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AN ALTERNATIVE LEU DESIGN FOR THE FRM-II

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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 design by the Technical University of Munich (TUM) and the alternative LEU design by ANL have the same fuel lifetime (50 days) and the same neutron flux performance ($8 \times 10^{14} \text{ n/cm}^2/\text{s}$ 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.

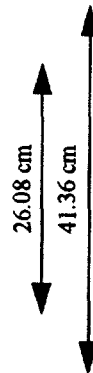
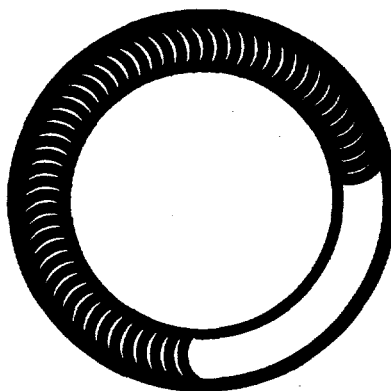
The following issues raised by TUM were addressed in Ref. 1: qualification of HEU and LEU silicide fuels, gamma heating in the heavy water reflector, radiological consequences of larger fission product and plutonium inventories in the LEU core, and cost and schedule. The conclusions of these analyses are summarized below. This paper addresses three additional safety issues that were raised by TUM in Ref. 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. Calculations were also done to address the possibility that new high density fuels could be developed that would allow conversion of the TUM HEU design to LEU fuel.

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.

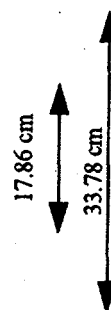
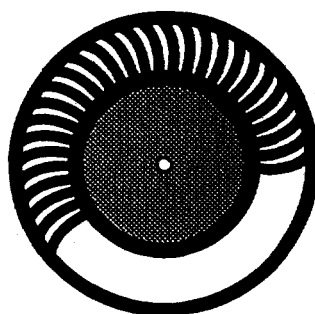
INTRODUCTION

Schematic diagrams of the FRM-II HEU and Alternative LEU designs are compared in Fig. 1 with the RHF reactor in Grenoble, France. Basic parameters of the FRM-II designs are shown in Table 1. Computer models for the HEU and LEU designs have been exchanged^[3,4] between TUM and ANL. As a result, consistent modifications were made to the aluminum-alloy compositions, the outer side plate thickness, and the control rod design in both the HEU and LEU cores. However, conclusions of the analyses were unchanged due to these modifications.

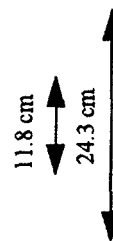
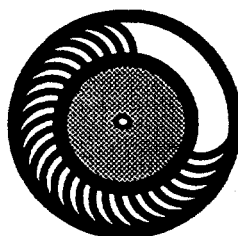
As part of our evaluation of the hydraulic stability of the LEU involute fuel plates, results are also shown for an LEU design which has the same 8.735 cm fuel plate width as the lower core of the Advanced Neutron Source (ANS) reactor designed by the Oak Ridge National Laboratory (ORNL). In ANL's LEU 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.



Grenoble RHF



FRM-II Alternative LEU Design



FRM-II HEU Design

Figure 1 Horizontal Sections of FRM-II HEU Core, Alternative LEU Core, and the Grenoble RHF.

For this reason, the ANS plate-width design is our preferred LEU design and the calculations reported in this paper were performed using this LEU model.

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, %	93.0	19.75	19.75
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, $k_{eff} \cdot \phi_{th,max}$ (n/cm ² /s)	8.0×10^{14}	8.1×10^{14}	8.0×10^{14}
Reflector Volume (liters) with $k_{eff} \cdot \phi_{th} > 7 \times 10^{14}$ n/cm ² /s	75	135	110
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	7.5
Fuel Type	U ₃ Si ₂	U ₃ Si ₂	U ₃ Si 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	1.1714	1.2105	1.2101
Core Average Burnup (% U-235 burned)	17.3	26.7	25.9
Average Fission Rate in Fuel Meat (fissions/cm ³ /s)	2.1×10^{14}	1.2×10^{14}	1.2×10^{14}
Peak Pointwise Fission Rate in Fuel Meat at BOC (c)	4.6×10^{14}	2.8×10^{14}	2.8×10^{14}
Average Fission Density in Fuel Meat (fissions/cm ³)	1.0×10^{21}	0.5×10^{21}	0.5×10^{21}
Peak Fission Density in Fuel Meat at EOC (c)	1.5×10^{21}	0.9×10^{21}	0.9×10^{21}
Average Power Density in Core (W/cm ³)	1139	788	773
Peak Power Density in Core - rod out at BOC	2497	1849	1835
Peak Temperature in Fuel Meat (°C) BOC/EOC	150/180	130/160	130/160

(a) EOC excess reactivity = 5% $\Delta k/k$ for both the HEU and LEU designs; (b) Based on 250 days operation per year;

(c) In 3.0 g/cm³ fuel of the HEU design.

SUMMARY OF ANALYSES FROM REF. 1

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 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 für 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.

ADDITIONAL ANALYSES

This paper 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). Calculations were also done to address the possibility that new high density fuels could be developed that would allow conversion of the TUM HEU design to LEU fuel.

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 20-22 m/s. Experiments and analyses performed at ORNL determined that the fuel plates in this design would be stable during operation^[5-9]. 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. The following analysis supports this conclusion.

Hydraulic Stability Analysis

To analyze the alternative LEU design for the FRM-II, a computer code was obtained from ORNL. This code (see Ref. 10, 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.

In addition, the Alternative LEU designs have peak fuel temperatures of 130/160°C at BOC/EOC, as compared with 150/180°C for the HEU design. The design coolant velocity in the LEU core can be reduced to 12 m/s from 18 m/s and have the same peak fuel temperatures of 150/180°C as the HEU design. This design velocity of 12 m/s would then be smaller than the calculated critical velocity by a factor of 5.6.

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 (over 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 , Reference 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	20 - 22	46.8
ANS Upper Fuel Element, HEU Design	1.27	1.27	7.03	20 - 22	64.6

* Calculations were performed using a code developed by ORNL. See ORNL-11580, Ref. 6.

** The coolant velocity in the Alternative LEU design could be reduced to 12 m/s and have the same peak fuel temperatures as the HEU design.

It is known that the methodology used for the analyses described above is based on older (1991) ANS project analysis techniques. A more sophisticated approach^[11] adopted later by ORNL for ANS plate design also involves analyses of the thermal stress loads, conditions for attachment of the fuel plates to the side plates, and dynamic pressures. These more sophisticated analyses and experimental work^[5-9] concluded that the design of the ANS lower fuel element is indeed hydraulically stable.

It is clear that if the alternative LEU design is adopted, detailed analyses and tests similar to those performed for the ANS in Refs. 5-9 would need to be done and a prototype core would need to be flow tested. However, based on the very positive results that have already been obtained for the ANS design, we believe that the Alternative LEU Design for the FRM-II has a large safety margin with respect to hydraulic stability.

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. 12. ANL has analyzed this transient for both the HEU and alternative LEU designs using essentially the same assumptions as in Ref. 12 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. 12, "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. 13),
- (4) The initial light water pool temperature is 37°C (Ref. 13),
- (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. 12).
- (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. 12, 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. 12) for the HEU design and 4.4 for the alternative LEU design after 2.3 seconds. (The value of the bubble detachment parameter η was taken to be 47.8 in calculating the margin against flow instability, as suggested in Ref. 14 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. 12). 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. 2 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.

LEU Conversion of HEU 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 HEU geometry.

To illustrate this point, calculations were done in which LEU uranium metal with a density of 19 g/cm³, 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/cm²-s in the heavy water reflector. This performance level would not be acceptable.

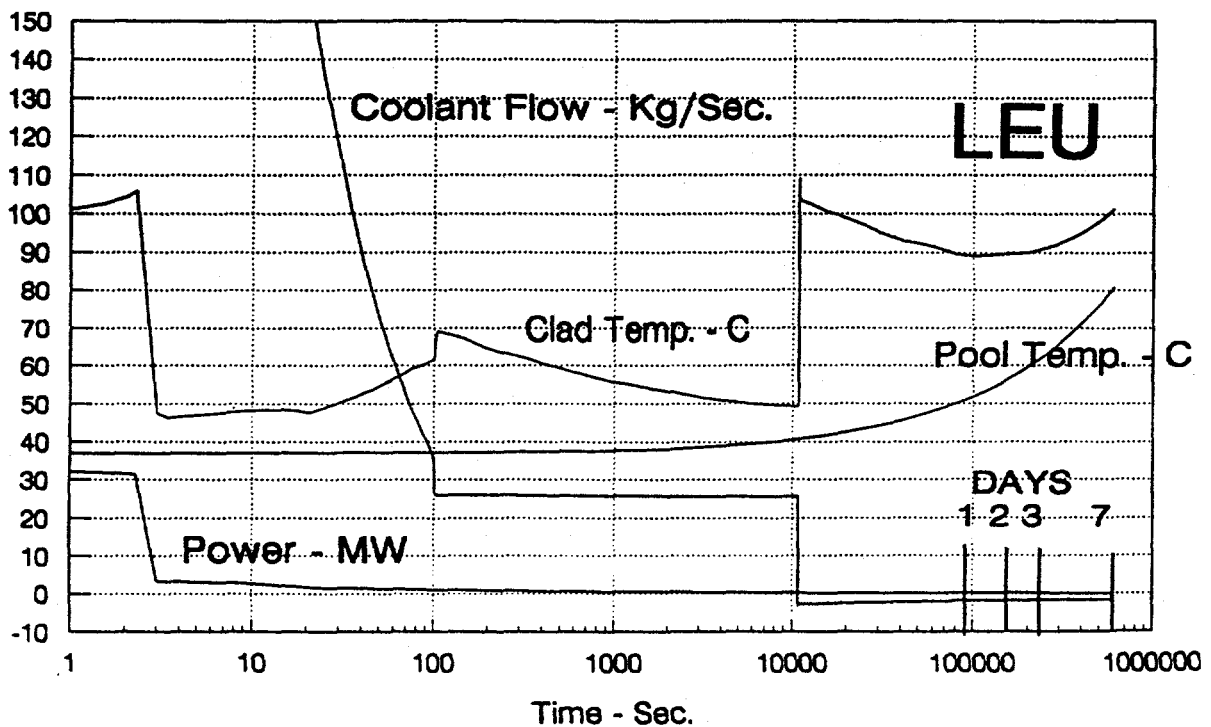
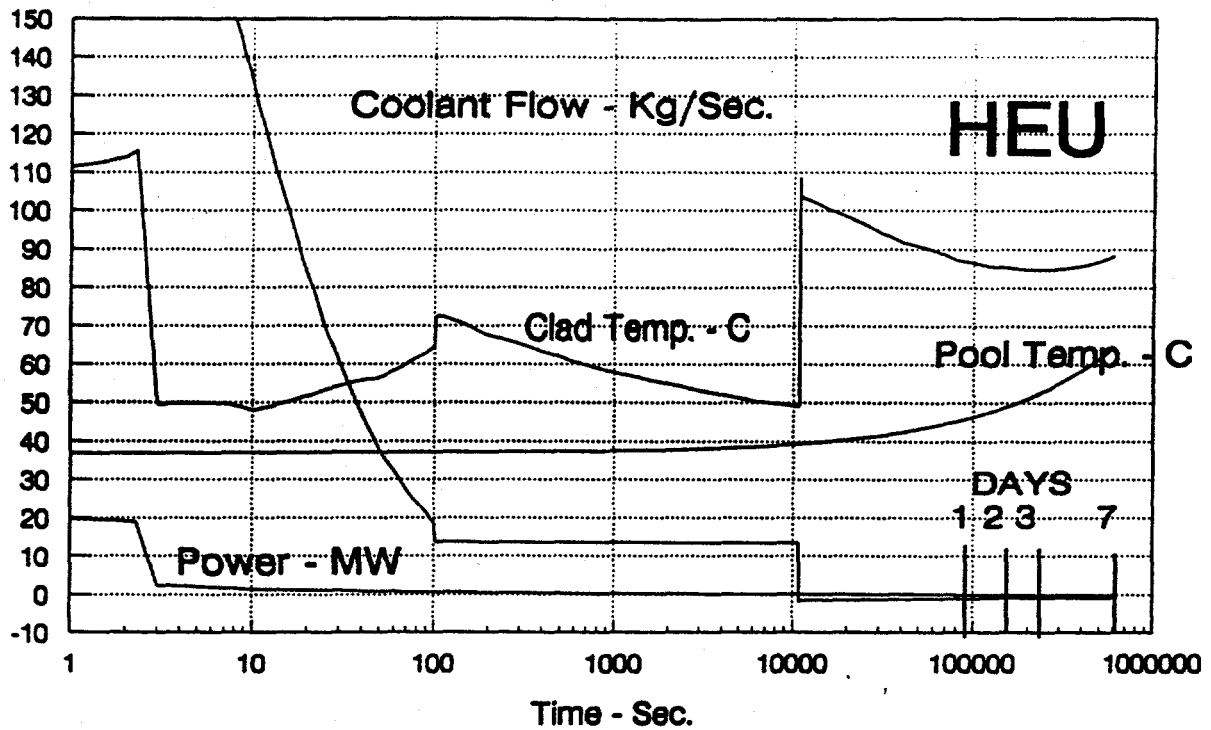
CONCLUSION

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.

A technical memorandum on the ANL design and safety studies for the FRM-II can be found in Ref. 15.

Figure 2. 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 XVIII International Meeting on Reduced Enrichment for Research and Test Reactors, 17-21 September 1995, Paris, France.
2. K. Böning, "Viewgraphs presented at the RERTR International Meeting in Paris, France, September 19, 1995", Attachment to letter from K. Böning, Technical University of Munich, to J.E. Matos, Argonne National Laboratory, October 6, 1995, and 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 XVIII International Meeting on Reduced Enrichment for Research and Test Reactors, 11-21 September 1995, Paris, France.
3. "U.S.-German Bilateral Technical Talks on FRM-II in Tegernsee on Jan. 18, 1996. ANL Proposal of an Alternative LEU Design", Letter from Klaus Böning, Technical University of Munich to J. E. Matos, Argonne National Laboratory, January 30, 1996.
4. "Resolution of TUM-ANL Discrepancy in Lifetime of Alternative LEU Design", Letter from J.E. Matos, Argonne National Laboratory to Klaus Böning, Technical University of Munich, February 23, 1996.
5. 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.
6. W. F. Swinson, R.L. Battiste, C.R. Luttrell, and G.T. Yahr, "Follow-up Fuel Plate Stability Experiments and Analyses for the Advanced Neutron Source", ORNL/TM-12629, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, November 1993.
7. W. F. Swinson, C.R. Luttrell, and G.T. Yahr, "An Examination of the Elastic Structural Response of the Advanced Neutron Source Fuel Plates", ORNL/TM-12712, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, September 1994.
8. W. F. Swinson, R.L. Battiste, and G.T. Yahr, "Circular Arc Fuel Plate Stability Experiments and Analyses for the Advanced Neutron Source", ORNL/TM-12977, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, August 1995.
9. W. F. Swinson, R.L. Battiste, and G.T. Yahr, "Structural Thermal Tests on Advanced Neutron Source Reactor Fuel Plates", ORNL/TM-13062, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory, August 1995.
10. 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.
11. "Comments on a paper presented by ANL at the U.S.-German Bilateral Technical Talks on FRM-II in Tegernsee, Germany on January 18, 1996", Letter from C.D. West, Oak Ridge National Laboratory to A. Travelli, Argonne National Laboratory, February 8, 1996.

12. 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.
13. "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.
14. "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.
15. 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, September 1996. This document is available at the World Wide Web address: <http://www.td.anl.gov/RERTR/RERTR.html> under Technical Memoranda.