the future mixed and LEU cores have been carried out in order to define the amount of U-235 and boron in the LEU elements. For this aim calculations were made to determine fuel burnup, neutron flux and power distribution as well as safety margins.

The present paper describes the latest results of the coupled MCNP-BURN calculations and gives an overview over the planned schedule for the HEU-LEU conversion.

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-beamwelded 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



Vertical cross section Horizontal cross section

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

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.

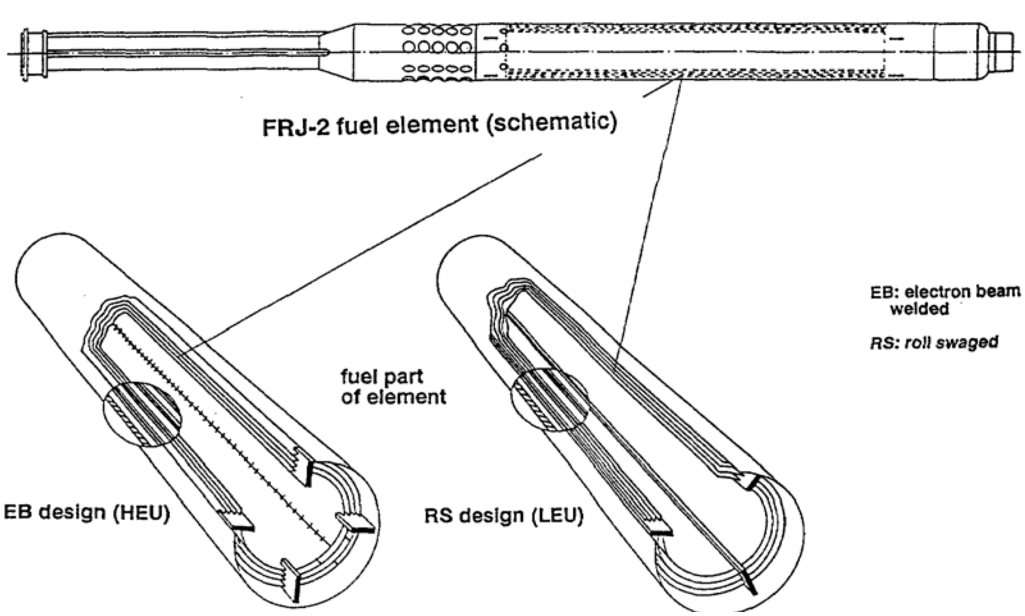


Fig. 2: Comparison of HEU and LEU fuel elements for FRJ-2

The fuel elements with HEU fuel contains UAl_x 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 an uranium mass of 200 g corresponding to latest calculations. 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.

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, part of the absorber arm and cooling gaps. For the detailed modeling each individual fuel element is divided into 15 axial, 35 radial and 6 azimuthal material zones. 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. The detailed segementation of the core and the surroundings for the MCNP and depletion calculation resulted in a model with 11250 material zones.
To achieve sufficient numbers of neutron tracks and events in all cells and consequently reduce the estimated error of the physical values (keff and local neutron flux) all simulations were run for a high number of particle history resulting in a standard deviation of 0.001 for the multiplication factor (i.e. 0.10 % dk/k) and 0.05 for 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.

4. Results of neutronic analysis

In the framework of a parameter study in order to define the content of uranium fuel in the future LEU fuel element the standard HEU fuel elements containing 150 g and 170 g U-235 were replaced by unique LEU fuel elements with a U-235 mass of 220 g at first. To obtain a high shutdown reactivity at the beginning of each operating cycle, the mass of the burnable absorber (boron) in the stand tube of the fuel element was increased from 0.4 g (HEU case) to 0.8 g. With the MCNP code the burnup of the different mixed cores representing the conversion phase from HEU to LEU could be simulated. Each individual step covering 21 operating days at 20 MW was calculated by 10 burnup and MCNP runs. Due to the fact that the whole boron is not completely depleted during an operating cycle, additional fuel is needed to compensate the existing boron at any state of the core. This results in a necessary increase of the average fuel mass which together with the absorbing effect of boron causes a reduction of the average neutron flux due to the absorbing effect of boron on one hand and due to the high amount of fuel on the other hand. A calculation of the first mixed core with 6 LEU fuel element containing 0.4 g boron (instead of 0.8 g) shows that shut down reactivity is decreased by 0.4 % dk/k per fuel element and the n flux is improved by 3 % at the position of cold neutron source and isotope production channels (2V3 and 2V8). The parameter study showed an optimum level of the n flux using unborated fuel element with a U-235 content of 200 g at an enrichment of about 20 %.

A comparison of the HEU and LEU case shows that in case of a LEU core, the amount of the U-235 has to be increased by 27 % in order to achieve the same reactivity state as in a HEU core at BOC. The increase in the amount of fissile uranium is a consequence of high absorption rate in U-238 due to the high density of the LEU fuel. The loading of the core with LEU fuel elements results in a different neutronic state with differences in the neutron flux, power distribution and burnup behavior. For a comparison, the distribution of fuel burnup and position factors has been given in Fig. 3 for a typical working HEU and LEU core, respectively. In the middle of an operation cycle the absolute burnup in the core is 3.75 % lower than in the HEU case. This is a consequence of lower neutron flux and burnup rate in the LEU core. Accordingly the rate of the U-235 burnup in the LEU core amounts to about 1.2 g/MWd and is approximately 5 % lower than in case of HEU core. The variation of the neutron flux and fuel burnup has been summarized in Fig. 4.
Accordingly the average burnup of the fuel in a HEU core is increased from 28.75 % to 41.10 % and in the LEU case from 25.5 % to 35.5 % representing a lower burnup rate. The neutron flux shows the same behavior. The lower burnup rate of U-235 in a LEU core is caused by generation and subsequent fission of fissile plutonium by a considerable amount. For an average LEU core, the amount of the fissile Plutonium isotopes is about 94 g and

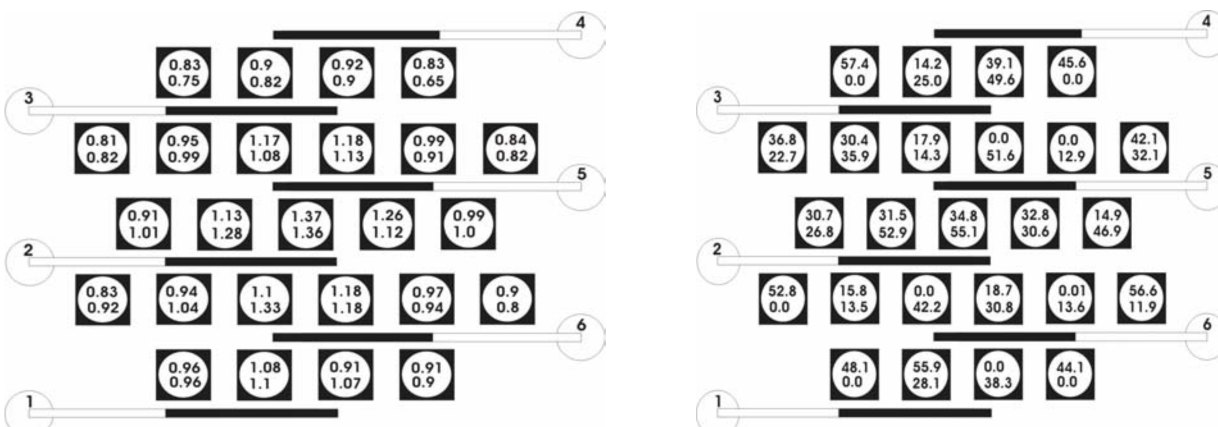


Fig. 3: Distribution of the position factors and burnups of the fuel elements in the HEU and LEU core

exceeds the plutonium content of HEU core by factor 8.2. A lower burnup rate in the LEU core results in an increase of possible irradiation time.

With regard to the level of thermal neutron flux the simulation shows a considerable reduction in the course of conversion of the core from HEU to the LEU fuel. The change of the neutron flux depends on the location inside the core and surrounding. In average the reduction in the core amounts to 19 %. In the peripheral zone of the core at the location of the beam tubes and irradiation channels the change of the neutron flux depends on the location of the channel. In the center of the cold neutron source (liquid hydrogen) located in the midplane at a distance of

0.6 m from the core axis, the change of thermal neutron flux is moderate (about 6 %) due to the thermal character of the reflector area. A compilation of the values of thermal neutron flux is given in Fig. 5. The distribution of the thermal power of the fuel elements has been depicted in Fig. 6 for the HEU and LEU core.

Fig. 4: Variation of the burnup and thermal flux during an operating cycle at 20 MW

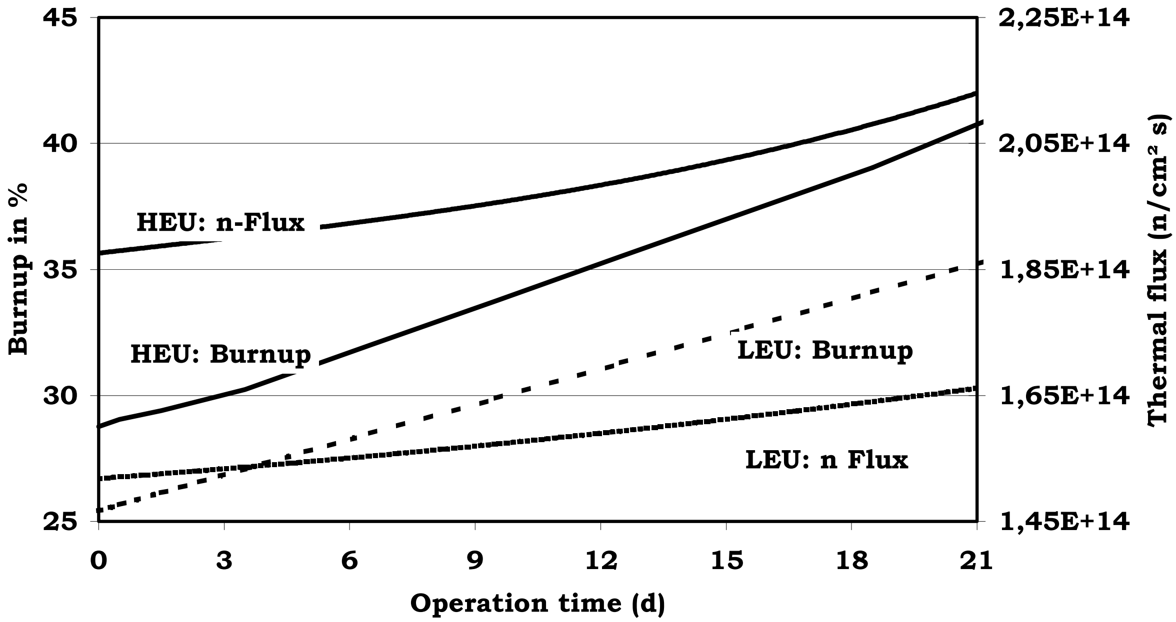
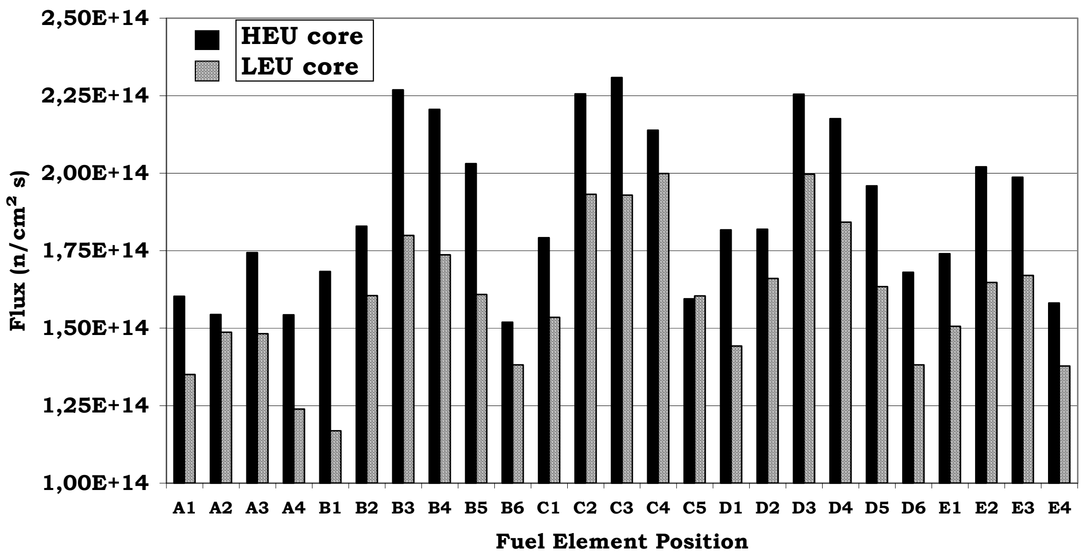
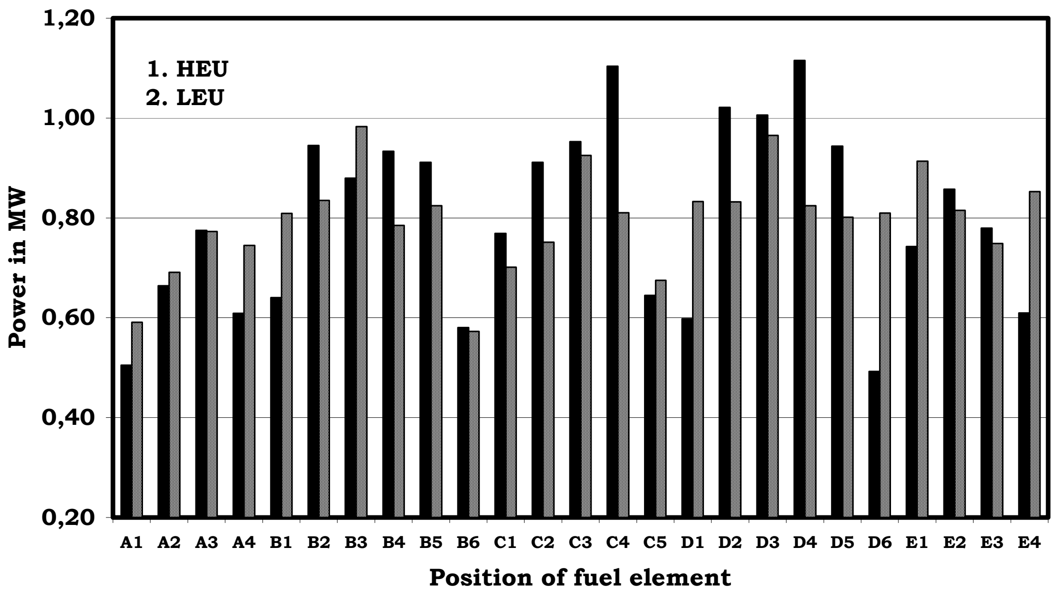


Fig. 5: Comparison of the distribution of the thermal neutron flux in the HEU and LEU core



Due to the high amount of U-238, the neutron spectrum is harder in case of LEU. As a result of the change of neutronic state of the core, the absorber rate decreases by about 15 % resulting in a decrease of reactivity from 18,4 %dk/k to 15,6 %dk/k at a CCA angle of 23°. The simulation shows a considerable reduction of the effectiveness of the absorber systems with the consequence of reduction of the of shutdown and excess reactivity in LEU case.

Fig. 6: Distribution of power of the fuel elements in the HEU and LEU core



The transient behavior of the core is significantly influenced by the kinetics and safety characteristics of the LEU core. Due to the variation of the neutron spectrum, the temperature coefficient of reactivity at the average operating temperature is significantly changed. The results of the calculations are summarized in Table I. Accordingly, the contribution of the moderator to the total reactivity coefficient is decreased by 18 %. By comparison, the Doppler coefficient increases by 63 % in the case of conversion of the core from HEU to LEU fuel. However, it remains lower than the moderator contribution by one order of magnitude. Due to shortening of the absorption length of the fission neutrons resulting from the higher absorption rate in U-238, the prompt neutron lifetime is reduced by 7 %. In the event of transients with reactivity insertion the change of the neutronic variables becomes faster causing a fast transient behavior in comparison to the HEU core. To cope with this type of transients the limiting values for the max. amount of reactivity insertion by experiments are accordingly fixed and modified. For the control of the design basis accident (rupture of the most effective absorber arm) with a high amount of the reactivity insertion, the loading is managed in such a way that the shutdown reactivity is not reduced below a minimum level required for a stable and continuous subcriticality.

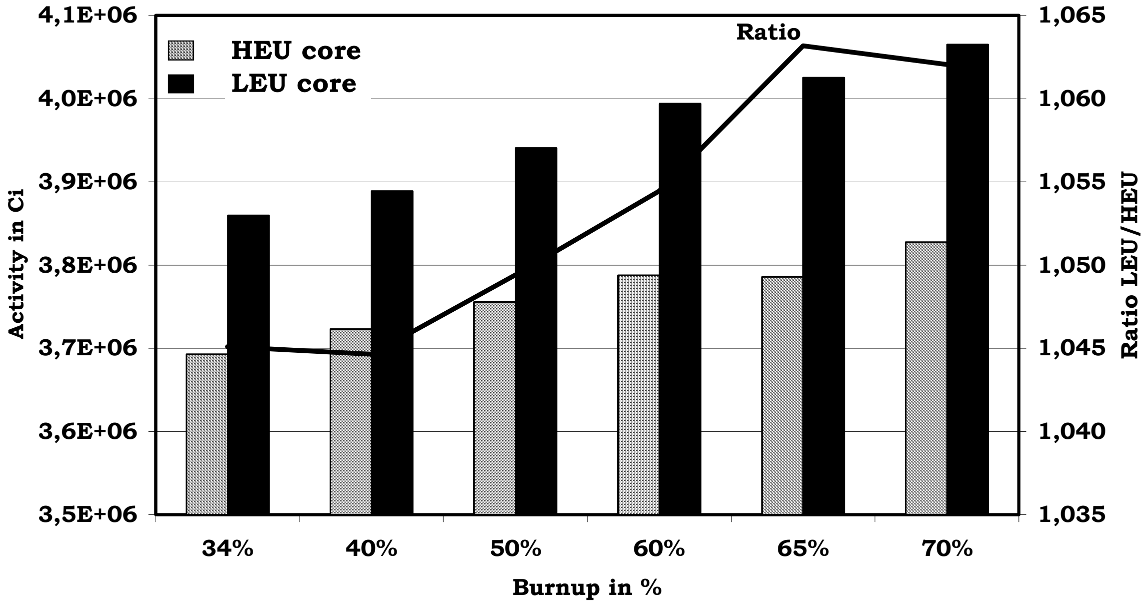
Table I: Safety-related parameters for HEU and LEU core

Core parameter ⟶	Moderator Coefficient	Doppler Coefficient	Prompt neutron lifetime
HEU Core	2.30E-2	1.14E-3	5.44E-4
LEU Core	1.90E-2	1.86E-3	5.10E-4
Ratio	0.82	1.63	0.94

Reactivity coefficients in % dk/k °C, neutron lifetime in s

Due to the difference of fuel composition and neutronic states in the HEU and LEU fuel elements, the inventory of the fission products becomes different in the process of power generation and irradiation. As discussed before, the high amount of U-238 causes a reduction of the neutron flux resulting in a decrease of U-235 fission product activity in a LEU fuel element in the case of the same amount of fissile U-235. However, due to the high amount of U-238 in a LEU fuel element and the contribution of fission products from fissile plutonium isotopes, the total activity of the fission products becomes slightly higher. The results of the calculation performed for a fuel element of average thermal load (800 kW) and maximum burnup of 60 % (fifa) are summarized in Fig. 7. Accordingly, the total activity of the fission products in a LEU fuel element type (200 g U-235) is approx. 6 % higher than in a standard HEU fuel element with 170 g U-235.

Fig. 7: Total activity as a function of burnup for a HEU and LEU fuelelement with the average power of 0,80 MW



5. Conversion schedule

The operators of the FRJ-2 have granted within the frame of the contract for spent fuel disposal with the US-DOE to use LEU fuel not later than June 2006 for reactor operation. Applying for the licence for HEU-LEU conversion in spring 2003 the approval is expected in spring 2004. This leaves 2 further years for the procurement of LEU fuel and fabrication of the LEU fuel elements. As the design of the elements is defined and the manufacturer CERCA is qualified and has already fabricated LEU test elements for the reactor it should be no problem to get LEU elements early enough before the mentioned deadline.

6. Conclusions

The preparations for the conversion of the German FRJ-2 research reactor from HEU to LEU fuel have made a great progress so that application for a license will be submitted to the authority in spring 2003. Due to long term character of licensing process, the final approval is expected in spring 2004. The reactor can be operated with LEU fuel beginning 2006.
In order to precalculate the reactor physical data of HEU, mixed and LEU cores needed for the application a sophisticated method was developed and validated on the basis of the Monte Carlo code MCNP coupled with the depletion code BURN. With help of such simulation calculations an optimum loading strategy for LEU cores could be found which minimizes reactivity losses due to higher Uranium density. The future LEU fuel element will contain 200 g U-235 without any boron. Further calculations showed that the burn-up rate of U-235 in the LEU core will be lower by about 5 % than in a HEU core due to the production of Pu which acts as an additional fissile fuel. The total fission products inventory will be increased not more than 6 %. Due to the

flexibility of the loading strategy the fuel element powers in the typical working LEU core show a moderate distribution with sufficient margins to the limiting values.

Resulting from the high amount of U-238 the mean n-flux in LEU core will be about 19 % lower on average. In the relevant positions of the reflector accommodating for instance the cold n-source the loss of n-flux will only be about 6 %. With the MCNP-BURN code it is possible to predict optimum fuel element loadings to improve the neutron flux at special positions in the core up to 10 %. Corresponding to the lower mean n flux and change in the n-spectrum in a LEU core the effectiveness of the absorber systems is reduced about 15 % by comparison to a typical HEU core. Due to high safety margins of FRJ-2, the reduction of effectiveness does not touch the safety behavior of the system. The simulation of transient behavior of FRJ-2 with the characteristics of the LEU core will be subject of further comprehensive analysis.

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