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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 (Table 1). 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 x 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. Computer models for the HEU and LEU designs have been exchanged between TUM and ANL and disprepancies have been resolved.

The following issues are addressed: qualification of HEU and LEU silicide fuels, stability of the involute fuel plates, gamma heating in the heavy water reflector, a hypothetical accident involving the configuration of the reflector, a loss of primary coolant flow transient due to an interrupted power supply, the radiological consequences of larger fission product and plutonium inventories in the LEU core, and cost and schedule. Calculations were also done to address the possibility that new high density LEU 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 (Ref. 1), 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.

SUMMARY OF ANALYSES

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 that is proposed by TUM for the HEU design 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 in the FRM-II.

Fuel Element Hydraulic Stability

The lower core of the Advanced Neutron Source (ANS) reactor designed by Oak Ridge National Laboratory (ORNL) had involute plates that were 1.27 mm thick and had a width of 8.735 cm. The 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. The alternative LEU design for the FRM-II 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 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. Analyses supporting this conclusion can be found in Ref. 1.

If the alternative LEU design is adopted, detailed analyses and tests similar to those performed for the ANS 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 from experiments and analyses 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.

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.

Hypothetical Accident Involving the Moderator Material of the Reflector

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. Thus, both cores satisfy this safety criteria.

Loss of Primary Coolant Flow Transient

A loss of primary flow transient analysis described by TUM for the FRM-II HEU design was analyzed for both the HEU and alternative LEU designs using essentially the same assumptions as TUM. The results show that fuel integrity is maintained with a considerable safety margin in both cases. 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.

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. Detailed analyses show 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 also show 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.

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 is thought to be important.

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 use in 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.

CONCLUSION

Based on the excellent results for the Alternative LEU Design that were obtained in these analyses (Ref. 1), 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.

Ref. 1. 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. This technical memorandum on the ANL design and safety studies for the FRM-II can be found at the World Wide Web address: http://www.td.anl.gov/RERTR/RERTR.html

Table 1: Key Parameters of the FRM-II HEU Design and the Alternative LEU Design.

	FRM-II HEU Design	FRM-II Alternative LEU Design (a)
Enrichment, %	93.0	19.75
Reactor Power (MW)	20	32
Cycle Length (Full Power Days) (b)	50	50
Average Number of Cores/Year (c)	5.0	5.0
Peak Thermal Flux, $k_{eff}\bullet\phi th,max$ ($n/cm^2/s$)	8.0×10^{14}	8.0×10^{14}
Reflector Volume (liters) with $k_{eff}\bullet\phi th>7\times10^{14}$ $n/cm^2/s$	75	110
Active Core Inner - Outer Radius (cm)	6.75 - 11.2	10.45 - 16.55
Active Core Height (cm)	70	80
Active Core Volume (liters)	17.6	41.4
Number of Fuel Plates	113	172
Core Loading (Kg U-235)	7.5	7.5
Fuel Type	U_3Si_2	U_3Si_2
Fuel Grading	Yes	No
Fuel Meat Uranium Density (g/cm^3)	3.0/1.5	4.5
Fuel Meat/Clad Thickness (mm)	0.60/0.38	0.76/0.38
Inner/Outer Side Plate Thickness (cm)	0.6/0.7	0.6/0.7
Coolant Channel Thickness (mm)	2.2	2:2
Design Coolant Velocity, m/s	18.0	18.0
Width of Involute Plate (cm)	6.83	8.735
K_{eff} at BOC	1.1714	1.2101
Core Average Burnup (% U-235 burned)	17.3	25.9
Average Fission Rate in Fuel Meat ($fissions/cm^3/s$)	2.1×10^{14}	1.2×10^{14}
Peak Pointwise Fission Rate in Fuel Meat at BOC (d)	4.6×10^{14}	2.8×10^{14}
Average Fission Density in Fuel Meat ($fissions/cm^3$)	1.0×10^{21}	0.5×10^{21}
Peak Fission Density in Fuel Meat at EOC (d)	1.5×10^{21}	0.9×10^{21}
Average Power Density in Core (W/cm^3)	1139	773
Peak Power Density in Core - rod out at BOC	2497	1835
Peak Temperature in Fuel Meat (°C) BOC/EOC	150/180	130/160

(a) With involute plate width of 8.735 cm, as in lower core of ORNL's Advanced Neutron Source design, (b) EOC excess reactivity = 5% $\Delta k/k$ for both the HEU and LEU designs; (c) Based on 250 days operation per year; (d) In 3.0 g/cm^3 fuel of the HEU design.