TRANSIENT ANALYSES FOR HEU AND LEU DESIGNS OF THE FRM-II*

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

Argonne National Laboratory
Argonne, Illinois 60439-4841 USA

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ABSTRACT

An 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 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.

Several issues raised by TUM have been addressed in Refs. 1-4. The conclusions of these analyses are summarized below. In this paper, two typical design basis transients[5] are analyzed: control rod withdrawal at different power levels and loss of primary flow. The results show that the HEU and the LEU cores behave in a similar manner and both have excellent safety margins.

Based on the excellent results for the Alternative LEU Design that were obtained in all analyses, the RERTR Program reiterates its conclusion that there are no major technical issues regarding use of LEU fuel instead of HEU fuel in the FRM-II and that it is definitely feasible to use LEU fuel in the FRM-II without compromising the safety or performance of the facility.

INTRODUCTION

Key parameters of the TUM HEU and the ANL Alternative LEU designs are summarized in Table 1. Several issues that were raised by TUM have been addressed in Refs. 1-4, and a summary of the principal results of those analyses is provided below.

Results of transient analyses performed for both the HEU and the LEU designs are presented here. First, the methods used and the results obtained for the kinetics parameters, reactivity feedback coefficients, and differential reactivity worths of the safety rods are discussed. The results of reactivity ramp insertions (control rod withdrawal) and loss of flow transients are then presented.

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

	FRM-II HEU Design	FRM-II Alternative LEU Design
Enrichment, %	93.0	19.75
Reactor Power (MW)	20	32
Cycle Length (Full Power Days) (b)	50	50
Peak Thermal Flux, $k_{eff} \bullet \Phi_{th}$,max ($n/cm^2/s$)	8×10^{14}	8×10^{14}
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 Meat Uranium Density (g/cm^3)	3.0/1.5	4.5
Fuel Meat/Clad /Coolant Thickness (mm)	0.60/0.38/2.2	0.76/0.38/2.2
Design Coolant Velocity, m/s	18.0	18.0
Width of Involute Plate (cm)	6.83	8.735

SUMMARY OF ANALYSES FROM REFERENCES 1, 2,3, AND 4

(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 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.

(2) Fuel Element Hydraulic Stability

The lower core of the Advanced Neutron Source (ANS) reactor designed by Oak Ridge National Laboratory had involute plates that were 1.27 mm thick and had a width of 8.735 cm. 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 Refs. 2 and 3.

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.

Based on the excellent results for the Alternative LEU Design that were obtained in all analyses, the RERTR Program reiterates its conclusion that there are no major technical issues regarding use of LEU fuel instead of HEU fuel in the FRM-II and that it is definitely feasible to use LEU fuel in the FRM-II without compromising the safety or performance of the facility.

(3) 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.

(4) Hypothetical Accident Involving the Moderator Material of the Reflector

Monte Carlo calculations were performed for the 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.

(5) 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.

(6) 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 x 10^{14} n/cm^2-s in the heavy water reflector. This performance level would not be acceptable.

(7) Heat Generation in the Cold Neutron Source (CNS)

Results of detailed calculations performed for CNS heating show that neutron and gamma heating in the LEU design is smaller than in the HEU design. Also, calculated neutron spectra at the tip of each of the beam tubes show that the thermal neutron fluxes are about the same or larger in the LEU design when compared with the HEU design.

(8) Gamma and Fast Neutron Fluxes in the Experiment Hall

Analyses of the two components of the reactor noise - fast neutron and gamma fluxes - that would be generated in neutron scattering experiments to be performed in the experiment hall, were performed for both the HEU and the LEU designs. The results of these analyses show that there are only very small differences between the gamma and fast neutron fluxes inside the beam tubes of the HEU and LEU designs.

(9) Fuel Cycle Length

A discrepancy in the fuel cycle length calculations for the LEU design performed by ANL and TUM was resolved by showing that the analysis performed by TUM (computer input provided by TUM to ANL) inadvertently used a uranium enrichment of 19.28% instead of 19.75% in the LEU design.

(10) Reactivity Worth of Cold Source, Beam Tubes and Other Experiment Facilities

Results of the reactivity worth of the beam tubes and other experiment facilities showed that the absolute value of the reactivity worth of these facilities is between 0.6% and 0.9% smaller in the LEU design. This difference in reactivity worth can be used to increase the fuel cycle length of the LEU design from 50 days to 53 or 54 days.

ADDITIONAL ANALYSES

This paper addresses the analysis of typical design basis accidents as defined by TUM in Ref. 5. These accidents include the uncontrolled withdrawal of the control rod and the loss of flow. First, the kinetics parameters (neutron generation time, delayed neutron fractions and decay constants), and the reactivity feedback coefficient for both designs were determined by using Monte Carlo (MCNP4B[6]) and diffusion/perturbation theory (VARI3D[7]) codes. Then, coupled neutronic thermal-hydraulics models were generated for RELAP5/MOD.3.2[8] and PARET[9] codes to perform the transient/accident analyses.

Kinetics Parameters

The kinetics parameters for the HEU and the LEU designs are provided in Table 2. These parameters, the effective delayed neutron fraction and the neutron generation time, were obtained using the perturbation theory code VARI3D. The neutron generation times were also calculated using the 1/v absorption method and the MCNP code; all three methods yielded very similar results (less than 10% difference). The data in Table 2 show that the kinetics parameters for both designs are very similar.

Table 2: Kinetics Parameters for the FRM-II HEU and Alternative LEU Designs

PARAMETER	HEU	LEU
Delayed Neutron Fraction, β_{eff}	7.49E-03	7.29E-03
Neutron Generation Time (s)	5.09E-04	4.39E-04

Reactivity Feedback Coefficients

Two different methods, diffusion theory and Monte Carlo, were used to determine the following reactivity feedback coefficients: coolant void, coolant temperature, and Doppler.

The first method used the diffusion theory code DIF3D[10]. First, sets of seven-group cross sections at different conditions (several temperatures and coolant void fractions) were generated using the WIMS-ANL code. These cross sections were then used in DIF3D to calculate the feedback coefficients. The second method used the MCNP4B Monte Carlo code. First, cross sections at different temperatures were generated using the NJOY[11] code, and then MCNP4B was used to calculate k_{eff} 's for the different conditions (temperatures and coolant void fractions). The results of the analyses (Table 3) show that the agreement between the two methods is very good.

Table 3: Reactivity Feedback Coefficients for the FRM-II and Alternative LEU Designs

	HEU		LEU	
	DIF3D	MCNP4B	DIFI3D	MCNP4B
Coolant Void ($ / % of void)	-1.90E-01	-2.03E-01 (± 1.80E-03)	-1.55E-01	-1.67E-01 (±1.67E-03)
Coolant Temperature ($ / °K)	-4.40E-03	-4.50E-03 (± 1.50E-04)	-7.05E-03	-7.22E-03 (± 1.6E-04)
Doppler ($ / °K)	-3.54E-04	-3.37E-04 (± 1.66E-05)	-1.80E-03	-2.03E-03 (± 3.02E-05)

Shutdown Systems

Both the HEU and LEU designs have two shutdown systems. The primary shutdown system in the HEU design "consists of five individual hafnium safety rods which are fully withdrawn during normal reactor operation but can be quickly inserted into the D_2O tank by means of pneumatic or spring forces and gravity; four of these rods are sufficient to scram the reactor"[5]. In the LEU design, the primary shutdown system is very similar to the HEU design but it consists of eight hafnium safety rods and seven of them are sufficient to scram the reactor. The secondary shutdown system in both designs is the central control rod.

For the accident analyses presented in this paper, only the primary shutdown system is used for scram. The reactivity worth of this system was calculated to be about 23 dollars for the HEU design and about 29 dollars for the LEU design. In both designs a very large shutdown margin (about 8 dollars) is available. The differential worth of these safety rods was also determined and the traditional S-curve was obtained for use in the transient analyses.

Typical Design Basis Accidents

The codes RELAP5-MOD3.2 and PARET were used to analyze two typical design basis accidents: the uncontrolled withdrawal of the central control rod, and the loss of primary flow. For all analyzed transients, the results of both codes are basically the same.

Uncontrolled Withdrawal of the Central Control Rod

Information on the reactivity rate introduced by the uncontrolled withdrawal of the central control rod for the HEU core is not provided in Reference 5. Therefore, several reactivity insertion rates were tried in order to match the power trace results presented in that paper. A rate equal to 6.6 ¢/s was found to reproduce the results in Reference 5, and is used in this paper. The following assumptions made in these analyses are the same as those made in Reference 5: a) the period trip is assumed to fail and the reactor scram occurs at 115% of operating power, and b) a time delay of 0.5 seconds is allowed for signal processing and for the release of the safety shutdown rods.

Figures 1 and 2 present the results of this accident for the HEU design. Figure 1 shows the power traces for different initial powers, and Figure 2 shows the hot channel coolant outlet temperature for initial powers of 1 W and 2 MW. The same results are shown in Figures 3,and 4 for the LEU design. It is seen from these results that the behavior of both designs is very similar, and that the initial power condition of 1 W (start-up accident) is the most demanding case. For the start-up accident the HEU design reaches a peak power equal to about 2.5 times the normal operating power and a peak coolant outlet temperature equal to 122 °C. The peak power for the LEU design is 2.1 times the normal operating power and the peak coolant outlet temperature is equal to 110 °C . In both designs the fuel integrity is maintained and the safety margin S against flow instability is much greater than the minimum required value of 1.5[13]. Figure 5 shows these results for both designs and it can be seen that the minimum value reached is about 3 for the HEU design and about 6 for the LEU design. Note that in Reference 5, it is stated that "the most pessimistic value of the reactor power at the beginning of the accident was found to be 2 MW," and this is not the case as seen in Figures 1 through 4.

Another case, using the same reactivity insertion rate as that in the RHF reactor[14], i.e. 10 ¢/s, was also analyzed and the results for both the HEU and LEU designs are presented in Figures 6 through 8. As seen in Figure 8, the safety margin against flow instability is still maintained in the LEU design. Note that this reactivity insertion rate is most probably not allowed by design in the HEU core.

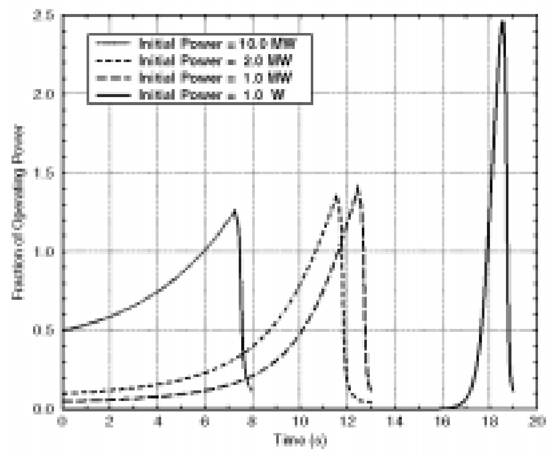

Figure 1. HEU Control Rod Withdrawal (6.6 ¢/s): Power History for Several Initial Power Levels

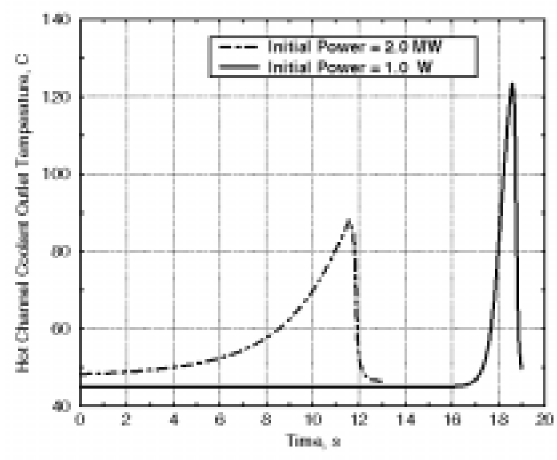

Figure 2. HEU Control Rod Withdrawal (6.6 ¢/s): Hot Channel Coolant Outlet Temperature

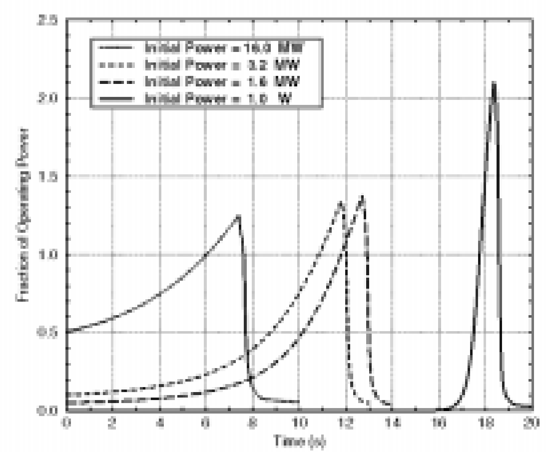

Figure 3. LEU Control Rod Withdrawal (6.6 ¢/s): Power Vs Time for Several Initial Power Levels

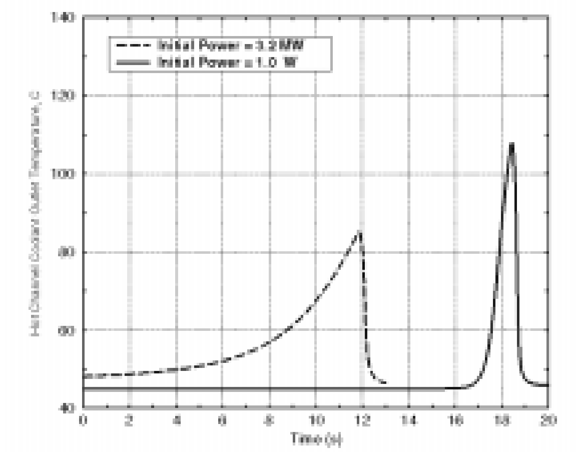

Figure 4. LEU Control Rod Withdrawal (6.6 ¢/s): Hot Channel Coolant Outlet Temperature

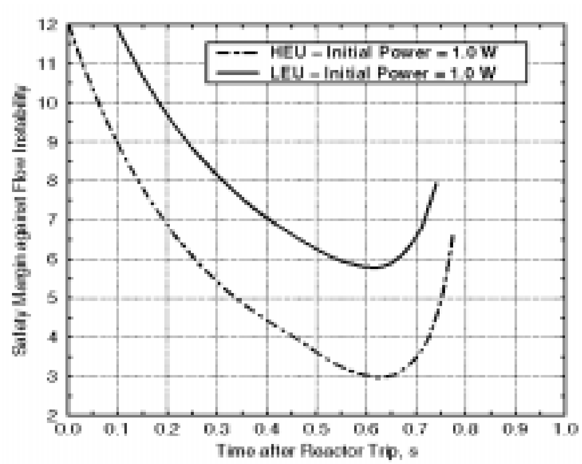

Figure 5. Control Rod Withdrawal (6.6 ¢/s: Safety Margin against Flow Instability

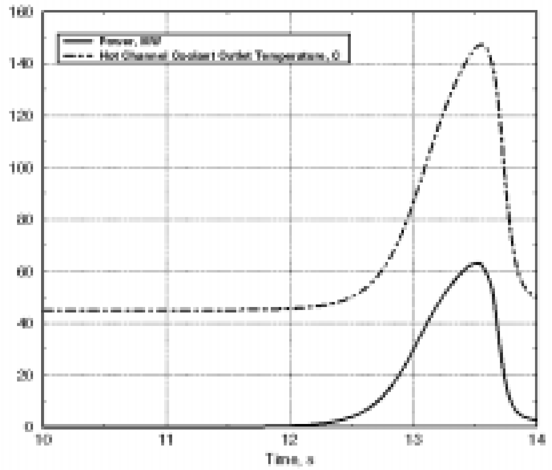

Figure 6 . HEU Control Rod Withdrawal (10 ¢/s): Power and Hot Channel Coolant Temperature

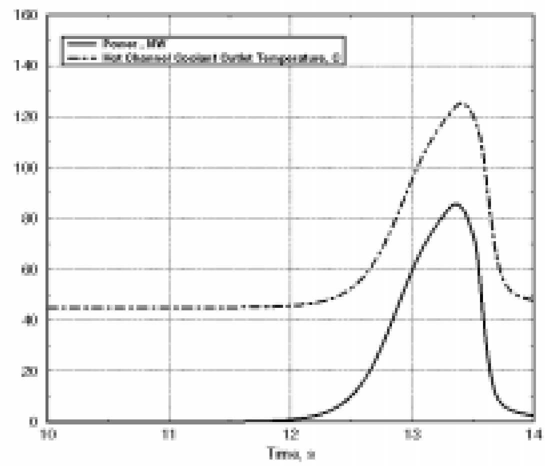

Figure 7 . LEU Control Rod Withdrawal (10 ¢/s): Power and Hot Channel Coolant Temperature

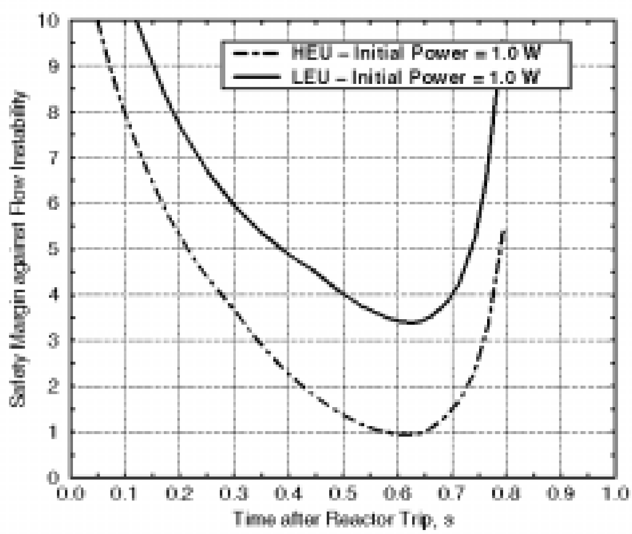

Figure 8. Control Rod Withdrawal (10 ¢/s): Safety Margin against Flow Instability

Loss of Primary Flow

This transient assumes simultaneous loss of all four primary pumps and is due to a loss of offsite power. The pumps are equipped with flywheels and by design, "simultaneously with the reactor scram battery-supplied emergency core cooling pumps are started to maintain forced flow for at least three hours"[5]. Only the initial part of the transient is analyzed here; the long term transient was presented in Reference 3.

The results are presented in Figures 9 and 10 for the HEU design, and in Figures 11 and 12 for the LEU design. These figures show that both designs have a high safety margin for this transient. The peak fuel temperatures reach about 140 °C and the peak coolant outlet temperature is about 85 °C for both designs. Note that the 0.5 seconds time delay for safety rods release (see assumption b) above) is used, and the scram signal in this case occurs when the flow rate reaches 85% of its design value.

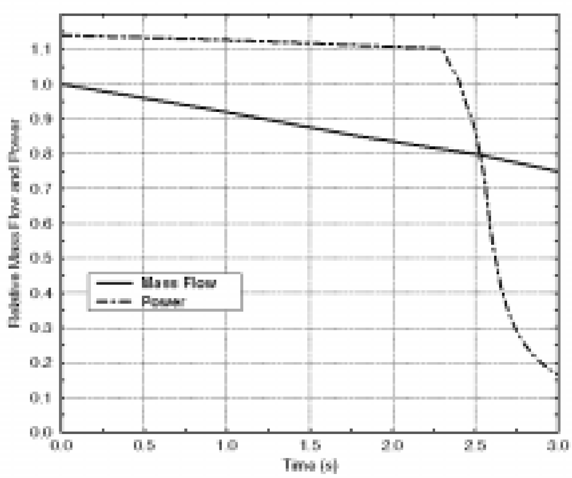

Figure 9. HEU Loss of Flow: Power and Mass Flow Rate

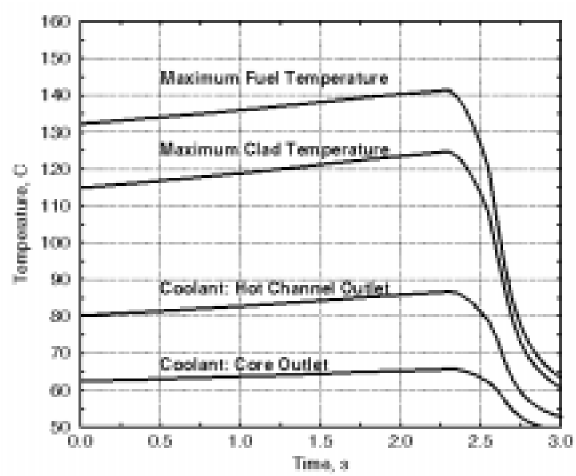

Figure 10. HEU Loss of Flow: Fuel, Clad, and Coolant Temperature

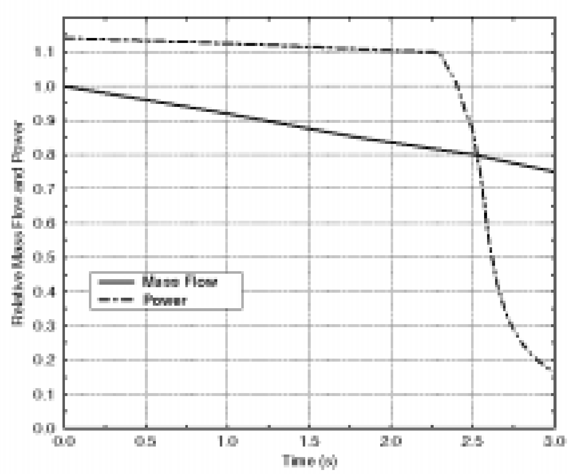

Figure 11. LEU Loss of Flow: Power and Mass Flow Rate

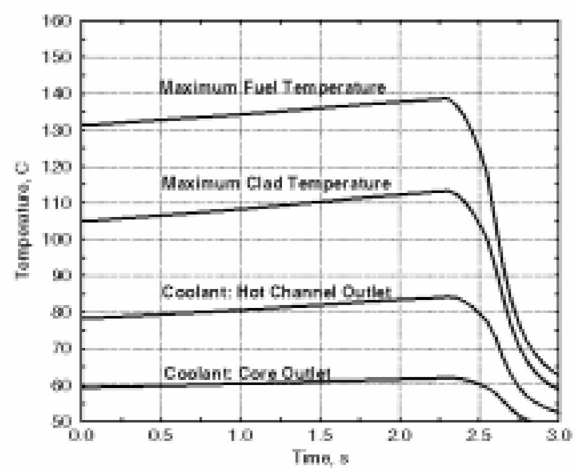

Figure 12. LEU Loss of Flow: Fuel, Clad, and Coolant Temperatures

CONCLUSIONS

The kinetics parameters, reactivity feedback coefficients, and the reactivity worth of the safety rods for both the HEU and the LEU designs were determined in order to perform two design basis accident analyses: the uncontrolled withdrawal of the central control rod and the loss of primary flow. The results show that both the HEU and the LEU designs have excellent safety margins.

Based on the excellent results for the Alternative LEU Design that were obtained in all analyses, the RERTR Program reiterates its conclusion that there are no major technical issues regarding use of LEU fuel instead of HEU fuel in the FRM-II and that it is definitely feasible to use LEU fuel in the FRM-II without compromising the safety or performance of the facility.

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