ATTACHMENT 2

Comparison of the FRM-II HEU Design With an Alternative LEU Design

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

Argonne National Laboratory
Argonne, Illinois 60439-4841 USA

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ABSTRACT

The Alternative LEU Design for the FRM-II proposed by the RERTR Program at Argonne National Laboratory (ANL) has a compact core consisting of a single fuel element that uses LEU silicide fuel with a uranium density of 4.5 g/cm^3 and has a power level of 32 MW. Both the HEU and LEU designs have the same fuel lifetime (50 days) and the same neutron flux performance (8×10^{14} $n/cm^2/s$ in the reflector). LEU silicide fuel with 4.5 g/cm^3 has been thoroughly tested and is fully-qualified, licensable, and available now for use in a high flux reactor such as the FRM-II.

The following issues raised by the Technical University of Munich (TUM) were addressed in Reference 1: qualification of HEU and LEU silicide fuels, gamma heating in the heavy water reflector, radiological consequences of larger fission product and plutonium inventories in the LEU core, and cost and schedule. The conclusions of these analyses are summarized below. This Attachment addresses three additional safety issues that were raised by TUM in Reference 2: stability of the involute fuel plates, a hypothetical accident involving the configuration of the reflector, and a loss of primary coolant flow transient due to an interrupted power supply.

Based on the excellent results for the Alternative LEU Design that were obtained in these analyses, the RERTR Program concludes that all of the major technical issues regarding use of LEU fuel instead of HEU fuel in the FRM-II have been successfully resolved and that it is definitely feasible to use LEU fuel in the FRM-II without compromising the safety or performance of the facility. In this regard, the RERTR Program would like to reiterate its strong support for construction of the FRM-II reactor using LEU silicide fuel and its readiness to exchange information with the TUM to resolve any technical issues that may still exist.

INTRODUCTION

The basic parameters of the FRM-II HEU and Alternative LEU designs are shown in Table 1 (Ref. 1). As part of our evaluation of the hydraulic stability of the LEU involute fuel plates, results are also shown for an LEU design which has the same 8.735 cm fuel plate width as the lower core of the Advanced Neutron Source (ANS) reactor designed by the Oak Ridge National Laboratory (ORNL). In ANL's LEU design for FRM-II, a thicker plate, a thicker water channel, and a lower coolant velocity serve to increase the hydraulic stability over that of an already stable ANS design.

Only by increasing the size of the HEU core is it possible to use LEU fuel in the FRM-II and have a comparable core lifetime and experiment performance. There is no possibility whatsoever that a suitable LEU fuel will be developed for the HEU geometry. To illustrate this point, calculations were done in which LEU uranium metal with a density of 19 g/cm^3, a totally unrealistic possibility, was substituted for the fuel meat of the HEU design. The result was that the core would operate for only about 25 days at a power level of 20 MW and would have a peak thermal flux of 7×10^{14} n/cm^2-s in the heavy water reflector. This performance level would not be acceptable.

Table 1: Key Parameters in the FRM-II HEU Design, the Second Alternative LEU Design, and an LEU Design that Has the Same Involute Plate Width as the Lower Core of ORNL's ANS Design.

	FRM-II HEU Design	2nd Alternative LEU Design	ANS Plate-Width LEU Design
Enrichment, %	93	20	20
Reactor Power (MW)	20	32	32
Cycle Length (Full Power Days) (a)	50	50	50
Average Number of Cores/Year (b)	5.0	5.0	5.0
Peak Thermal Flux, $k_{eff}\bullet\varphi th,max$ (n/cm^2/s)	8.0×10^{14}	8.1×10^{14}	8.0×10^{14}
Reflector Volume (liters) with $k_{eff}\bullet\varphi th>7x10^{14}$ n/cm^2/s	82	146	117
Active Core Inner - Outer Radius (cm)	6.75 - 11.2	9.78 - 16.04	10.45 - 16.55
Active Core Height (cm)	70	80	80
Active Core Volume (liters)	17.6	40.6	41.4
Number of Fuel Plates	113	161	172
Core Loading (Kg U-235)	7.5	7.5	7.6
Fuel Type	U_3Si_2	U_3Si_2	U_3Si_2
Fuel Grading	Yes	No	No
Fuel Meat Uranium Density (g/cm^3)	3.0/1.5	4.5	4.5
Fuel Meat/Clad Thickness (mm)	0.60/0.38	0.76/0.38	0.76/0.38
Coolant Channel Thickness (mm)	2.2	2.2	2.2
Length of Involute Plate (cm)	6.83	9.15	8.735
K_{eff} at BOC	1.1937	1.2334	1.2322
Core Average Burnup (% U-235 burned)	17.3	26.5	25.9
Average Fission Rate in Fuel Meat (fissions/cm^3/s) Peak Pointwise Fiss. Rate in Fuel Meat at BOC (c)	2.1×10^{14} 4.7×10^{14}	1.2×10^{14} 2.9×10^{14}	1.2×10^{14} 2.9×10^{14}
Average Fission Density in Fuel Meat (fissions/cm^3) Peak Fission Density in Fuel Meat at EOC (c)	1.0×10^{21} 1.5×10^{21}	0.5×10^{21} 0.9×10^{21}	0.5×10^{21} 0.9×10^{21}
Average Power Density in Core (W/ cm^3) Peak Power Density in Core - rod out at BOC	1139 2537	788 1919	773 1872
Peak Temperature in Fuel Meat (°C) BOC/EOC	150/180	130/160	130/160 (d)

(a) EOC excess reactivity = 7% Δk/k; (b) Based on 250 days operation per year; (c) In 3.0 g/cm^3 fuel of the HEU design. (d) Estimated.

The following paragraphs summarize the conclusions of the analyses presented in Ref. 1.

(1) Qualification of HEU and LEU Silicide Fuels

HEU silicide fuel (U_3Si_2-Al) with 93% enrichment and a uranium density of 3.0 g/cm^3 is totally untested and is not likely to be licensable without specific test data to qualify the fuel for use in the FRM-II. Normal licensing practices in many countries require that tests be performed on the specific fuel that will be used in a reactor in order to provide the data on fuel behavior that is required for licensing.

LEU silicide fuel (U_3Si_2-Al) with uranium densities up to 4.8 g/cm^3 is fully-qualified for conditions close to those of the FRM-II LEU design. The fuel was qualified by means of extensive irradiation testing and post-irradiation examination of miniature fuel plates, full size elements, and a whole-core demonstration. This fuel is available today and can be licensed for routine use today in the FRM-II.

(2) Gamma Heating in the Heavy Water Reflector

Detailed analyses comparing the energy deposited (gamma heating) in the heavy water reflector of both the FRM-II HEU design and the alternative LEU design showed that a cold source operating in the heavy water reflector of the LEU design would make a superb experimental facility even though the gamma heating would be slightly higher than in the HEU design. At a distance of 50 cm from the reactor vessel, the gamma heating in the HEU design would be a factor of 2.1 times lower than in the RHF reactor at Grenoble, France, and the gamma heating in the LEU design would be a factor of 1.8 lower than in the RHF.

(3) Radiological Consequences

Analyses of the radiological consequences of increased plutonium production in LEU fuel and larger fission product inventory in the higher-powered alternative LEU design for the case of hypothetical accidents involving core melting show that the alternative LEU design meets in full the radiological consequences criteria set by the German Ministry of Environment (Bundesministerium fur Umwelt - BMU).

The plutonium that would be produced in the HEU and LEU cores were calculated to be 10.4 g and 158.5 g, respectively. Analyses performed in Ref. 1 showed that the increased plutonium 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. Analyses in Ref. 1 also showed that the radiological consequences for a wet core melt with either the HEU design or the alternative LEU design are within the norms established by the BMU.

(4) Cost and Schedule

The design features and results obtained by ANL for the alternative LEU design were very different from those used by TUM in its assessment of the costs involved in using LEU fuel in the FRM-II. Thus, a careful review of both cost and schedule issues was thought to be important.

This Attachment

This Attachment addresses three additional safety issues that were raised by TUM in Reference 2. These issues are: (1) stability of the involute fuel plates, (2) a hypothetical accident involving the moderator material in the reflector, and (3) a loss of primary coolant flow transient due to an interrupted power supply (station blackout).

Fuel Element Hydraulic Instability

In Reference 2, TUM refers to the ANL alternative LEU design with an involute-type fuel plate having a width of 9.15 cm and a water velocity of 18 m/s and states: "Even if the somewhat lower power density and, therefore, coolant velocity is taken into account, this large value of the plate width could never guarantee the required plate stability." ANL does not agree with this statement by TUM. The analyses presented below show that the fuel element of both the HEU design and the alternative LEU design have hydraulic stability margins that are more than adequate.

Reactors and Designs Using Involute-Type Fuel Plates

The 100 MW High Flux Isotope Reactor (HFIR) at ORNL has operated successfully since 1965 using involute plates having a width of 8.38 cm in the inner fuel element and 7.48 cm in the outer fuel element. The nominal light water coolant velocity is 15.5 m/s.

The RHF reactor located at the Institut Laue-Langevin in Grenoble, France, has operated successfully since 1971 using involute plates having a width of 7.59 cm. The nominal heavy water coolant velocity of 15.5 m/s.

The ANS reactor design at ORNL had a lower fuel element containing involute plates having a width of 8.735 cm and a thickness of 1.27 mm. The water channel thickness was 1.27 mm and the nominal water velocity was 24 m/s. Experiments and analyses performed at ORNL determined that the fuel plates in this design would be stable during operation (Ref. 3). The "ANS plate-width" LEU design for the FRM-II shown in Table 1 has fuel plates having the same width (8.735 cm), but the plate thickness is 1.52 mm, the water channel thickness is 2.2 mm, and the nominal coolant velocity is only 18 m/s. All three factors (a thicker plate, a thicker water channel, and a lower coolant velocity) will increase the hydraulic stability of these LEU fuel plates over that of the already stable ANS design.

Hydraulic Stability Analysis

The analyses presented below show that the involute-shaped fuel element used in the alternative LEU design has a large safety margin with respect to hydraulic instability.

To analyze the alternative LEU design for the FRM-II, a computer code was obtained from ORNL. This code (see Ref. 4, W. K. Sartory, "Analysis of Hydraulic Instability of ANS Involute Fuel Plates," ORNL/TM-11580) was one of the codes used in the ANS design to assess the hydraulic stability of the involute plates. After obtaining the ORNL code, the results presented in Figure 2 of ORNL/TM-11580 were reproduced to verify the correct use of the code. The code was then used to calculate the critical velocity for both the FRM-II HEU design and two alternative LEU designs, for the inner and outer fuel elements of the HFIR reactor at ORNL, for the RHF reactor in Grenoble, France, and for the upper and lower fuel elements of the ANS core designed by ORNL. These results are shown in Table 2. As seen from these data, the design coolant velocity for the FRM-II LEU design is smaller than the calculated critical velocity by a factor of about 3.7. These results show clearly that there is no hydraulic stability problem with the alternative LEU design for the FRM-II.

It is important to note that the critical velocity for the lower fuel element of the ANS is calculated to be about 47 m/s (about two times greater than the design velocity). Tests performed by ORNL (ORNL/TM-12353), "using full scale epoxy plate models of the aluminum/uranium silicide ANS involute-shaped fuel plates" show that if hydraulic instability were to occur, it would occur at a coolant velocity greater than the critical velocity predicted by the code. This indicates that the calculated results for the critical velocity presented in Table 2 are conservative and that both the HEU and alternative LEU designs have more than adequate hydraulic stability margins.

Table 2. Calculated Critical Coolant Velocity for Operating Reactors and Reactor Designs Using Involute-Shaped Fuel Plates.

Reactor or Reactor Design	Fuel Plate Thick., mm	Coolant Channel Thick., mm	Involute Plate Width, cm	Design Coolant Velocity m/s	Calculated* Critical Coolant Velocity, m/s
FRM-II HEU	1.36	2.20	6.83	18.0	89.9
FRM-II LEU 4.5 g/cm^3, Ref. 1	1.52	2.20	9.15	18.0	66.6
FRM-II LEU 4.5 g/cm^3, Same Plate-Width (8.735 cm) as ANS Lower Fuel Element	1.52	2.20	8.735	18.0	67.8
RHF Grenoble, HEU	1.27	1.80	7.59	15.5	73.8
HFIR Inner Fuel Element, HEU	1.27	1.27	8.38	15.5	58.8
HFIR Outer Fuel Element, HEU	1.27	1.27	7.48	15.5	65.8
ANS Lower Fuel Element, HEU Design	1.27	1.27	8.735	24.0	46.8
ANS Upper Fuel Element, HEU Design	1.27	1.27	7.03	24.0	64.6

*Calculations were performed using a code developed by ORNL. See ORNL-11580, Ref. 4.

Hypothetical Accident Involving the Moderator Material of the Reflector

In Reference 2, TUM correctly claims that in “mere light water the HEU fuel element would not get critical at all.” Analyses performed by ANL show that the same conclusion is also true for the alternative LEU fuel element.

Monte Carlo calculations were performed for FRM-II HEU design and the alternative LEU design to evaluate the subcriticality margins for a hypothetical accident in which the heavy water reflector is replaced by light water. Results of this analysis show that the HEU design is subcritical by about 16% $\Delta k/k$ and that the alternative LEU designs is subcritical by about 8% $\Delta k/k$. These results conservatively assume that the central control rod has its beryllium follower in the core in its most reactive configuration. As a result, both cores satisfy this safety criteria.

Loss of Primary Coolant Flow Transient

A loss of primary flow transient analysis for the FRM-II HEU design is described by TUM in Ref. 5. ANL has analyzed this transient for both the HEU and alternative LEU designs using essentially the same assumptions as in Ref. 5 and concludes that fuel integrity is maintained with a considerable safety margin in both cases. Decay heat can be removed by natural circulation from both the HEU and LEU cores for at least seven days, making a strong inherent safety case for both designs.

Transient Description

Based on Ref. 5, "A loss of offsite power causes the simultaneous loss of all four primary pumps. In order to provide sufficient time to detect the accident and shutdown the reactor with the safety rods, the pumps are equipped with flywheels. Including uncertainties, the reactor trip point of the mass flow signal is reached 1.8 seconds after the loss of the pumps and the reactor is shutdown 0.5 seconds later." The following assumptions were made in the ANL analyses because detailed design information was not available:

(1) Before initiation of the transient, the HEU design was operated for 50 days at its nominal power of 20 MW and the LEU design was operated for 50 days at its nominal power of 32 MW. These conditions were used to generate the power history for the decay heat in the HEU and LEU cases.
(2) All decay heat (gamma and beta) is deposited in the fuel. That is, the peak power profile is the same as for the reactor during operation. Actually, a major fraction of the gamma power is deposited outside of the core in the heavy water reflector and reactor shielding.
(3) The light water pool has a diameter of 5 m and a depth of 14 m (Ref. 6),
(4) The initial light water pool temperature is 37°C (Ref. 6),
(5) The DC-driven pumps have a flow rate equivalent to 6% of the AC-driven pumps,
(6) The DC-driven pumps operate for only three hours after onset of the transient (Ref. 5).
(7) There is no heat loss from the light water pool during the transient.
(8) At 100 seconds after the onset of the transient, the battery-supplied emergency core cooling pumps are started to maintain the forced flow for three hours. In Ref. 5, these pumps are started at the time of the trip, at about 1.8 seconds after the transient is initiated. Thereafter, the natural circulation flaps open automatically and the decay heat is removed by natural circulation.

Results

The safety margin against flow instability reaches its lowest value of 4.0 (3.5 in Ref. 5) for the HEU design and 4.4 for the alternative LEU design after 2.3 seconds. (The value of the bubble detachment parameter eta was taken to be 47.8 in calculating the margin against flow instability, as suggested in Ref. 7 by Interatom, now Siemens). These safety margins of 4.0 for the HEU design and 4.4 for the LEU design are to be compared with the minimum required value of 1.5 (Ref. 5). It follows that the fuel integrity is maintained with a considerable safety margin in both the HEU and alternative LEU designs.

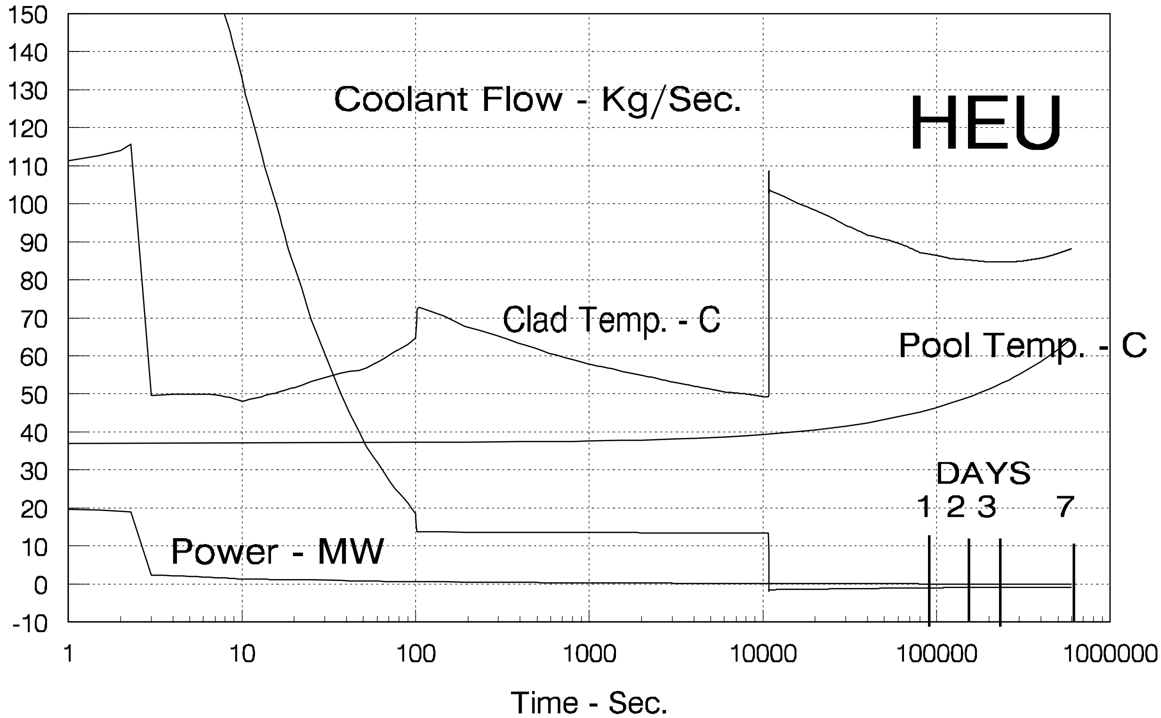
Additional results of this analysis are plotted in Fig. 1 for both the HEU and alternative LEU designs. During the first seven days after initiation of the transient: (1) the temperature of the cladding in both cores is less than 120°C, far below the clad melting temperature of about 580°C and (2) the temperature of the light water pool is about 80°C in the alternative LEU design and about 60°C in the HEU design. As a result, the decay heat can be safely removed from the core by natural circulation for at least seven days, making a strong inherent safety case for both designs.

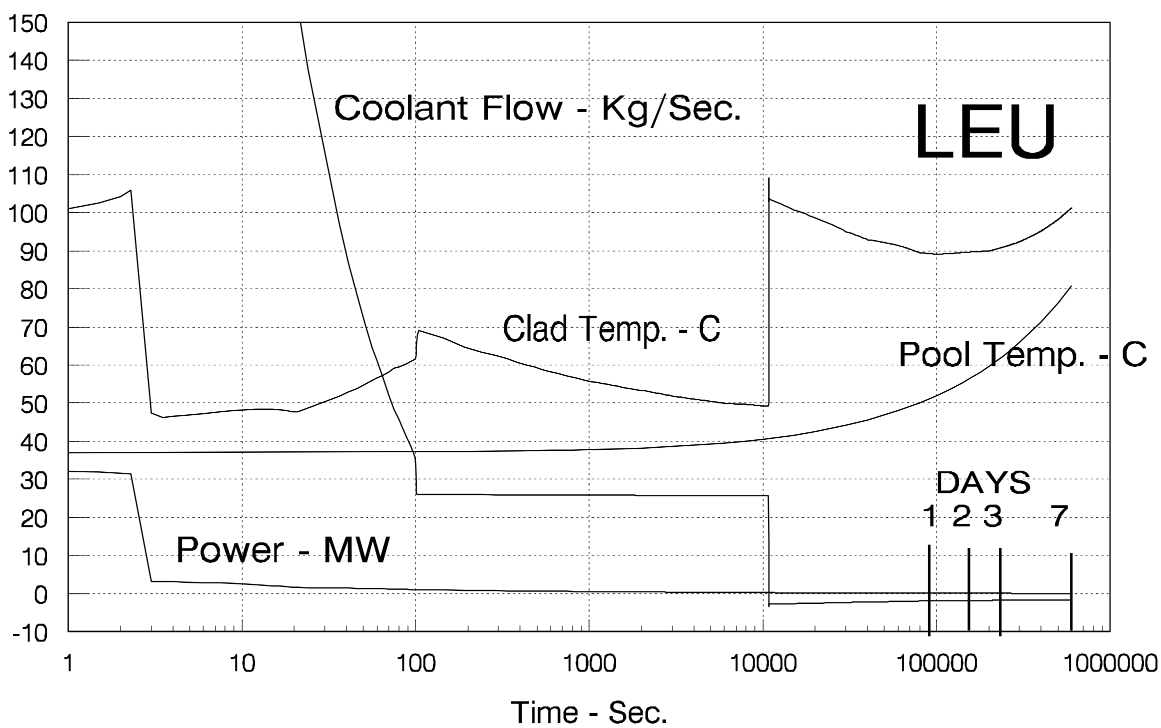
Conclusion

Based on the excellent results for the Alternative LEU Design that were obtained in these analyses, the RERTR Program concludes that all of the major technical issues regarding use of LEU fuel instead of HEU fuel in the FRM-II have been successfully resolved and that it is definitely feasible to use LEU fuel in the FRM-II without compromising the safety or performance of the facility. In this regard, the RERTR Program would like to reiterate its strong support for construction of the FRM-II reactor using LEU silicide fuel and its readiness to exchange information with the TUM to resolve any technical issues that may still exist.

Figure 1. Loss of Primary Flow Transient Analysis for FRM-II HEU and Alternative LEU Designs

(DC pump operation at 6% flow for 3 hours)





REFERENCES

1. 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 1995 International Meeting on Reduced Enrichment for Research and Test Reactors, 18-22 September 1995, Paris, France (to be published).

2. K. Böning, “Comment on the Contribution of S.C. Mo, N.A. Hanan and J.E. Matos: Comparison of the FRM-II HEU Design With an Alternative LEU Design”, Proceedings of the 1995 International Meeting on Reduced Enrichment for Research and Test Reactors, 18-22 September 1995, Paris, France (to be published).

3. W. F. Swinson, R.L. Battiste, C.R. Luttrell, and G.T. Yahr, “Fuel Plate Stability experiments and Analysis for the Advanced Neutron Source”, ORNL/TM-12353, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, May 1993.

4. W.K. Sartory, “Analysis of Hydraulic Instability of ANS Involute Fuel Plates”, ORNL/TM-11580, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, November 1991.

5. K. Böning and J. Bombach, “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, 4-7 November 1991, Jakarta, Indonesia, published by Badan Tenaga Atom Nasional, 1995, p. 367.

6. “Neutron Source Munich FRM-II, Project Status Report presented by project group “New Research Reactor” of the Department of Physics E21, Technical University of Munich, March 1992, ORNL/TR-92/17.

7. “INTERATOM: Safety Analyses for the IAEA Generic 10 MW Reactor“ IAEA Research Reactor Core Conversion Guidebook, IAEA-TECDOC-643, April 1992, Volume 2, Analysis, page 15.