ALTERNATIVE LEU DESIGNS FOR THE FRM-II WITH POWER LEVELS OF 20-22 MW*

N. A. Hanan, R. S. Smith, and J. E. Matos

Argonne National Laboratory
Argonne, Illinois 60439-4815 USA

ABSTRACT

Alternative LEU Designs for the FRM-II have been developed by the RERTR Program at Argonne National Laboratory (ANL) at the request of an FRM-II Expert Commission established by the German Federal Government in January 1999 to evaluate the options for using LEU fuel instead of HEU fuel in cores with power levels of 20 MW. The ANL designs would use the same building structure and maintain as many of the HEU design features as practical. The range of potential LEU fuels was expanded from previous studies to include already-tested silicide fuels with uranium densities up to 6.7 g/cm^3 and the new U-Mo fuels that show excellent prospects for achieving uranium densities in the 8-9 g/cm^3 range. For each of the LEU cores, the design parameters were chosen to match the 50 day cycle length of the HEU core and to maximize the thermal neutron flux in the Cold Neutron Source and beam tubes. The studies concluded that an LEU core with a diameter of about 29 cm instead of 24 cm in HEU design and operating at a power level of 20 MW would have thermal neutron fluxes that are 0.85-0.86 times that of the HEU design in the Cold Neutron Source. With a potential future upgrade to a power of 22 MW, this ratio would increase to 0.92-0.93.

INTRODUCTION

The FRM-II reactor, under construction since 1996 at the Technical University of Munich (TUM), is a high performance beam tube research reactor[1] with a compact core that was designed to use weapons-grade highly enriched uranium (HEU) in its fuel. The HEU design has a power level of 20 MW and is cooled by light water and moderated by heavy water. The peak unperturbed thermal flux in the moderator tank is 8×10^{14} n/cm^2-s.

In previous analyses[2-6] the RERTR Program at ANL designed an LEU core using U_3Si_2-Al fuel with a uranium density of 4.5 g/cm^3 that was thoroughly tested and licensable. The design with this fuel required a larger core (35.0 cm OD instead of 24.3 cm OD in the HEU design) and a power level of 32 MW in order to match both the 50-day fuel cycle length and the thermal flux performance of the HEU design. All of the key safety criteria for the LEU core were shown to be satisfied.

In January 1999, the German Federal Government, through the Federal Ministry of Education and Research (BMBF), established an FRM-II Expert Commission to evaluate the options for using

LEU fuel in the FRM-II. This group concluded at an early stage that it was not feasible to implement the 32 MW LEU design, since substantial cost and licensing penalties would be incurred to implement this design, given the state of construction of the HEU design. ANL was asked by the FRM-II Expert Commission to determine the design options and performance of 20 MW cores that use advanced LEU fuels, the same building structure, and as many of the features of the HEU design as practical. ANL presented this information to the Commission at a meeting in Bonn, Germany, in April 1999. This paper presents a summary of the work that was done for this meeting, along with follow-up analyses resulting from issues that were raised at the meeting.

FRM-II Design Models

A schematic diagram of the FRM-II design[7] is shown in Fig. 1 and the ANL model of the reactor core, Cold Neutron Source, heavy water moderator, light water pool, and concrete shield based on the MCNP Monte Carlo code[8] are shown in Fig. 2. A detailed analytical model of the HEU core with circular arc-type fuel plates instead of involute-type fuel plates is shown in Fig. 3. Burnup calculations were performed using the REBUS-3 code[9], the DIF3D diffusion theory code[10], and the WIMS-ANL cross section generation code[11] with a library based on ENDF/B-VI data.

The ANL HEU design model is based on information provided to ANL by TUM. The design and location of the Cold Neutron Source (CNS) is based on information published[12] by TUM in May 1998. However, detailed information on certain design features such as the exact geometry of the Cold Neutron Source, other experiment facilities in the D_2O tank, and the angles of the beam tubes are not available from TUM and have been inferred from papers published by TUM. In all cases in this study, the HEU and LEU cores have been treated in a consistent manner and the results are expected to be valid if new details on the current TUM design are made available to us.

The design parameters for the Cold Neutron Source were taken from general specifications published by TUM in Ref. 12. These specifications are: (1) the shape is cylindrical with a diameter of 30 cm, (2) the volume of the moderator cell is 20 liters, (3) the moderator is deuterium with 5 wt-% hydrogen, (4) the zircaloy shell of the cold source is 0.5 mm thick, (5) the distance from the centerline of the core to the centerline of the cold source is 40 cm. The re-entrant hole in the CNS facing beam tubes 1 and 2 was not modeled because this information was not available to us.

Fuels Used in This Study

The core design options that are possible and the neutron flux performance that can be achieved are dependent on the LEU fuels that are available and the uranium densities that can be achieved. The fuels that were used in this study are shown in Table 1. TUM plans to utilize HEU U_3Si_2-Al fuel with 3.0 and 1.5 gU/cm^3 in the HEU core, even though the HEU fuel with 3.0 gU/cm^3 has never been irradiation tested. The LEU designs used fuels with three different uranium compounds in the dispersed phase (U_3Si_2, U_3Si, and U-6Mo). Each dispersant has a higher density and is capable of achieving higher uranium densities in the fuel meat for the same volume percent of dispersed phase. The U_3Si_2-Al and U_3Si-Al fuels have been irradiation tested and are suitable for use in the FRM-II without further testing. Microplates with U-Mo fuels have been

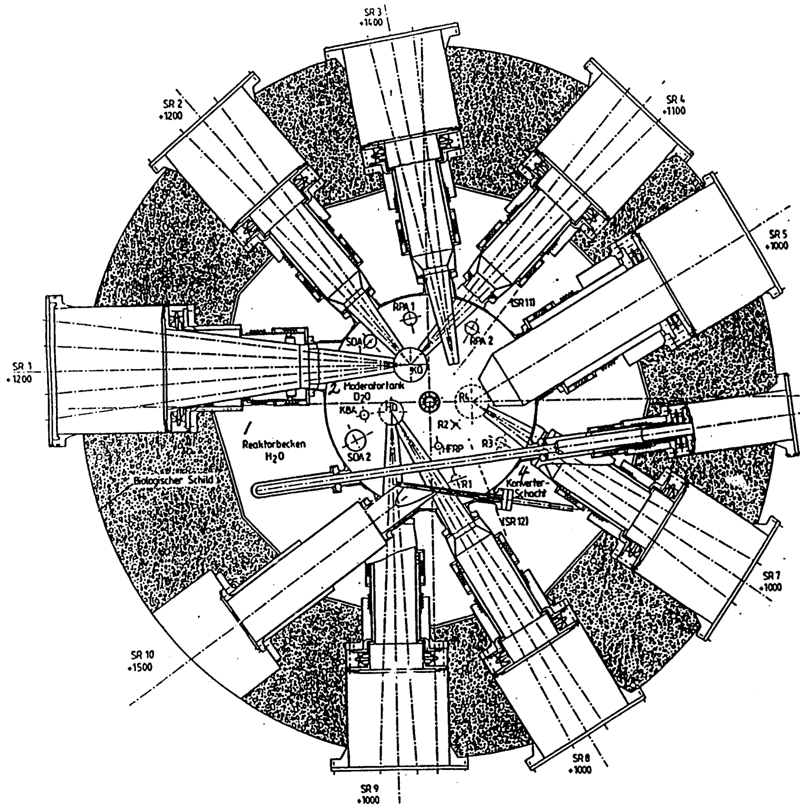

Figure 1: Horizontal Section Through TUM FRM-II
(Reproduced from Figure 4 of Ref. 7)

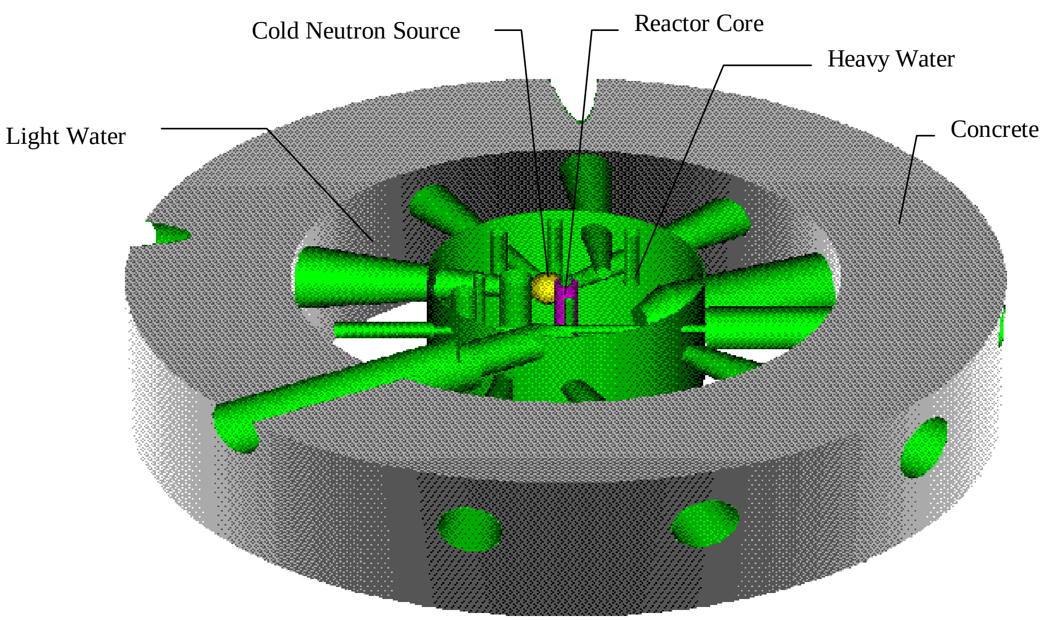

Figure 2. View of Model Used in MCNP Analyses

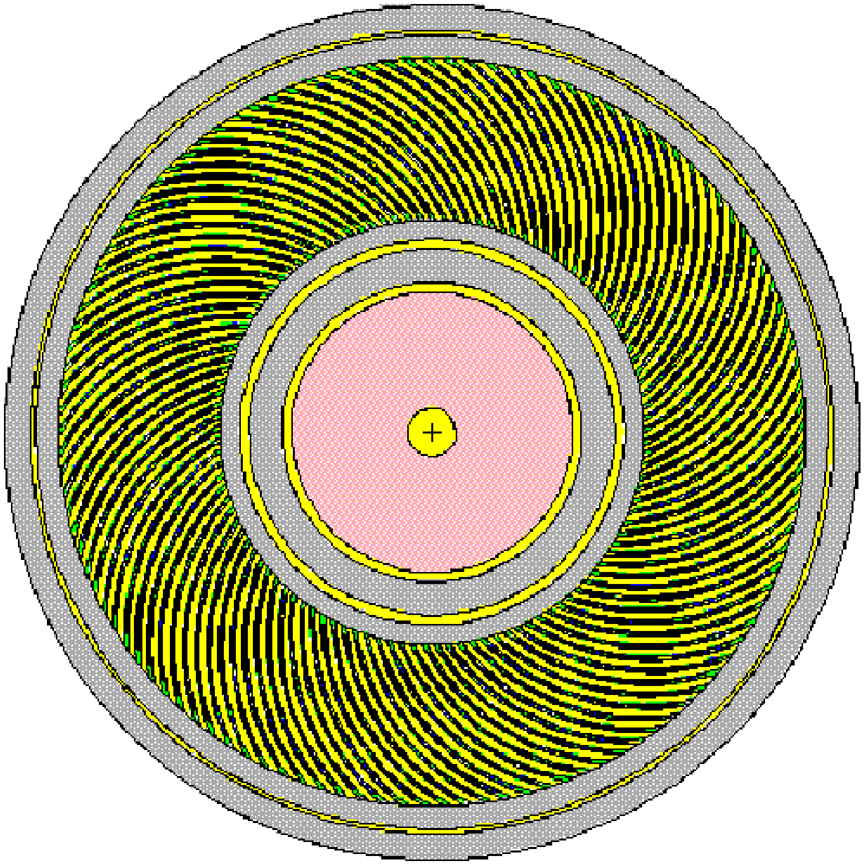

Figure 3. FRM-II HEU Core Modeled Using Arc-Type Fuel Plates
Instead of Involute-Type Fuel Plates with the MCNP Monte Carlo Code

irradiation tested[13] to about 70% ^{235}U burnup in the ATR reactor in Idaho. Post-irradiation examination[14] of these microplates has been very encouraging. Further tests[15] of microplates are planned in the ATR beginning in October 1999. Tests of full-sized plates and fuel assemblies are planned in several European research reactors.

Table 1. Fuels Used in this Study

Enrich-ment	Fuel Type	Uranium Density, g/cm^3	Volume-% Dispersed Phase
HEU (93%)	U_3Si_2-Al	1.5 & 3.0	13 - 27
LEU (19.75%)	U_3Si_2-Al	4.5 - 5.8	40 - 51
LEU (19.75%)	U3Si-Al	5.8 - 6.7	40 - 46
LEU (19.75%)	U-6Mo-Al	6.6 - 7.5	40 - 45

CERCA, the fuel fabricator for FRM-II, does not currently fabricate U_3Si-Al fuel because it has no customers for this fuel. CERCA is also not equipped to manufacture production quantities of U_3Si powder. However, AECL Canada has produced LEU U_3Si powder on a commercial scale[16] for use in the 125 MW NRU research reactor in Canada and the 30 MW HANARO research reactor in the Republic of Korea since 1991. As of May 1999, approximately 2,200 kg of U_3Si powder had been produced with excellent results. The AECL powder production facilities currently have excess capacity and could supply LEU U_3Si powder to CERCA for use in manufacturing fuel plates for the FRM-II reactor.

LEU U_3Si-Al fuel was chosen as a candidate fuel for the FRM-II only because the ^{235}U burnup anticipated in the FRM-II reactor fuel is so low. The average ^{235}U burnup is only ~17% and the peak ^{235}U burnup is only ~42%. In the 1980s, CERCA manufactured four full-sized test plates with LEU U_3Si-Al fuel and uranium densities of 5.5 and 6.0 g/cm^3 that were successfully irradiated[17] to 53-54% average ^{235}U burnup in the SILOE reactor in France. CERCA also manufactured a full-sized element with LEU U_3Si-Al fuel and 6.0 gU/cm^3 that was successfully irradiated in SILOE to 55% average burnup. In addition, 48 U_3Si-Al miniplates with uranium densities from 4.8-7.1 g/cm^3 were fabricated by ANL, NUKEM, and CNEA and irradiation tested in the Oak Ridge Research Reactor[18] in the U.S. and in the FRJ-2 reactor[19] in Germany, many to ^{235}U average burnups of over 80%. On this basis, the RERTR Program considers that LEU U_3Si-Al fuel is a viable candidate for use in the FRM-II.

Alternative LEU Core Designs

Table 2 contains information on the core designs and performance for the TUM HEU design with a power level of 20 MW, LEU cores with a power level of 20 MW, and optional LEU upgrade cores with a power level of 22 MW. Calculations were performed for active fuel meat heights of 70 cm and 80 cm. The objective was to maintain as many features of the HEU design as practical and change only those features that would be necessary to use LEU fuel. There is an optional 22 MW upgrade core with the same geometry and higher uranium density in the fuel meat for each of the 20 MW cores. In each case, the design parameters were chosen to match the 50-day cycle length of the HEU core and to maximize the thermal neutron flux in the Cold Neutron Source and the beam tubes.

LEU Fuel Meat Substitution in HEU Design

Only by increasing the size of the HEU core will it ever be possible to use LEU fuel in the FRM-II and have comparable core lifetime and experiment performance. There is no possibility whatsoever that a suitable LEU fuel will ever be developed for direct substitution into the fuel plates of the HEU core. To illustrate this point, calculations were done (See Case 2 in Table 2) in which LEU U-6Mo dispersion fuel with a uranium density of 9.0 g/cm^3 was substituted for the fuel meat of the HEU design. This core would have an LEU/HEU thermal neutron flux ratio of ~0.99 in the CNS, but would operate for less than 1 full power day at a power level of 20 MW. Using a completely hypothetical fuel with 12.0 gU/cm^3, the core would operate for less than 5 full power days at a power level of 20 MW.

Table 2. Alternative LEU Designs with Power Levels of 20 MW and 22 MW

Case	Enr., %	Power, MW	Core Ass'y Inner Diam., Cm (1)	Core Ass'y Outer Diam., Cm (1)	Active Fuel Height cm	Active Core Volume liters	Fuel Type	Full Power Days	Uranium Density, g/cm^3	Peak Fuel Temp., C w & w/o HCF (2)	LEU/HEU CNS Flux Ratio (3)	LEU/HEU Unpert. Peak Thermal Flux Ratio	LEU/HEU CNS Heating Ratio
HEU Core Geometry with 20 MW Power													
1	93	20	11.8	24.3	70	17.6	U_3Si_2	50	3.0/1.5	118/150	-	-	1.0
2	19.75	20	11.8	24.3	70	17.6	U-6Mo	<1	9.0 (4)	-	0.99		-
LEU Cores with 20-22 MW Power and 70 cm Active Fuel Height													
3	24	20	14.36	29.4	70	27.4	U_3Si_2	50	4.8	121/151	0.86	0.80	
4	19.75	20	14.36	29.4	70	27.4	U_3Si_2	42	5.8	149/186	0.86	0.79	
5	19.75	20	14.36	29.4	70	27.4	U_3Si	50	6.2	121/151	0.85	0.79	0.97
6	19.75	20	14.36	29.4	70	27.4	U-6Mo	50	6.6	116/142	0.84	0.77	0.95
7	19.75	22	14.36	29.4	70	27.4	U_3Si	50	6.6	137/156	0.93	0.85	1.05
8	19.75	22	14.36	29.4	70	27.4	U-6Mo	50	7.0	125/157	0.92	0.84	1.05
LEU Cores with 20-22 MW Power and 80 cm Active Fuel Height													
9	19.75	20	14.36	29.4	80	31.3	U_3Si_2	45	4.8	115/142	0.84	0.77	
10	19.75	20	14.36	29.4	80	31.3	U_3Si_2	50	5.1	124/154	0.83	0.76	
11	19.75	20	14.36	29.0	80	29.9	U_3Si_2	50	5.4	132/164	0.84	0.77	
12	19.75	20	14.36	28.6	80	28.6	U_3Si_2	50	5.8	144/179	0.85	0.79	
13	19.75	20	14.36	28.6	80	28.6	U_3Si	50	5.8	113/140	0.85	0.79	0.92
14	19.75	20	14.36	27.9	80	26.3	U_3Si	50	6.7	135/169	0.86	0.81	0.88
15	19.75	20	14.36	27.9	80	26.3	U-6Mo	50	7.0	122/152	0.86	0.79	0.90
16	19.75	22	14.36	28.6	80	28.6	U_3Si	50	6.1	124/156	0.93	0.86	1.02
17	19.75	22	14.36	27.9	80	26.3	U-6Mo	50	7.5	138/174	0.94	0.87	0.96

(1) Dimensions of the physical core assembly, including inner and outer sideplate structures.
(2) Peak fuel temperature with and without hot channel factors.
(3) LEU/HEU unperturbed thermal (<0.625 eV) neutron flux in the Cold Neutron Source.
(4) Core with 9.0 gU/cm^3 LEU fuel operates for less than 1 day at a power level of 20 MW (<5 days with 12 gU/cm^3 LEU fuel)

LEU Design Options

The availability of fuels strongly dictates core design possibilities. The major tradeoffs are on core diameter and height, power level, and thermal fluxes in the experiment regions. For the LEU cores, the inner diameter of the core was fixed at 14.36 cm and the outer core diameter, fuel type, and LEU density were adjusted to provide a compact core that would operate for 50 days and have peak fuel temperatures that are comparable with those of the HEU design.

Cases 3-8 in Table 2 represent cores with an active fuel height of 70 cm, power levels of 20 MW and 22 MW, and three potential fuel types. With 5.8 gU/cm^3 LEU U_3Si_2-Al fuel, the cycle length at a power level of 20 MW would be only 42 days, resulting in the use of six cores per year instead of five cores per year with the HEU design. A startup core with 24% enrichment in 4.8 gU/cm^3 U_3Si_2-Al fuel would allow operation for 50 full power days. Another alternative that would allow operation for 50 days at 20 MW is to utilize LEU U_3Si-Al fuel with 6.2 gU/cm^3 in the same core geometry. All three of these cores have an LEU/HEU thermal flux ratio of 0.85-0.86 in the Cold Neutron Source. The startup core with 24% enrichment would be operated only until LEU U-6Mo-Al fuel with 6.6 gU/cm^3 is adequately tested and licensed. This LEU fuel could be directly substituted into the same fuel plate and core geometry. Note also that the LEU/HEU ratio of gamma plus neutron heating in the Cold Neutron Source is 0.95-0.97 at a power of 20 MW. This lower heating in the CNS with LEU cores results from the larger ^{238}U content in LEU fuel, and hence the much larger electron density that gamma rays must pass through to escape into the moderator tank and be absorbed in the CNS.

After a few years operation at 20 MW, the power level could be upgraded to 22 MW by increasing the loading of the LEU U-6Mo-Al fuel to 7.0 gU/cm^3. The cycle length would still be 50 days at 22 MW and the LEU/HEU thermal flux ratio in the Cold Neutron Source would be 0.92-0.93. The gamma plus neutron heating in the CNS would be larger by about 5% with LEU fuel.

Cases 9-17 in Table 2 represent cores with an active fuel height of 80 cm, power levels of 20 MW and 22 MW, and the same three potential fuel types discussed in the preceding paragraph. The larger fuel height allows a 50-day cycle length to be achieved with lower uranium densities. It also allows the possibility of reducing the core diameter to increase the thermal flux in the experiment regions. Examination of these cases in Table 2 shows that the LEU/HEU thermal flux ratio in the Cold Neutron Source in the cores with an 80 cm active fuel height are essentially the same as those with a 70 cm active fuel height.

Figure 4 shows unperturbed neutron fluxes at a power level of 20 MW for the HEU design and a LEU design using U_3Si-Al fuel with 5.8 gU/cm^3 and an 80 cm active fuel height based on MCNP Monte Carlo calculations with a model that does not include the CNS and beam tubes. The figure shows that the LEU core has a lower thermal neutron flux, a slightly larger fast neutron flux, a lower total gamma flux, and lower gamma plus neutron heating at the location of the CNS and throughout the moderator tank. Figure 5 shows the neutron spectra in CNS with the HEU and LEU cores based on MCNP Monte Carlo calculations with the CNS and beam tubes in place. Overall, the neutron spectra in the CNS are nearly the same with the two cores. Both the HEU core and the LEU core would provide experimenters with excellent cold neutron beams for neutron scattering research.

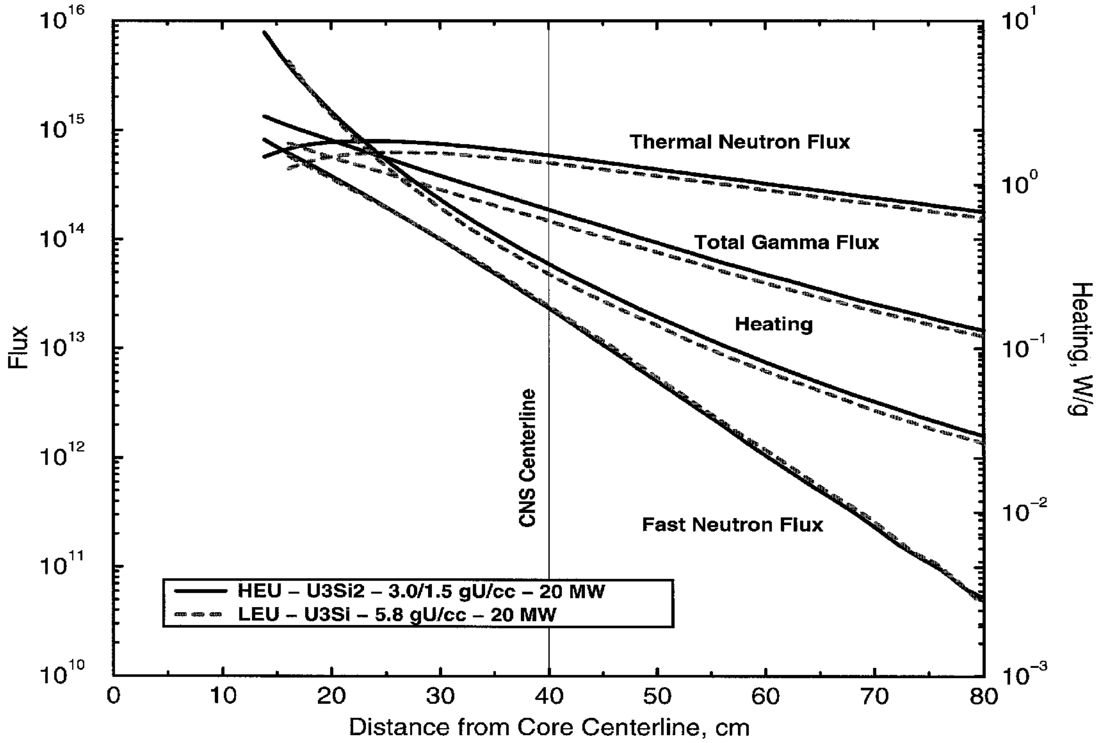

Figure 4. HEU and LEU (20 MW, 5.8 gU/cm^3 U_3Si-Al Fuel) Unperturbed Neutron Fluxes, Gamma Flux, and Heating in the Heavy Water Reflector Bases on MCNP Monte Carlo Calculations.

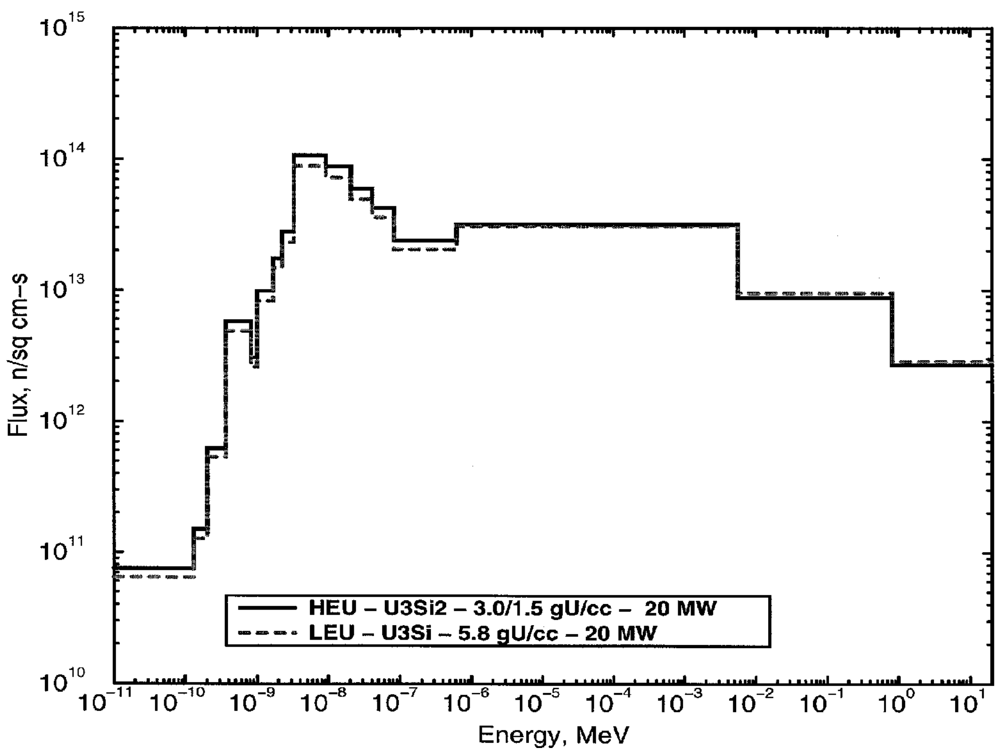

Figure 5. HEU and LEU (20 MW, 5.8 gU/cm^3, U_3Si-Al Fuel) Neutron Spectra in the Cold Neutron Source (Based on Monte Carlo Calculations Using the MCNP Code)

Reactivity Worth Comparisons

Table 3 shows that the reactivity worths of the CNS and the beam tubes in the HEU core and in LEU cores with an active fuel height of 70 cm are nearly identical. This reactivity worth is only about 0.2% Δk higher in the cores with 80 cm fuel height and is easily accommodated.

Table 3. Reactivity Worth of CNS and Beam Tubes in the HEU Core and the LEU Cores

Core	MW	Inner/Outer Core Diameter/Active Height	Fuel Type	U Density g/cm^3	React. Worth % Δk/k
HEU	20	11.8/24.3/70	U_3Si_2	3.0/1.5	3.33 ± 0.09
LEU	20	14.36/29.4/70	U_3Si	6.2	3.24 ± 0.09
LEU	22	14.36/29.4/70	U_3Si	6.6	3.25 ± 0.10
LEU	20	14.36/28.6/80	U_3Si	5.8	3.50 ± 0.09
LEU	20	14.36/27.9/80	U_3Si	6.7	3.49 ± 0.09
LEU	22	14.36/28.6/80	U_3Si	6.1	3.34 ± 0.09
LEU	22	14.36/27.9/80	U-6Mo	7.5	3.51 ± 0.10

The HEU and the 20-22 MW alternative LEU designs considered by ANL in this study have the same five safety rods that act as the primary shutdown system. Four of these rods are sufficient to scram the reactor. The design of the five safety rods in the heavy water reflector of the FRM-II HEU design was provided[20] by TUM near the end of April 1999. The safety rods have an outer diameter of 10 cm and the hafnium absorber is 1.0 cm thick. Using essentially the same design (i.e. maintaining dimensions, angles, and separation from the core vessel) these five safety rods were used in one LEU design (Core 13 in Table 2). Assuming that the most reactive rod is stuck out of the core, the results in Table 4 show that the same five safety rods designed for use in the HEU core can also be used in the LEU core. The reactivity worth of the four least reactive rods is smaller in the LEU core, but the shutdown margin is still more than adequate to satisfy the safety criteria.

Table 4. Reactivity Worth of Four of the Five Safety Rods

Core	MW	Inner/Outer Core Diameter/Active Height	Fuel Type	U Density g/cm^3	K-effective
HEU	20	11.8/24.3/70	U_3Si_2	3.0/1.5	0.9103 ± 0.0009 (- 9.86 % Δk/k)
LEU	20	14.36/28.6/80	U_3Si	5.8	0.9291 ± 0.0010 (- 7.64 % Δk/k)

Safety-Related Issues

The following issues were addressed in detail for the ANL LEU design[2-6] with a power level of 32 MW: hydraulic stability of the involute plates; a hypothetical accident involving the moderator material; transients due to uncontrolled withdrawal of the central control rod;

transients due to loss of primary coolant flow; and the radiological consequences of a hypothetical accident involving release of fission products and plutonium.

Monte Carlo calculations with the CNS and beam tubes included were redone for one of the 20 MW LEU designs to evaluate the subcriticality margin for a hypothetical accident in which the heavy water reflector is replaced by light water. In both the HEU and LEU cores, it was assumed that the central control rod has its beryllium follower in the core in its most reactive configuration. The results in Table 5 show that the HEU design is subcritical by about 13.5% $\Delta k/k$ and that the 22 MW LEU design with 6.1 g/cm^3 fuel is subcritical by about 11.3 % $\Delta k/k$. Both the HEU and LEU designs satisfy this safety criterion.

Table 5. Reactivity Values for a Hypothetical Accident in which the Heavy Water Reflector is Replaced by Light Water

Core	MW	Inner/Outer Core Diameter/Active Height	Fuel Type	U Density g/cm^3	K-effective
HEU	20	11.8/24.3/70	U_3Si_2	3.0/1.5	0.8812 ± 0.0010 (- 13.5 % $\Delta k/k$)
LEU	22	14.36/28.6/80	U_3Si	6.1	0.8986 ± 0.0016 (- 11.3 % $\Delta k/k$)

Corresponding calculations for the other issues listed above have not yet been performed for the LEU designs discussed in this paper. However, we expect that all of the safety criteria for the 20-22 MW LEU designs will be satisfied. For example, the 32 MW design was calculated to contain about 158 grams of total plutonium at end of cycle. Analyses concluded that the increased plutonium and fission product inventory in the LEU core would have no impact on the radiological consequences of hypothetical accidents involving melting of the core in water, even with very conservative release assumptions. The 32 MW LEU design would have met the radiological consequence criteria set by the BMU.

The LEU designs considered in this paper were calculated to contain 108-116 grams of total plutonium at a power of 20 MW and 122-132 g of plutonium at a power of 22 MW. Since the plutonium and fission product inventories are much less at 20-22 MW than at 32 MW, we expect that all of the 20-22 MW designs will meet the radiological consequence criteria set by the BMU.

Conclusion

Based on the excellent results that were obtained for the alternative LEU designs with power levels of 20 - 22 MW, the RERTR Program concludes that it is feasible to use LEU fuel instead of HEU fuel in the FRM-II with relatively small changes in the core geometry. Several designs are feasible. The thermal neutron flux in the Cold Neutron Source of the LEU design with a power level of 20 MW is expected to be about 85-86% of the HEU design. At a power level of 22 MW, this value is expected to be about 92-93% of the HEU design. All safety criteria for the LEU designs at power levels of 20 MW and 22 MW are expected to be satisfied. The RERTR Program would like to reiterate its strong support for construction of the FRM-II reactor using

LEU fuel and its readiness to exchange information with the TUM to resolve any technical issue that may still exist.

Key parameters of the 20 MW HEU design and two ANL alternative LEU designs with power levels of 20 MW and 22 MW are summarized in Table 6.

Table 6. Key Parameters of the FRM-II HEU Design and Two Alternative LEU Designs with Power Levels of 20 MW and 22 MW.

Parameter	TUM HEU Design	ANL Alternative LEU Design #1			ANL Alternative LEU Design #2	
	20 MW	20 MW	20 MW	22 MW	20 MW	22 MW
Number of Fuel Elements	One	One	One	One	One	One
Core Lifetime (Full Power Days)	50	50	42	50	50	50
LEU/HEU Thermal flux in CNS	1.00	0.86	0.86	0.92	0.85	0.93
Fuel Type	U_3Si_2	U_3Si_2	U_3Si_2	U-6Mo	U_3Si	U_3Si
Uranium Density, g/cm^3	3.0 & 1.5	4.8	5.8*	7.0	5.8	6.1
Uranium Enrichment, % ^{235}U	93	24	19.75	19.75	19.75	19.75
Outer Core Diameter, cm	24.3	29.4	29.4	29.4	28.6	28.6
Active Fuel Height, cm	70	70	70	70	80	80
Fuel Meat/Clad Thickness, mm	0.60/0.38	0.76/0.38	0.76/0.38	0.76/0.38	0.76/0.38	0.76/0.38
Water Channel Thickness, mm	2.2	2.2	2.2	2.2	2.2	2.2
All Safety Criteria Satisfied?	Yes	Very Likely	Very Likely	Very Likely	Very Likely	Very Likely

* LEU U_3Si-Al fuel with a uranium density of 6.2 g/cm^3 will allow operation for 50 days.

REFERENCES

1. K. Boening and J. Blombach, “Design and Safety Features of the Planned Compact Core Research Reactor FRM-II,” Proceedings of the XIV International Meeting on Reduced Enrichment for Research and Test Reactors, Jakarta, Indonesia, 4-7 November 1991.

2. N. A. Hanan, S. C. Mo, and J. E. Matos, “Transient Analyses for HEU and LEU Designs of the FRM-II”, Proceedings of the 1998 International Meeting on Reduced Enrichment for Research and Test Reactors, Sao Paulo, Brazil, 18 -23 October 1998. See also: http://www.td.anl.gov/RERTR/RERTR.html

3. N. A. Hanan and J. E. Matos, “Fluxes at Experiment Facilities in HEU and LEU Designs for the FRM-II”, Proceedings of the 1997 International Meeting on Reduced Enrichment for Research and Test Reactors, Jackson Hole, Wyoming, USA, 5 -10 October 1997. See also: http://www.td.anl.gov/RERTR/RERTR.html

4. N. A. Hanan, S. C. Mo, R. S. Smith, and J. E. Matos, “An Alternative LEU Design for the FRM-II,” ANL/RERTR/TM-27, October 1996. See also: http://www.td.anl.gov/RERTR/RERTR.html

5. N. A. Hanan, S. C. Mo, R. S. Smith, and J. E. Matos, "An Alternative LEU Design for the FRM-II," Proceedings of the XIX International Meeting on Reduced Enrichment for Research and Test Reactors, Seoul, Korea, 7-10 October 1996. See also: http://www.td.anl.gov/RERTR/RERTR.html

6. S. C. Mo, N. A. Hanan, and J. E. Matos, "Comparison of the FRM-II HEU Design With an Alternative LEU Design," and N. A. Hanan, S. C. Mo, R. S. Smith, and J. E. Matos, "Attachment to Comparison of the FRM-II HEU Design With an Alternative LEU Design," Proceedings of the XVIII International Meeting on Reduced Enrichment for Research and Test Reactors, Paris, France, 17-21 September 1995. See also: http://www.td.anl.gov/RERTR/RERTR.html

7. "Neutron Source Munich FRM-II (Neutronenquelle Munchen FRM-II): Project Status Report," presented by Project Group "New Research Reactor" of the Department of Physics E21 of the Technical University of Munich, ORNL/TR-92/17, March 1992.

8. J. F. Briesmeister, ed., "MCNP- A General Monte Carlo N-Particle Transport Code, Version 4B" LA-12625-M, Los Alamos National Laboratory, 1997

9. B.J. Toppel, "A User's Guide for the REBUS-3 Fuel Cycle Analysis Capability", ANL-83-2 (March 1983).

10. K. L. Derstine, "DIF3D: A Code to Solve One, Two, and Three-Dimensional Finite-Difference Diffusion Theory Problems," Argonne National Laboratory Report ANL-82-64 (April 1984).

11. J. R. Deen, W. L. Woodruff, C. I. Costescu, and L.S. Leopando, "WIMS-ANL User Manual, Rev. 3," ANL/RERTR/TM-23 (March 1999).

12. K. Gobrecht, Technical University of Munich, "Progress on the Cold Neutron Source of the Garching Neutron Research Facility FRM-II", Proceedings of the 6th Meeting of the International Group on Research Reactors, Taejon, The Republic of Korea, April 29 – May 1, 1998, p. 377.

13. M.K. Meyer, J.L. Snelgrove, G.L. Hofman, and S.L. Hayes, "US-RERTR Advanced Fuel Development Plans: 1999", Proceedings of the 1998 International Meeting on Reduced Enrichment for Research and Test Reactors, Sao Paulo, Brazil, 18 -23 October 1998. See also: http://www.td.anl.gov/RERTR/RERTR.html

14. S.L. Hayes, M.K. Meyer, G.L. Hofman, and R.V. Strain, "Postirradiation Examination of High Density Uranium Alloy Dispersion Fuels", ibid.

15. M.K. Meyer, S.L. Hayes, C.R. Clark, and T.C. Wiencek, "Prototypic Irradiation Testing of High-Density U-Mo Alloy Dispersion Fuels" (these proceedings).

16. Letter from R. J. Harrison, Branch Manager, AECL Fuel Fabrication Branch, and D. F. Sears, Section Head, Research Reactor Fuel, AECL Fuel Development Branch, to Dr. Wilfried Krull, GKSS, 12 May 1999.

17. C. Baas, M. Barnier, J.P. Beylot, P. Martel, and F. Merchie, "LEU and MEU Fuel Testing in CEA Reactors", published in Research Reactor Core Conversion Guidebook, Volume 4, Fuels, IAEA-TECDOC-643, April 1992, p. 315; and Proceedings of the 1986 International Meeting on Reduced Enrichment for Research and Test Reactors, Gatlinburg, Tennessee, November 3-6, 1986, p.231

18. "Miniplate Irradiations in the Oak Ridge Research Reactor", published in Research Reactor Core Conversion Guidebook, Volume 4, Fuels, IAEA-TECDOC-643, April 1992, p. 239

19. G. Thamm, E. Groos, and W. Krug, "LEU-Fuel Testing at KFA-Juelich Under the German AF-Program, Plate Irradiation and PIE", Proceedings of the 1987 International Meeting on Reduced Enrichment for Research and Test Reactors, Buenos Aires, Argentina, September 28 – October 1, 1987.

20. Letter from Klaus Boening, Technical University of Munich to Armando Travelli, RERTR Program, Argonne National Laboratory, 29 April 1999.