NEUTRONIC SAFETY PARAMETERS AND TRANSIENT ANALYSES FOR POLAND'S MARIA RESEARCH REACTOR*

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Presented at the
1999 International Meeting on Reduced Enrichment
for Research and Test Reactors

October 3-8, 1999

Budapest, Hungary

*Work supported by the U.S. Department of Energy
Office of Nonproliferation and National Security
under Contract No. W-31-109-ENG-38

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ABSTRACT

Reactor kinetic parameters, reactivity feedback coefficients, and control rod reactivity worths have been calculated for the MARIA Research Reactor (Swierk, Poland) for M6-type fuel assemblies with ^{235}U enrichments of 80% and 19.7%. Kinetic parameters were evaluated for family-dependent effective delayed neutron fractions, decay constants, and prompt neutron lifetimes and neutron generation times. Reactivity feedback coefficients were determined for fuel Doppler coefficients, coolant (H_2O) void and temperature coefficients, and for in-core and ex-core beryllium temperature coefficients. Total and differential control rod worths and safety rod worths were calculated for each fuel type. These parameters were used to calculate generic transients for fast and slow reactivity insertions with both HEU and LEU fuels. The analyses show that the HEU and LEU cores have very similar responses to these transients.

INTRODUCTION

The MARIA Research Reactor currently uses HEU (80% ^{235}U) 6-tube fuel assemblies cooled with water and located on a square grid within a beryllium matrix. Although the supply of HEU fuel is nearly exhausted, there is an on-hand inventory of 49 fresh 36%-enriched fuel assemblies fabricated in Russia. Equilibrium fuel cycle analyses were used to determine UO_2-Al LEU (19.7% ^{235}U) fuel requirements needed to match the performance of the 80% - enriched reference fuel and the anticipated performance of the highly-loaded 36% - enriched fuel. Results from these studies (Ref. 1) were based on generic cores consisting of 16 fuel assemblies (FA) with HEU(80%) and LEU(19.7%) fuel and 14 fuel assemblies with HEU(36%) and LEU(19.7%) fuel. Most of the neutronic safety parameters and transient analyses reported here were done for the 16 fuel assembly core with performance-matching 80%-enriched and 19.7%-enriched fuel assemblies.

THE MARIA RESEARCH REACTOR

Poland's MARIA Research Reactor is a high-flux multipurpose reactor which is water-cooled and moderated with both water and beryllium. Standard U-Al alloy HEU (80%) fuel assemblies are of the M6-type (Fig. 1) which consist of six concentric circular fuel tubes clad in aluminum and cooled with water. The fuel assemblies are located within a beryllium matrix on a square grid with a pitch of 13.0 cm on the core midplane. Reactor power depends on the core configuration, but is typically of the order of 20 MW. The MARIA reactor, fuel assemblies, and operational characteristics are described in Ref.'s 2 and 3. Some reactor parameters for the 16 fuel assembly reference core configuration are given in Table 1. The ^{235}U loading for the LEU-1 (402g ^{235}U) fuel assembly was chosen to match the performance of the HEU (350g ^{235}U) reference fuel. However, the loading for the LEU-2 (600g ^{235}U) fuel assembly was chosen to match the performance of the unused inventory of 36%-enriched MARIA fuel assemblies (550g ^{235}U). Figure 2 shows the 16 fuel assembly reference core configuration used in this study. For the MARIA M6 fuel assembly water flows downward in the three outer coolant channels and upward in the inner coolant channels. Thus, the third fuel tube is cooled on the outside by downward flow and on the inside by upward flow.

Table 1. MARIA Reactor Parameters
(16 Fuel Assembly Core)

Parameter	HEU	LEU-1	LEU-2
Fuel Type:	UAl-Alloy	UO_2-Al	UO_2-Al
Enrichment (wt % ^{235}U):	80.0	19.7	19.7
Uranium density (g U/cm^3):	1.28	2.53	3.78
Uranium dispersant volume fraction (%):	28.3	27.6	41.3
Meat/clad/coolant thickness (mm):	0.40/0.80/2.5	0.94/0.53/2.5	0.94/0.53/2.5
Height of fuel column (cm):	100	100	100
^{235}U mass per fuel assembly (g):	350	402	600
Reactor Power (MW):	17	17	17
Cycle Length (full power days)[a]:	7.5	8.8	13.0
Peak Thermal Neutron Flux (n/cm^2-sec)[b]	9.68E+13	9.41E+13	9.02E+13

[a] For ^{235}U average discharge burnup equal to 45%.

[b] $k_{eff} * \phi_{th}$ on core midplane of the h-8 water channel (see Fig. 2) for neutron energies < 0.625 eV.

METHODS AND CODES

Homogenized microscopic fuel cross sections were generated for each uranium loading and for different fuel meat temperatures and coolant temperatures and densities using the WIMS-ANL code and a 69-group ENDF/B-VI-based library[4]. Cross sections were also created for the non-fueled regions in the reactor including the beryllium matrix (with its poisons), graphite and water reflectors, in-core water holes, and control rods (30 wt % B_4C and 70 wt % Al), Al control rod followers, and control rod channels. They were collapsed into 7 broad groups with energy boundaries of 10.0 MeV, 0.821 MeV, 5.530 keV, 4.0 eV, 0.625 eV, 0.250 eV, 0.058 eV, and 1.0E-5 eV. These cross section sets were used in DIF3D[5] diffusion calculations and in REBUS[6] equilibrium cycle fuel depletion calculations. Group-dependent internal boundary conditions (i.e. neutron current-to-flux ratios) were calculated at the clad surface of the B_4C-Al control rods with the transport code, TWODANT[7]. These internal boundary conditions were used in diffusion calculations to determine total and differential control rod worths. Effective delayed neutron parameters were calculated with the MC^2 code[8] and VARI3D[9]. The code RELAP5[10] was used for the transient analyses.

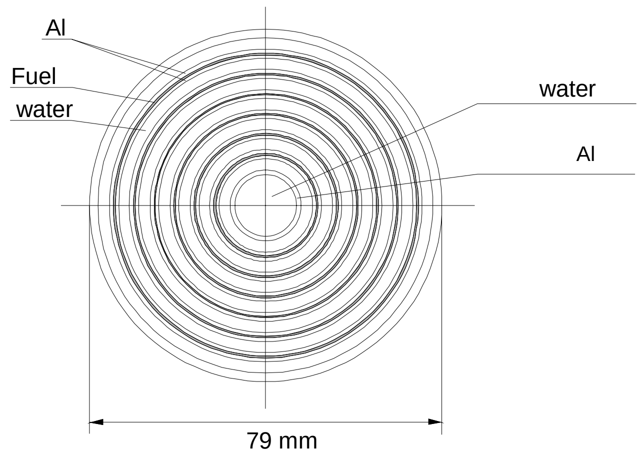
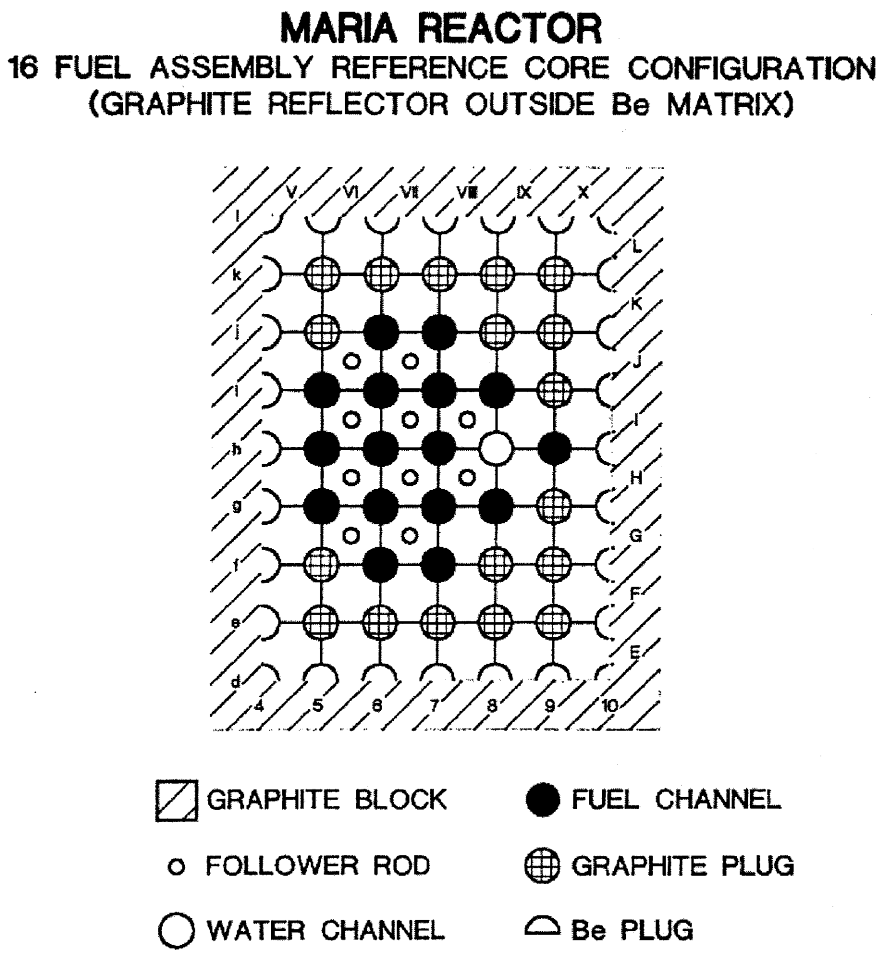


Figure 1. Horizontal Cross Section of the MARIA M6 Fuel Assembly



Location of Safety Rods: H-VI, J-VI, G-VII, I-VII, H-VIII
Location of Control Rods: G-VI, I-VI, J-VII, I-VIII, H-VII

Figure 2

KINETIC PARAMETERS

The 1/v insertion method, described in Ref. 11, was used to calculate prompt neutron lifetimes. According to this method, the prompt neutron lifetime is

$$l_p = \mathrm{Lim}_{(N\rightarrow 0)}\, l_p' \quad \text{where} \quad l_p' = (k_0/k_p - 1)/N\sigma_{a0}v_0$$

and where k_p is the eigenvalue for the reactor uniformly poisoned with a 1/v absorber of concentration N and neutron absorption cross section σ_{a0} at neutron speed v_0. The reference eigenvalue is k_0 for which N = 0. A good approximation for a 1/v absorber is ^{10}B for which σ_{a0} = 3839.5 ± 9 barns (ENDF/B-VI) at v_0 = 2200 m/sec. Using region-dependent ^{10}B broad group cross sections obtained from WIMS-ANL, three-dimensional diffusion calculations were used to determine k_0 and k_p eigenvalues.

Prompt neutron lifetimes were calculated for the MARIA reactor with 16 fuel assemblies (Fig. 2) of fresh 80%-enriched fuel, with 16 fuel assemblies of fresh 19.7%-enriched fuel, and with the control rods fully withdrawn. In addition, prompt neutron lifetimes were evaluated for the 16-fuel-assembly HEU equilibrium core described in Ref. 1. Table 2 summarizes the results. Since the prompt neutron lifetime varies inversely with neutron speed, its value increases as the neutron spectrum softens as Table 2 shows. The neutron generation time is equal to l_p / k_0.

Table 2. Prompt Neutron Lifetime Calculations for the MARIA Reactor With 16 Fuel Assemblies and Withdrawn Control Rods

Parameter	HEU				LEU-1	LEU-2
	(80.0%)				(19.7%)	(19.7%)
Grams ^{235}U / FA	350				402	600
	Fresh	BOEC	MOEC	EOEC [a]	Fresh	Fresh
k_0	1.2203	1.0789	1.0650	1.0564	1.2087	1.3685
l_p (micro-sec)	180.4	185.6	187.6	189.2	168.2	124.5
Neutron Gen. Time (μs)	147.8	172.0	176.2	179.1	139.2	90.9

[a] The average ^{235}U discharge burnup is 45% for a cycle length of 7.50 days at 17 MW.

Effective delayed fission neutron fractions were calculated from flux and adjoint distributions using the VARI3D[9] perturbation code and ENDF/B-VI delayed neutron data provided by the MC^2 code[8]. Perturbation-independent methods for calculating family-dependent $\beta_{eff,\,i}$ values are discussed in Ref. 11 which shows that these methods give results that are within less than 1% of the corresponding VARI3D values.

Table 3 summarizes the VARI3D delayed fission neutron data for fresh HEU and LEU-2 (600g ^{235}U/FA) fuel. Note that the delayed neutron fractions for the HEU and LEU-2 fuels are nearly the same as are the family-dependent λ_i decay constants. Therefore, no delayed neutron fractions were calculated for the LEU-1 (402g ^{235}U/FA) fuel. Additional calculations showed that for the HEU equilibrium core delayed neutron fractions at the end of the equilibrium cycle (EOEC) were only about 0.4% smaller than the corresponding values for fresh fuel. This shows that the delayed neutron fractions are quite insensitive to neutron spectrum changes.

Table 3. Effective Delayed Neutron Fractions and Decay Constants for the MARIA Reactor with 16 Fresh Fuel Assemblies and Withdrawn Control Rods (Based on ENDF/B-VI Data)

Delayed Fission Neutrons (DFN)				
	HEU Fuel: 350g ^{235}U/FA		LEU-2 Fuel: 600g ^{235}U/FA	
Family, i	$^{DFN}\beta_{eff,i}$	λ_i (sec^{-1})	$^{DFN}\beta_{eff,i}$	λ_i (sec^{-1})
1	2.4764E-04	1.3336E-02	2.4622E-04	1.3337E-02
2	1.2758E-03	3.2739E-02	1.2668E-03	3.2737E-02
3	1.2223E-03	1.2078E-01	1.2181E-03	1.2079E-01
4	2.7244E-03	3.0279E-01	2.7214E-03	3.0292E-01
5	1.1191E-03	8.4951E-01	1.1221E-03	8.5009E-01
6	4.6875E-04	2.8531E+00	4.6989E-04	2.8550E+00
Total = $^{DFN}\beta_{eff}$	7.0580E-03		7.0444E-03	
$^{DPN}\beta_{eff}$ [a]	1.366E-04		1.366E-04	
β_{eff} = sum	7.1946E-03		7.1810E-03	

[a] The delayed photo-neutron (DPN) data are taken from G. Robert Keepin, Physics of Nuclear Kinetics, Addison-Wesley (1965). It is assumed here that the delayed photo-neutron precursor concentrations are saturated and that the effectiveness of the delayed photo-neutrons is 0.90.

Delayed photo-neutrons (DPN) come from the interaction of fission product gamma rays on beryllium. This ^{9}Be (γ,n) reaction has a threshold energy of about 1.67 MeV. The effectiveness of these delayed photo-neutrons depends on their energy spectrum, which is softer than that of delayed fission neutrons (DFN), and on the energy degradation, absorption and leakage of fission product gamma rays with energies above 1.67 MeV. Keepin's data[12] show that these delayed photo-neutrons, even without any precursor gamma ray losses, contribute less than 2% to the total value of β_{eff} . Therefore, a detailed calculation of $^{DPN}\beta_{eff}$ in the MARIA reactor was not made. Rather, the effectiveness of the ^{9}Be(γ,n) photo-neutrons was assumed to be 0.90. The total delayed photo-neutron fraction shown at the bottom of Table 3 is based on this assumption and on Keepin's data.

REACTIVITY FEEDBACK COEFFICIENTS

The reactivity feedback coefficient, α_x, is defined by the equation

$$\alpha_x \equiv d\rho(x)/dx = d[k_0^{-1} - k(x)^{-1}]/dx = k(x)^{-2}dk(x)/dx$$

where k_0 is the eigenvalue of the reactor for the reference configuration and where k(x) is the eigenvalue for the x-modified state of the reactor. For example, x may correspond to a modified coolant, fuel, or reflector temperature or to a different coolant void fraction.

WIMS-ANL cross sections were generated for numerous “x” values. These multigroup cross sections were used in a series of three-dimensional diffusion calculations to determine a corresponding set of k(x) eigenvalues. For a particular x-type, such as the coolant void fraction, the k(x) values were fit to a polynomial of order m by the least squares process.

$$k(x) = a_1 + a_2x + a_3x^2 + \cdots + a_mx^{m-1} = \Sigma_{i=1,m}\, a_i\, x^{i-1}$$

$$dk(x)/dx = a_2 + 2a_3x + \cdots + (m-1)a_mx^{m-2} = \Sigma_{i=2,m}\, a_i\, (i - 1)\, x^{i-2}$$

With the fitting coefficients, a_i, determined from a least squares polynomial fit of k(x), these equations were used to calculate the reactivity feedback coefficients α_x. The statistical uncertainty in α_x, σ_α, follows from the standard deviations of the fitting coefficients while the value of m was chosen so as to minimize σ_α.

Table 4 summarizes reactivity feedback coefficients calculated for the MARIA reactor 16 fuel assembly core (Fig. 2) with fresh HEU, LEU-1 and LEU-2 fuel. For these cases the control rods were fully withdrawn. This table shows that the coolant void coefficients are well-determined by a 4^{th} order polynomial fit of k(x). In most of the other cases, however, a linear fit was used because of large uncertainties in the fitting coefficients for higher order polynomials. Note that the coolant void coefficient, the coolant temperature coefficient, and the fuel Doppler coefficient are negative. For in-core and ex-core beryllium, however, the temperature coefficient is positive.

CONTROL ROD WORTHS

MARIA reactor absorber rods have two different compositions but the same dimensions. The “old” absorbers had a radius of 0.90 cm and were made of boral (30 wt % B_4C, 70 wt % Al, and a density of about 2.38 g/cm^3) while the “new” absorbers consist of a B_4C-Al dispersion (50 wt % B_4C, 50 wt % Al, and a density of 2.02 g/cm^3). The absorbers are clad in aluminum and have aluminum control rod followers below them. Rod worths were calculated with the DIF3D code[5] using group-dependent internal boundary conditions (current-to-flux ratios) applied at the clad surfaces of the absorber. These internal boundary conditions were obtained from P_1S_{16} TWODANT[7] transport calculations. Because both rod types have the same surface area and are “black” to low-energy neutrons, the control rod worths are nearly the same for fresh rods of either type. Table 5 gives the results of control rod worth calculations for the 16 fuel assembly core (Fig. 2) with fresh fuel. The same methods were used to obtain differential rod worth curves needed for some of the transient analysis.

For this 16 fuel assembly configuration the excess reactivity at the BOEC is about 4.5% $\delta k/k^2$ for the HEU fuel and about 4.9% $\delta k/k^2$ for the LEU-1 fuel. These excess reactivities include a DIF3D reactivity bias correction of 2.2% $\delta k/k^2$ relative to a detailed Monte Carlo calculation (see Ref. 1). Table 5 shows that the worth of the control rods is adequate, but additional control rods could be added at the edge of the core if needed. However, the LEU-2 type fuel is probably too reactive for this 16-fuel-assembly core. A core with 14 LEU-2 fuel assemblies was suggested in Ref. 1.

Table 4. Reactivity Feedback Coefficients ($\alpha_x = d\rho/dx = k(x)^{-2}\ dk(x)/dx$) for the MARIA Reactor with 16 Fresh Fuel Assemblies and Withdrawn Control Rods (Based on ENDF/B-VI Data)

Material	x Units	x Values	k(x) vs x Polyfit Order m	HEU: 350g ^{235}U/FA		LEU-1 402g ^{235}U/FA		LEU-2 600g ^{235}U/FA	
				α_x	σ_α - %	α_x	σ_α- %	α_x	σ_α - %
H_2O	Void, %	0.0	4	-7.381-5	0.662	-1.803-4	0.174	-4.212-4	0.023
Coolant		2.4596		-9.880-5	0.609	-2.030-4	0.190	-4.394-4	0.027
		3.8645		-1.134-4	0.654	-2.162-4	0.220	-4.501-4	0.032
		9.6848		-1.762-4	0.912	-2.727-4	0.381	-4.968-4	0.064
		19.7198		-2.942-4	1.381	-3.776-4	0.694	-5.866-4	0.138
H_2O	Temp, K	300	2	-1.083-4	0.676	-9.592-5	0.554	-3.365-5	1.627
Coolant		350		-1.097-4	0.676	-9.702-5	0.555	-3.379-5	1.627
		400		-1.112-4	0.676	-9.818-5	0.554	-3.395-5	1.627
		450		-1.128-4	0.677	-9.937-5	0.555	-3.411-5	1.626
		500		-1.142-4	0.676	-1.005-4	0.555	-3.427-5	1.627
Fuel	Temp, K	300	2 - HEU	-8.410-7	6.403	-1.068-5	5.728	-1.149-5	0.800
Meat		350	2 -LEU1	-8.412-7	6.403	-1.070-5	5.728	-1.101-5	0.885
		400	3 -LEU2	-8.412-7	6.403	-1.071-5	5.728	-1.053-5	0.982
		450		-8.413-7	6.403	-1.072-5	5.728	-1.005-5	1.092
		500		-8.414-7	6.403	-1.074-5	5.728	-9.565-6	1.218
		550		-8.415-7	6.403	-1.075-5	5.728	-9.079-6	1.361
In-Core	Temp, K	300	2	+6.361-5	3.969	+6.565-5	3.851	+4.395-5	6.304
Beryllium		350		+6.303-5	3.968	+6.504-5	3.850		
		400		+6.253-5	3.969	+6.451-5	3.850	+4.337-5	6.305
		450		+6.207-5	3.968	+6.403-5	3.850		
		500		+6.168-5	3.969	+6.361-5	3.851	+4.292-5	6.304
Ex-Core	Temp, K	300	2	+4.042-5	3.403	+4.298-5	3.361	+3.163-5	5.695
Beryllium		350		+4.019-5	3.402	+4.272-5	3.360		
		400		+3.999-5	3.403	+4.249-5	3.361	+3.133-5	5.695
		450		+3.980-5	3.402	+4.228-5	3.360		
		500		+3.963-5	3.403	+4.210-5	3.361	+1.109-5	5.605

Table 5. Control and Safety Rod Worths in the MARIA Reactor with 16 Fresh Fuel Assemblies

Boral Absorber Rods [a]	Rod Worth in Units of -% $\delta k/k^2$		
	HEU Fuel	LEU-1 Fuel	LEU-2 Fuel
	350g ^{235}U/FA	402g ^{235}U/FA	600g ^{235}U/FA
Control Rod Bank: (G-VI, I-VI, J-VII, I-VIII, H-VII)	8.05	7.93	6.90
Safety Rod Bank: (H-VI, J-VI, G-VII, H-VIII)	5.29	5.21	4.55
Maximum Worth Safety Rod: (I-VII)	2.09	2.05	1.79

[a] See Figure 2 for absorber rod locations.

REPRESENTATIVE TRANSIENTS

The RELAP5 code[10] was used to calculate the response to fast ($1.2/sec for 1.0 sec) and slow ($0.04/sec) reactivity insertions in the 16-fuel assembly MARIA reactor starting from a reactor power equal to 1.0 W. Table 6 shows some input parameters used in RELAP5 as well as important output results. Kinetic parameters and control rod worths are given in Tables 2,3 and 5. Coolant void, coolant temperature, and fuel Doppler feedback coefficients used in the transient analyses are the 300K values in Table 4. The model used in RELAP5 includes one fuel assembly (FA) to represent the aggregate of 15 average FA's and one FA to represent the peak power assembly. All the coolant channels and fuel tubes in the FA are explicitly represented (as parallel plates) as is the downward and upward coolant flow within the FA.

Table 6. Input Parameters and Output Results for the Transient Analyses

Input Parameter			Value	
Total Coolant Flow Rate (Core):			522 m^3/hr	
Inlet Coolant Temperature:			54 ^{o}C	
Inlet Pressure:			1.317 MPa (191 psi)	
Steady State Power:			17.0 MW	
Power Trip – 20% Above Steady State Power:			20.4 MW	
Peak-to-Average Power Density:			2.815 - HEU, 2.900 - LEU-1	
Delay and Drop Times of Shutdown Rods:			0.10 and 0.90 sec	
Fast Reactivity Insertion Rate:			$1.2/sec	
Slow Reactivity Insertion Rate:			$0.04/sec	
Thermal Conductivity:			134 (HEU), 120 (LEU-1) W/m-^{o}K	
Peak Output Values from Transient Analyses				
Transient	Max. Fuel, ^{o}C	Max. Clad, ^{o}C	Max. Coolant, ^{o}C [a]	Max. PWR, MW (At Time, sec)
	HEU/LEU-1	HEU/LEU-1	HEU/LEU-1	HEU/LEU-1
$1.2/sec	209/187	199/176	145/132	45.2/37.2
				(2.32/2.24)
$0.04/sec	155/155	149/148	116/117	28.2/22.2
				(25.6/25.8)

[a] The saturation temperature at 1.317 MPa is 192 ^{o}C.

Figure 3 shows the response of the MARIA reactor to the fast reactivity insertion rate ($1.2/sec). Similar results are shown in Fig. 4 for the slow reactivity insertion rate ($0.04/sec). These plots and the results given in Table 6 show that the peak values for reactor power, fuel temperature, clad temperature, and coolant temperature are similar for both the HEU and LEU-1 fuels and are all well within acceptable limits.

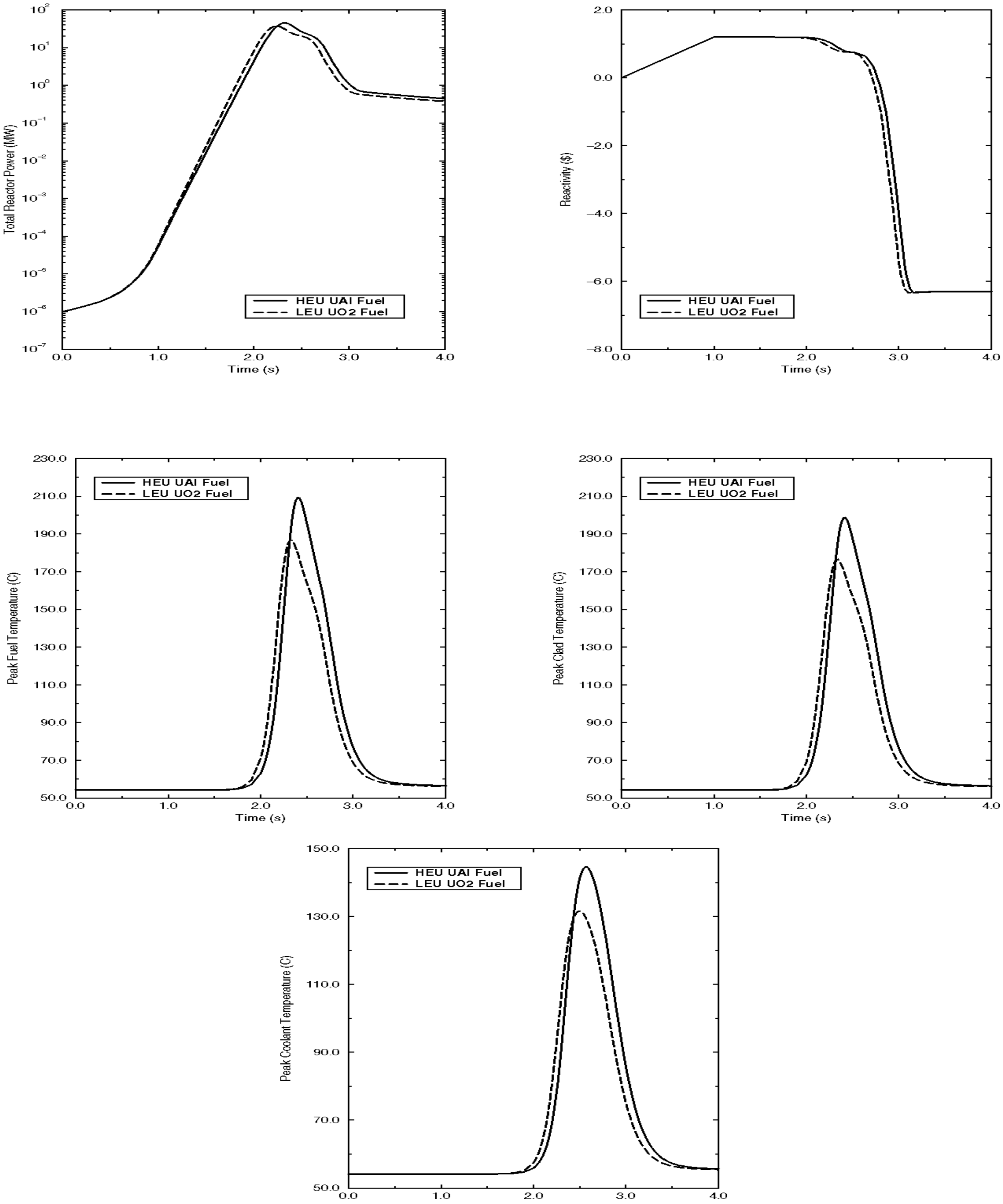


Figure 3.

MARIA Reactor Transients (Reactor Power and Peak Fuel, Clad, and Coolant Temperatures) from a $1.2/sec Reactivity Insertion Rate for 1.0 Second.

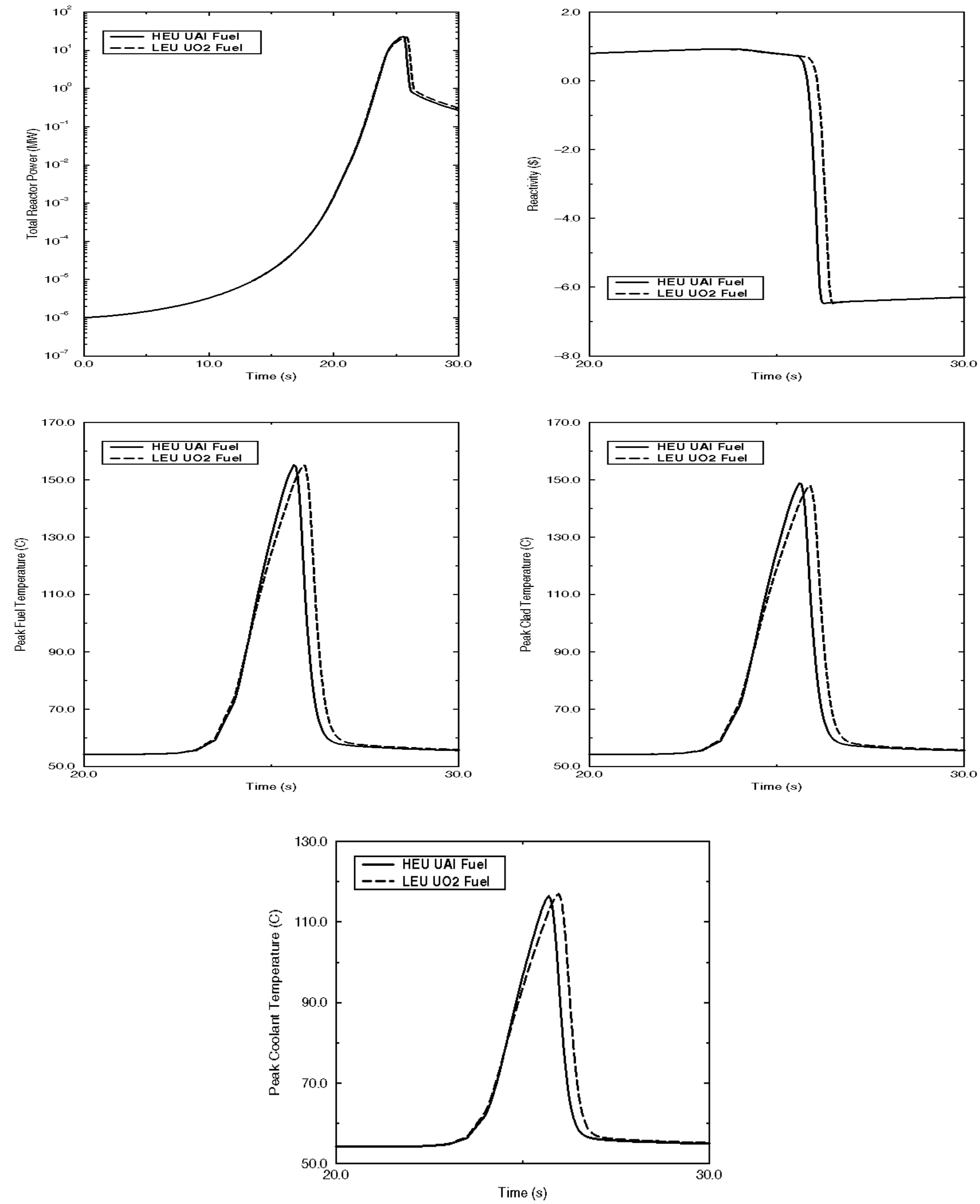


Figure 4.

MARIA Reactor Transients (Reactor Power and Peak Fuel, Clad, and Coolant Temperatures) from a $0.04/sec Reactivity Insertion Rate.

SUMMARY AND CONCLUSIONS

Based on equilibrium fuel cycle calculations[1] for the MARIA Research Reactor, a core configuration of 16 M6-type fuel assemblies (Fig. 2) with UO_2-Al LEU fuel (19.7% enriched, 402 g ^{235}U/FA, and ρ = 2.53 gU/cm^3) has a fuel cycle length that is 18% longer than the current core with U-Al HEU (80% enriched, 350 g ^{235}U/FA) fuel for the same 45% average discharge burnup. The peak thermal neutron flux in a key irradiation channel is about 3% lower in the LEU core than in the HEU core. Successful irradiation tests in the 1980's of UO_2-Al dispersion fuel (36% enriched) with ρ = 2.5 gU/cm^3 by the Russian reduced enrichment program have been reported[13].

For the transient analyses given in this report, kinetic parameters, reactivity feedback coefficients, control rod reactivity worths, and power distributions were calculated for the 16 fuel assembly MARIA reactor. The 1/v insertion method was used to determine prompt neutron lifetimes and prompt neutron generation times for fresh and burned HEU fuel and for fresh LEU. The results demonstrate the expected trend that the prompt neutron lifetime decreases as the neutron spectrum hardens when HEU fuel is replaced with LEU fuel.

Effective delayed neutron fractions were calculated with a perturbation code and ENDF/B-VI delayed neutron data. Family-dependent values for $\beta_{eff,i}$ and the decay constants λ_i are nearly the same for HEU and LEU cores which shows that these kinetic parameters are very insensitive to neutron spectrum changes. Delayed photo-neutrons from fission product gamma rays on beryllium contribute less than 2% to the total delayed neutron fraction. However, the decay constants for delayed photo-neutrons are much smaller than those for delayed fission neutrons.

Temperature and void reactivity feedback coefficients were determined for fresh HEU and LEU fuels. These feedback coefficients, $\alpha_x = d\rho(x)/dx$, and their statistical uncertainties, σ_x, were calculated by fitting a set of k(x) eigenvalues to a polynomial in x by the least squares process. Based on statistical considerations, temperature reactivity feedback coefficients are best determined by 2nd order (linear) fits for the temperature range (300K-550K) used in this study. However, a 4th order polynomial fit was used for void reactivity feedback coefficients. For the MARIA reactor, all the reactivity feedback coefficients are negative except for the beryllium temperature feedback coefficient which is positive.

Diffusion theory with transport-calculated group-dependent internal boundary conditions was used to determine control rod reactivity worths. These calculations, together with some results from Ref. 1, show that the control rods provide adequate shutdown margins for the 16 fuel assembly equilibrium cores with the HEU (80%, 350 g ^{235}U/FA) or with the LEU-1 (19.7%, 402 g ^{235}U/FA) fuels. Power distributions needed for the transient analyses were determined from 3D diffusion calculations with the safety rods withdrawn and with the bottom of the control rod absorbers located on the core midplane.

These kinetic parameters, reactivity feedback coefficients, and control and safety rod reactivity worths for both the HEU and the LEU-1 fuels were used to perform transient analyses.

Fast and slow reactivity insertion transients were analyzed for both the HEU and LEU-1 cases. Results show that both fuels respond to these transients in a very similar manner without excessive fuel, clad, or coolant peak temperatures.

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