NEUTRONIC SAFETY PARAMETERS AND TRANSIENT ANALYSES FOR POTENTIAL LEU CONVERSION OF THE IR-8 RESEARCH REACTOR

J. R. Deen, N. A. Hanan, R. S. Smith and J. E. Matos
Argonne National Laboratory
Argonne, IL, USA 60349-4815

and

P. M. Egorenkov and V. A. Nasonov
Russian Research Centre "Kurchatov Institute"
Reactor Technology and Material Research Institute
Moscow 123182, Russia

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ABSTRACT

Kinetic parameters, isothermal reactivity feedback coefficients and three transients for the IR-8 research reactor cores loaded with either HEU(90%), HEU(36%), or LEU (19.75%) fuel assemblies (FA) were calculated using three dimensional diffusion theory flux solutions, RELAP5/MOD3.2 and PARET. The prompt neutron generation time and effective delayed neutron fractions were calculated for fresh and beginning-of-equilibrium-cycle cores. Isothermal reactivity feedback coefficients were calculated for changes in coolant density, coolant temperature and fuel temperature in fresh and equilibrium cores. These kinetic parameters and reactivity coefficients were used in transient analysis models to predict power histories, and peak fuel, clad and coolant temperatures. The transients modeled were a rapid and slow loss-of-flow, a slow reactivity insertion, and a fast reactivity insertion.

INTRODUCTION

The IR-8 reactor, located at the Russian Research Center "Kurchatov Institute" in Moscow, has utilized IRT-3M FA containing HEU(90%) since 1981.[1] An IRT-3M FA with HEU(36%) and an IRT-4M with LEU(19.75%) have been proposed[2] as possible alternative fuels. The purpose of these calculations is to compare the performance of the reactor using all three FA during four hypothetical transients. The first two transients were a rapid and a slow loss-of-flow. The second two were slow and fast reactivity insertion transients. Reactor power histories and, peak clad, fuel and coolant temperatures were calculated. All transients were done using the kinetics and reactivity coefficients obtained from fresh cores. Use of fresh core data was found to be conservative compared to equilibrium cores. All reactivity coefficients become more negative and power peaking is reduced as core burnup increases.

CORE AND FUEL ASSEMBLY DESCRIPTIONS

The active core consists of 16 IRT-3M six-tube FA arranged in a 4x4 array. The core has a large 30 cm Be reflector on all radial sides as shown in Fig. 1 (reproduced here from Ref. 1). The central hole of the four corner FA, which is used for sample irradiations, was assumed to be filled with water. The remaining 12 FA each have control rods located in the center. Each control rod consists of a B_4C absorber section followed by an aluminum (SAV-1) displacer which is present in the core when the absorber is withdrawn. The core has 12 beam tubes positioned along the core mid-plane in the stationary Be reflector.

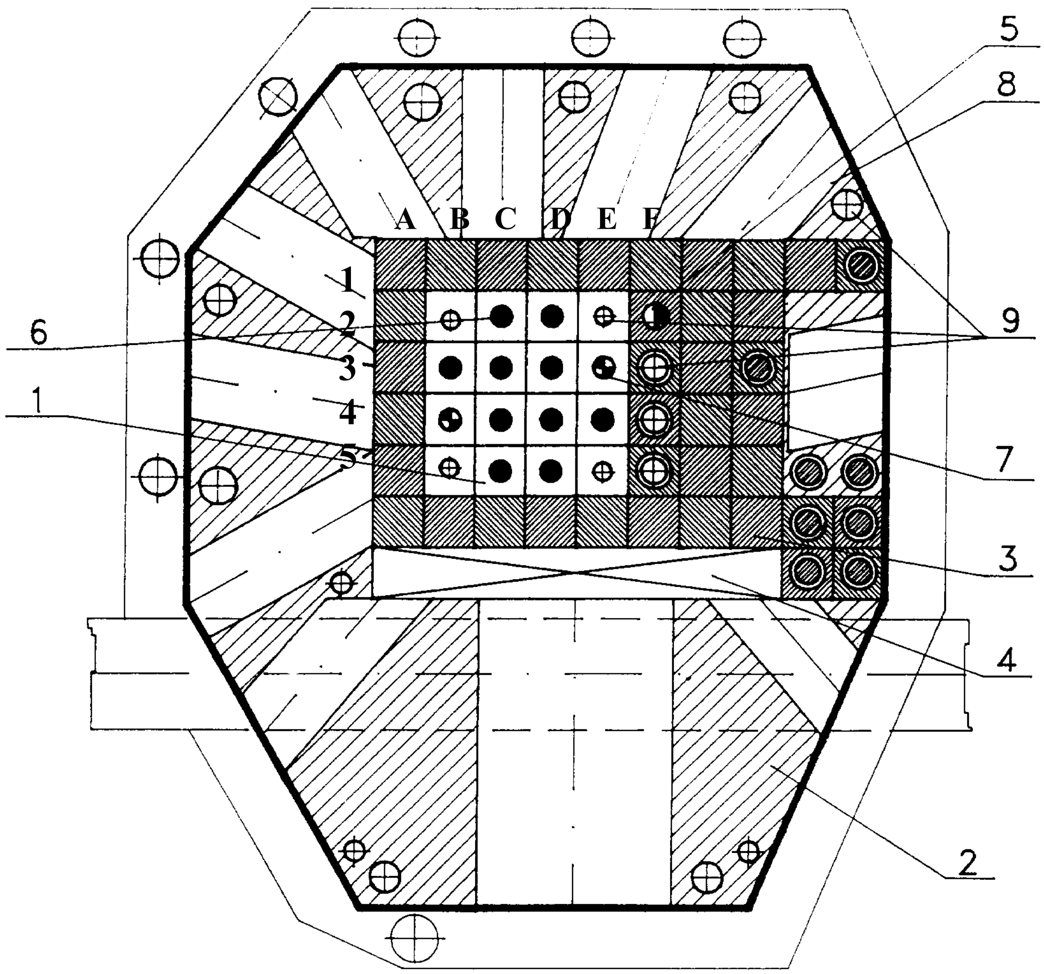


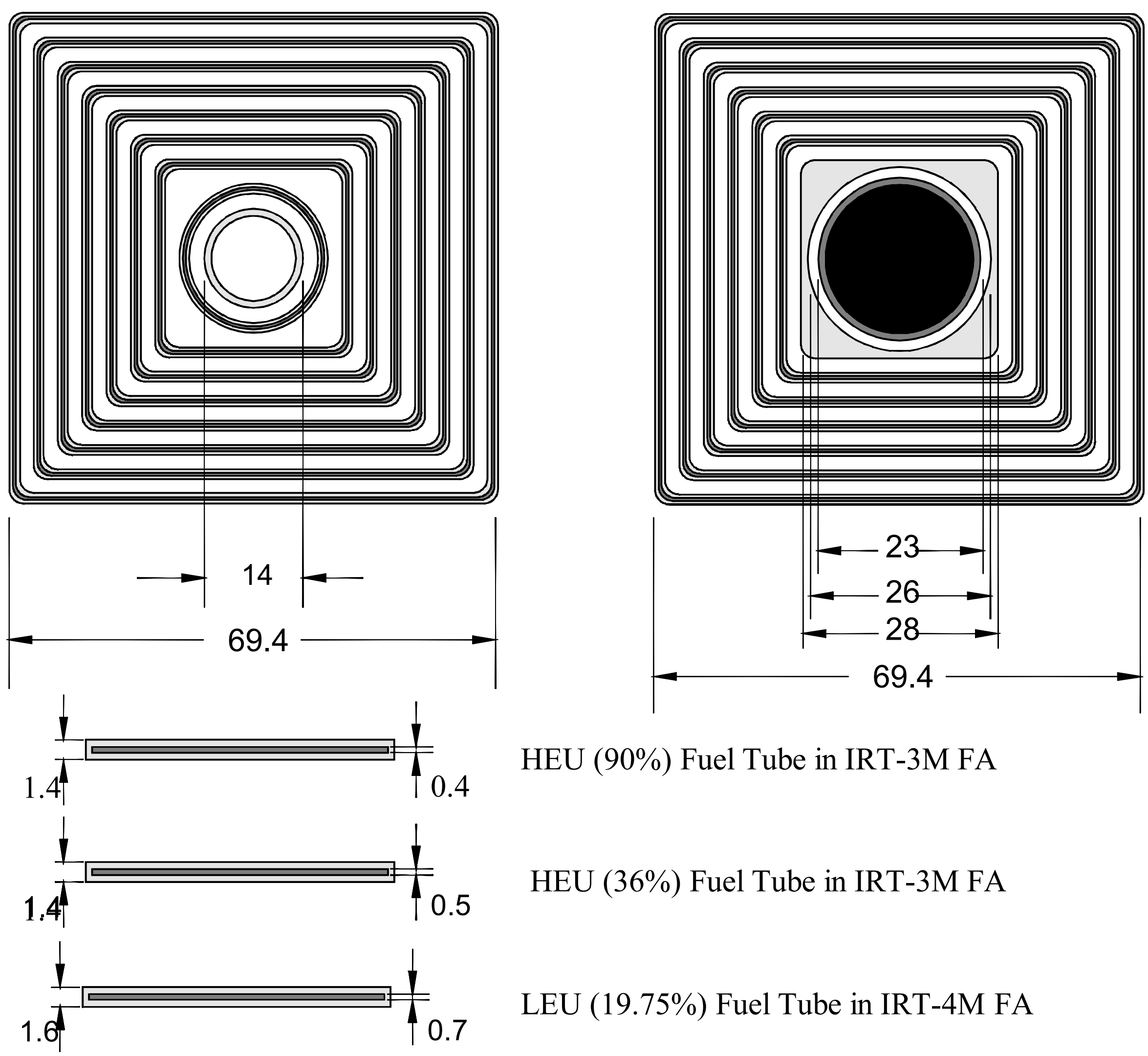
Figure 1. Load of the IR-8 Reactor Core.

1 - 6-tube FA; 2 - Blocks of stationary beryllium reflector; 3 - Removable beryllium block; 4 - Lead shield; 5 - Channel with automatic regulating rod of CPS; 6 - Channel with shim-safety rod of CPS; 7 - Channel with safety rod of CPS; 8 - Beam tube; 9 - Vertical experimental channel

The reactor currently uses the IRT-3M FA with 90% enriched uranium and a U-235 as-built loading of 272 g per 6-tube FA. A horizontal slice through the active fuel zone is shown in Fig. 2 for the 8-tube FA and companion 6-tube FA. All fuel tubes have rounded corners with an inner tube radius of 2.80 mm. The uranium density in the fuel meat of the HEU(90%) FA is 1.1 g/cm^3. The water channel thickness between fuel tubes is 2.05 mm. The fuel tubes are 1.4 mm thick with a 0.4 mm thick fuel meat region. The 36% enriched IRT-3M FA maintains the same control rod specifications and fuel tube dimensions, except that the fuel meat thickness is 0.5 mm and the clad thickness is 0.45 mm. The uranium density in the 36% enriched IRT-3M FA is 2.51 g/cm^3 which results in a loading of 309 g ^{235}U/ 6-tube FA.

Figure 2. IRT-3M Fuel Assembly Cross Section

8-Tube FA **6-Tube FA**



The IRT-4M FA design has the same number of fuel tubes as the IRT-3M FA but the fuel tube thickness is increased[3] from 1.4 mm to 1.6 mm. The fuel meat thickness is 0.7 mm and the clad thickness is 0.45 mm. The coolant channel between fuel tubes is reduced to 1.85 mm. Calculations have been performed for an LEU ^{235}U loadings of 352g (3.71 gU/cm^3) with UO_2-Al dispersion fuel.

REACTOR CORE NEUTRONICS MODEL DESCRIPTION

The reactor core and ex-core materials were modeled using XYZ multi-group diffusion theory as described in Reference 2. All REBUS3[4] fresh and equilibrium core models assume that the core is symmetrical about the core midplane. The neutron cross sections for the core materials were generated using the WIMS-ANL code[5] and a library with 69 energy groups based on ENDF-B/VI data and collapsed to seven broad energy groups for use in REBUS3.

KINETIC PARAMETERS FOR FRESH AND EQUILIBRIUM CORES

The kinetic parameters, namely the prompt neutron generation time (Λ) and the effective delayed fission neutron fraction (β_{eff}) were calculated for fresh and beginning of equilibrium cycle (BOEC) cores fueled with either HEU(90%) IRT-3M, HEU(36%) IRT-3M, or LEU(19.75%) IRT-4M fuel assemblies. The effective delayed fission neutron fractions were calculated in a perturbation theory code VARI3D[6] using delayed fission neutron data based on ENDF/B-VI. The delayed photoneutron contribution to the total β_{eff}, coming from the interaction of fission product gamma rays in the beryllium reflector, was not calculated. Based on Keepin's data[7], these delayed photoneutrons contribute less than 2% to the total value of β_{eff}. Prompt neutron generation times were calculated using the 1/v insertion method[8]. If a dilute and uniform distribution of a purely 1/v absorber is added to the entire reactor, the prompt neutron generation time can be calculated from the perturbed and unperturbed eigenvalues[9].

Table 1. IR-8 Equilibrium Core and Fresh Core Effective Delayed Neutron Fractions and Prompt Neutron Generation Times

Fuel Type (Enrichment)	Core Burnup	Effective Delayed Neutron Fraction	Prompt Neutron Generation Time
IRT-3M (90%)	BOL / BOEC	0.007509 / 0.007544	70.6 / 87.4
IRT-3M (36%)	BOL / BOEC	0.007448 / 0.007357	68.3 / 80.6
IRT-4M (19.75%)	BOL / BOEC	0.007485 / 0.007274	72.7 / 81.3

The kinetic parameters are presented in Table 1. There are only very small reductions in β_{eff} for the fresh and equilibrium IRT-3M(36%) and the IRT-4M(19.75%) fueled cores compared to the IRT-3M(90%) fueled cores. The Λ shows only small changes as a function of fuel assembly loading in the core. There is a few percent increase in Λ for the equilibrium cores compared to the fresh core values.

REACTIVITY FEEDBACK COEFFICIENTS

The changes in whole core reactivity, caused by isothermal increases in fuel and coolant temperatures and decreases in coolant density, were calculated using the DIF3D code[10] for fresh and equilibrium cores. From these computed reactivities, reactivity coefficients were calculated as a function of fuel or coolant temperature increase or coolant void fraction.

Each of the changes in temperature or coolant density required a different set of microscopic neutron cross sections for the whole core model calculations. The reference coolant temperature was 23°C. Other coolant temperatures calculated were 47.5°C, 75°C and 100°C. The reference fuel temperature was 27°C. Fuel temperature increases to 47.5°C, 75°C, 100°C, 200°C, and 400°C were used to calculate the Doppler coefficient. Decreases in coolant density were calculated from a reference water density of 0.9975 g/cm^3 to model heating of the coolant and resultant boiling to a coolant void fraction of 20%.

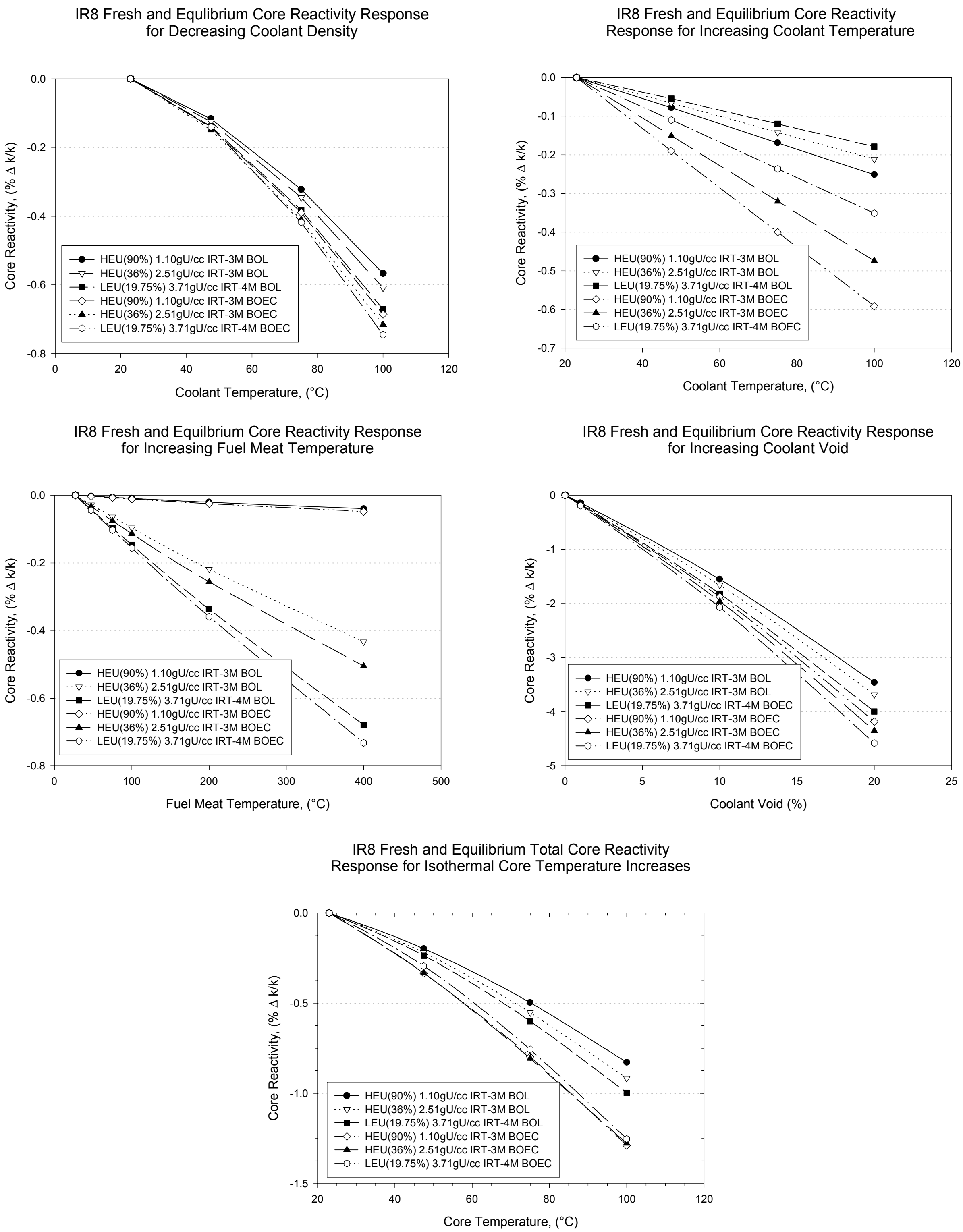
The coolant temperature reactivity coefficients for the fresh and equilibrium cores are shown in Table 2 and plotted as a function of temperature in Fig. 3. This coefficient is influenced by the thermal neutron spectrum of the core. As the enrichment is reduced and more U-235 is added, the neutron spectrum becomes harder causing a reduction in the coolant temperature coefficient for the reduced enrichment fresh cores. The coolant temperature coefficients for equilibrium cores are more negative than in fresh cores because of the softer neutron spectrum in burned cores.

The coolant density coefficient is the largest contributor to the total core reactivity coefficient for isothermal core heating from 27°C to 100°C. The core reactivity response to coolant voiding is shown in Table 2 and plotted in Fig. 3. Lower enrichment and higher burnup cores have larger negative coolant void reactivity coefficients. The harder neutron spectrum in the reduced enrichment cores leads to greater neutron leakage than in the HEU(90%) core.

The core reactivity response to fuel temperature increases in fresh and equilibrium cores is also presented in Fig. 3. The magnitude of the Doppler coefficient is determined primarily by the quantity of ^{238}U and the neutron spectrum in the cores.

The last plot shown in Fig. 3 presents the total reactivity loss in each core upon isothermal heating from 23°C to 100°C. For the fresh cores, the LEU case is more negative than either the HEU(90%) or HEU(36%) cases. The equilibrium cores have more negative reactivity values than fresh cores over the temperature range shown. The last plot was calculated by summing the fuel and coolant temperature and the coolant density components. Three separate DIF3D calculations at 100°C were performed to check the validity of summing the three components. The reactivity response at 100°C indicated close agreement to the reactivity obtained by summing the components

Figure 3. IR-8 Fresh and Equilibrium Core Reactivity Response to Changes in Fuel Temperature and Coolant Temperature and Density



REPRESENTATIVE TRANSIENTS

Four generic transients were modeled using the RELAP5 code[11] and kinetic parameters and reactivity coefficients from fresh cores with 90%, 36%, and 19.75% enrichments. Radial and axial peaking factors were calculated for each of the cores with all control rods positioned at the core mid-plane. The total peaking factor for the HEU(90%) core was 2.39 with an axial peaking factor of 1.25. All transient input parameters are presented in Table 2.

The first transient was a fast reactivity insertion of $1.2\beta_{eff}$ in one second at a power of 1 Watt. There was assumed to be an overpower trip set at 120 % power (9.6 MW) without a period trip before a linear shutdown reactivity insertion of $-10\beta_{eff}$ in 0.5 seconds with a 25ms delay before the rods begin to drop. A series of five plots are presented in Figure 4 which trace the reactivity, total power, and the peak fuel, clad, and coolant temperature histories during the transient. The peak fuel and clad temperatures are separated by less than 5°C with a peak fuel temperature of 111°C as shown in Table 3. The peak coolant temperature is below 70°C. The transient response presented in Figure 4 is similar for all three fuels modeled. Similar results were obtained using the PARET[12] single plate model for this transient.

The second transient was a slow reactivity insertion of $0.10\beta_{eff}$ /second beginning at a power of 1 Watt until overpower trip was initiated at 9.6 MW. The plots of reactivity, power, and peak temperatures presented in Figure 5 show that it takes about 10.5 seconds before the scram is initiated. The peak clad temperature is about 105°C and the peak coolant temperature is about 70°C. Again there is no significant difference in the peak temperatures or power histories for the HEU(90%), HEU(36%), or LEU(19.75%) fueled cores.

The results of the slow loss-of-flow transient is presented in Figure 6. This transient was initiated at 8 MW with an exponential reduction in flow which takes about 4 seconds to reach 85% of full flow. The flow rate trip is set at 85% of full flow. The time delay for initiation of the scram is 200 ms. During the flow rate reduction before scram, temperatures increase about 5°C until a peak clad temperature of 124°C is reached for the LEU core. Peak clad temperatures for the HEU(90%) and HEU(36%) cores were 120°C and 124°, respectively. The HEU peak clad temperature at the initiation of this transient of 116°C is in good agreement with peak temperature of 110°C reported by RRCKI[1]. The peak coolant temperatures also increased 5°C until a peak of 80°C was reached before the scram. After the scram there is sufficient flow rate to reduce the peak clad and fuel temperatures more than 60°C to 55°C in less than 1 second.

The results of the rapid loss-of-flow transient is presented in Figure 7. This transient was also initiated at 8 MW with a flow rate trip set at 85% of full flow. The flow was assumed to decrease exponentially to 13% of full flow in 2 seconds. The flow rate is reduced by 15% of full flow in 0.16 seconds. During this brief interval temperature increases were the same as those calculated for the slow-loss-flow reported in Figure 6. The rapid reduction in power initiated by the scram caused a drop in peak fuel and clad temperatures of 55°C during the first second of the transient.

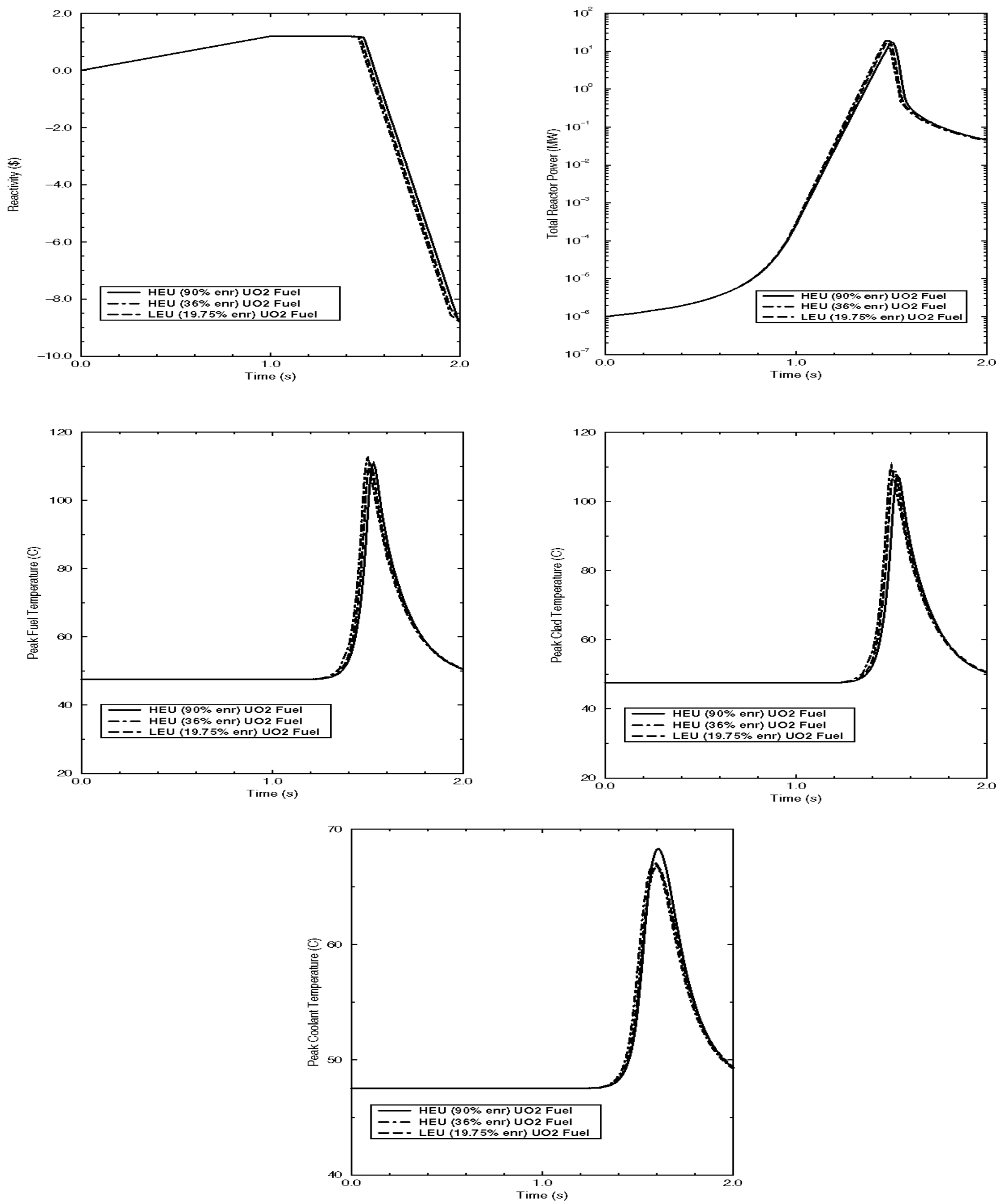


Figure 4. IR-8 Fast Reactivity Insertion of 1.2β_{eff} in 1.0 sec. with Power Trip at 9.6 MW

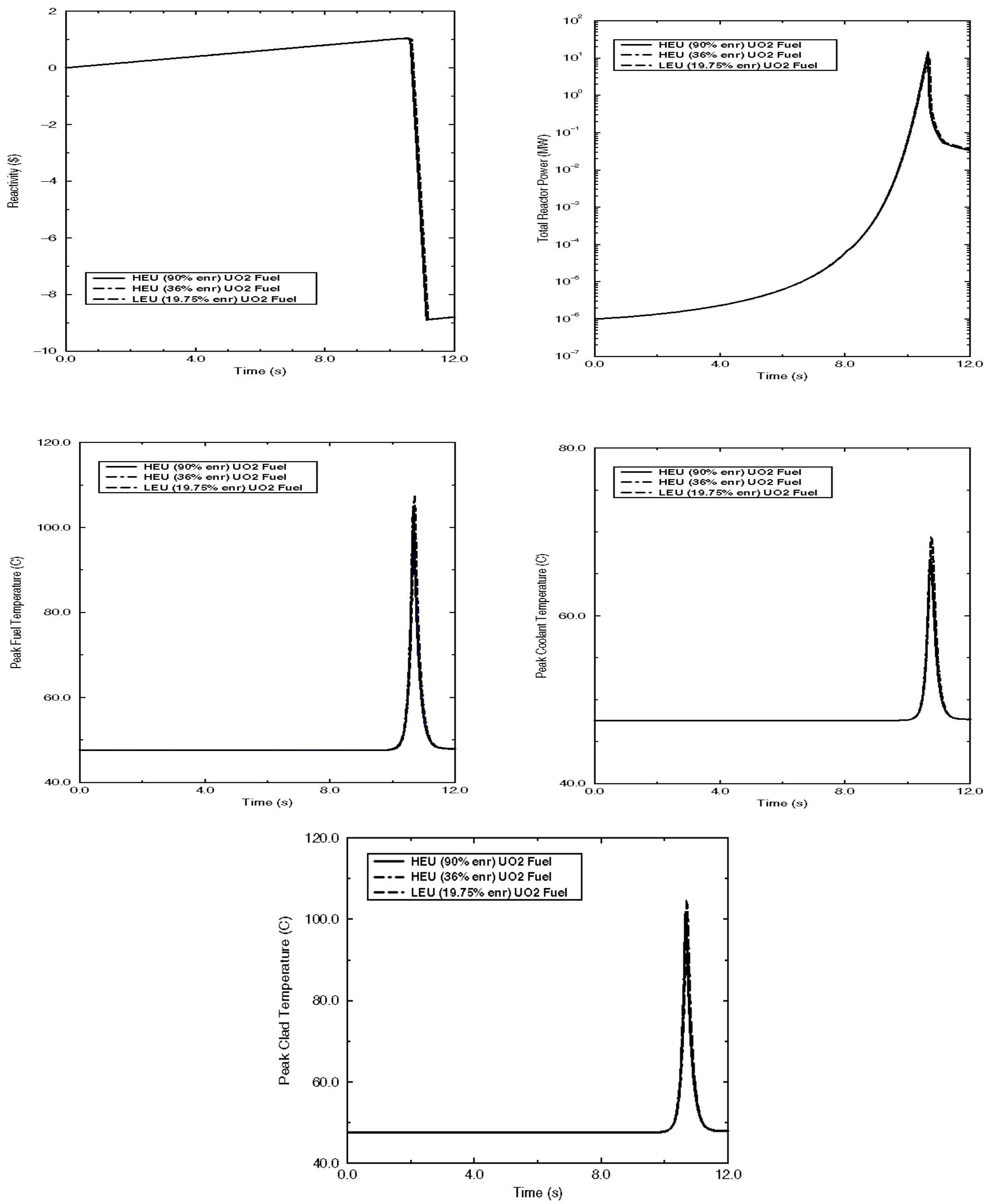


Figure 5. IR-8 Slow Reactivity Insertion of 0.10β_{eff} /sec. with Power Trip at 9.6 MW

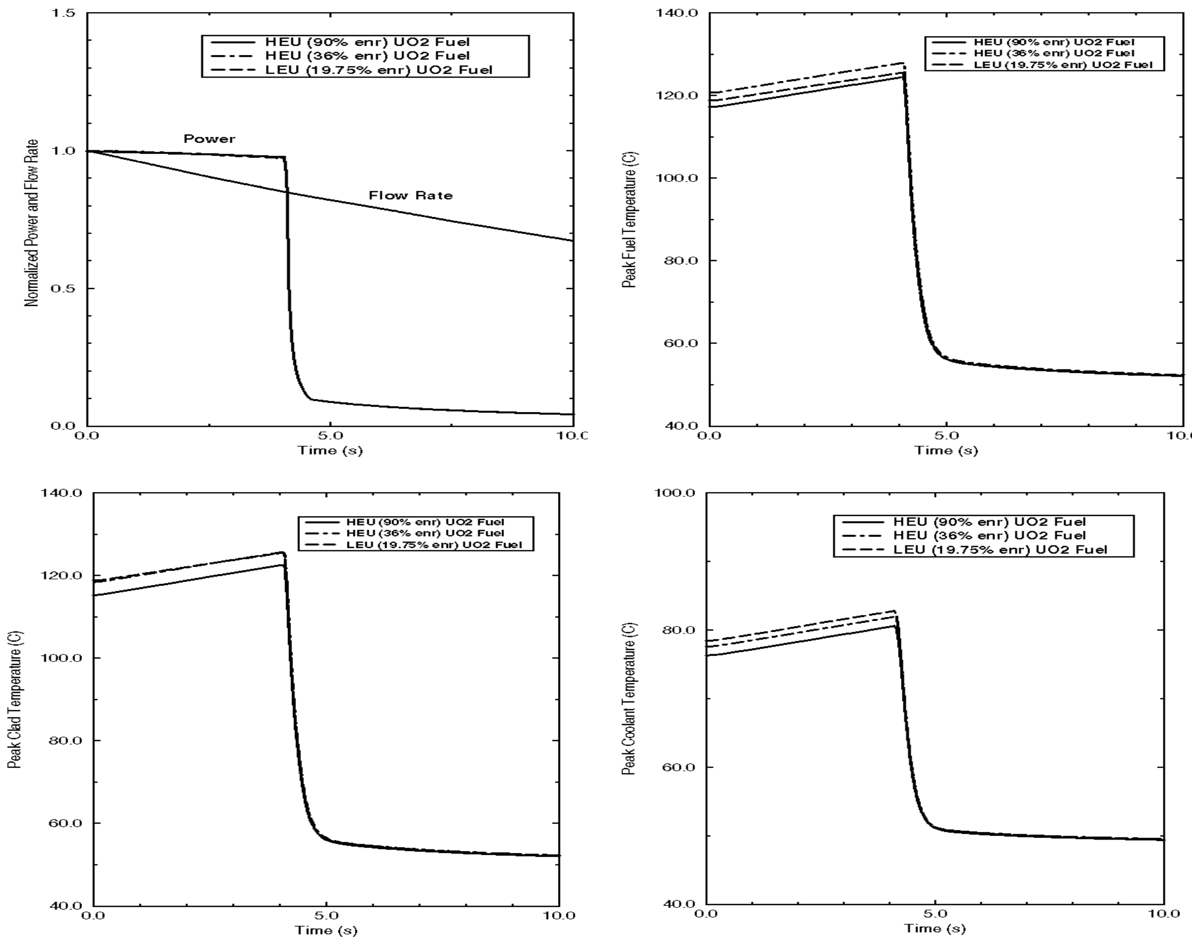


Figure 6. IR-8 Slow Loss of Flow with Trip at 85% of Full Flow

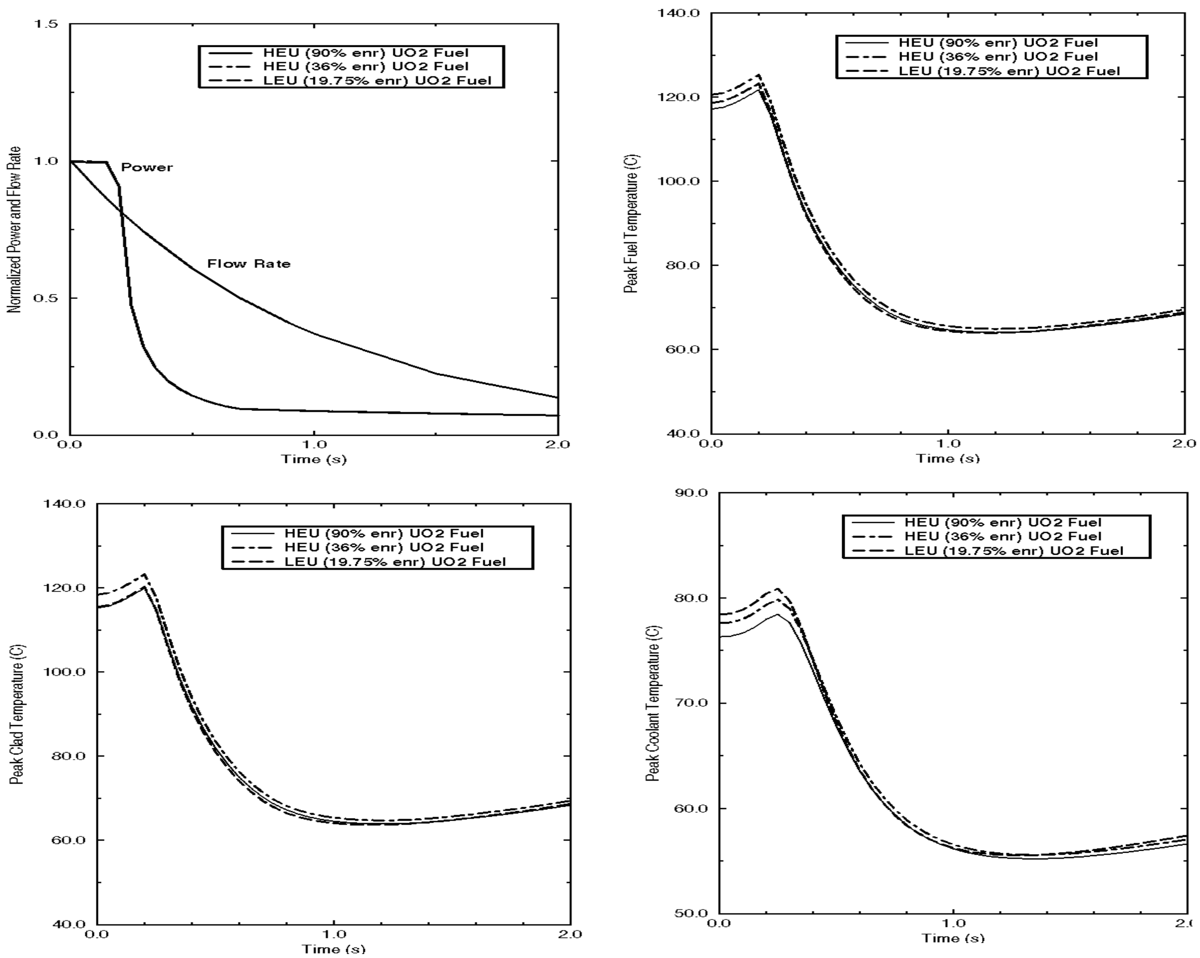


Figure 7. IR-8 Fast Loss of Flow at 8 MW with trip at 85% of Full Flow

Table 2. IR-8 Fresh and Equilibrium Core Reactivity Feedback Coefficients (- % $\Delta\rho$/°C)

Reactivity Coefficient	Core Burnup	IRT-3M (90%)	IRT-3M (36%)	IRT-4M (19.75%)
Coolant Density	BOL / BOEC	4.78E-3 / 6.06E-3	5.14E-3 / 6.09E-3	5.69E-3 / 5.73E-3
Fuel Temperature	BOL / BOEC	1.39E-4 / 1.70E-4	1.39E-3 / 1.65E-3	2.09E-3 / 2.18E-3
Coolant Temperature	BOL / BOEC	3.19E-3 / 7.76E-3	2.67E-3 / 6.16E-3	2.25E-3 / 4.49E-3
Total	BOL / BOEC	8.11E-3 / 1.40E-2	9.20E-3 / 1.39E-2	1.00E-2 / 1.24E-2

Table 3. Input Parameters for Transient Analyses

Input Parameter	Value
Steady State Power	8.0 MW
Coolant Flow Rate/Fuel Assembly	30 m^3/hr
Inlet Coolant Temperature	47.5 °C
Power Trip - 20% Above Steady State Power	9.6 MW
Flow Trip - 85% of Full Flow/Fuel Assembly	25.5 m^3/hr
Peak-to-Average Power Density	2.85/3.02/3.11*
Safety Rods Worth (assumed)	10 β_{eff}
Safety Rods Insertion Time	0.5 sec.
Safety Rods Time Delay in Reactivity Insertion	25 ms
Safety Rods Time Delay in Loss-of-Flow	200 ms
Fast Reactivity Insertion	1.2 β_{eff} in 1.0 sec.
Slow Reactivity Insertion Rate	0.10 β_{eff}/sec
Thermal Conductivity	172/138/60* W/m-°K

* HEU(90%) / HEU(36%) / LEU(19.75%)
Peak to average power density is for control rods positioned at core midplane.

Table 4. Peak Temperatures and Powers from Transient Analyses

Transient	Fuel Assembly Type	Peak Fuel Temperature (°C)	Peak Clad Temperature (°C)	Peak Coolant Temperature (°C)	Peak Power (MW)
1.2 β_{eff} in 1.0 sec.	IRT-3M(90%)	111	108	68	17.4
	IRT-3M(36%)	113	110	67	18.9
	IRT-4M(19.75%)	111	109	67	18.9
0.10 β_{eff} /sec.	IRT-3M(90%)	104	102	66	12.1
	IRT-3M(36%)	104	101	67	11.2
	IRT-4M(19.75%)	108	104	69	11.0
Fast Loss-of-Flow	IRT-3M(90%)	122	120	78	N/A
	IRT-3M(36%)	125	123	80	N/A
	IRT-4M(19.75%)	123	120	81	N/A
Slow Loss-of-Flow	IRT-3M(90%)	124	123	81	N/A
	IRT-3M(36%)	128	126	82	N/A
	IRT-4M(19.75%)	126	122	83	N/A

CONCLUSIONS

The IR-8 core was safely shutdown without coolant boiling after each reactivity insertion and loss-of-flow transient modeled with RELAP5 using either the fresh HEU(90%), HEU(36%) or LEU(19.75%) fuels. The power histories, peak clad and fuel meat temperatures, and peak coolant temperatures for each of the three cores were very close during each transient. Therefore the use of either reduced enrichment fuel design has almost no impact upon the result of any of the transients modeled in this paper. The use of fresh fuel core models is conservative from a safety perspective since power peaking factors become smaller as the core burns.

There are only very small reductions in β_{eff} for the fresh and equilibrium IRT-3M(36%) and the IRT-4M(19.75%) fueled cores compared to the IRT-3M(90%) fueled cores. The prompt neutron generation times show only small changes as a function of fuel assembly loaded in the core. There is a few percent increase in prompt neutron generation times for the equilibrium cores compared to the fresh core values.

The total isothermal reactivity coefficient becomes more negative as the fuel enrichment is reduced for fresh cores. As each core depletes, the reactivity coefficients become more negative than in a fresh core.

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