KINETIC STUDY OF TRR CORE IN FUEL CONVERSION FROM HEU TO LEU

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Abstract

In the conversion of fuel in the Tehran Research Reactor core from Highly-Enriched Uranium to Low-Enriched Uranium fuel neutronics analysis, thermal hydraulic calculations and kinetic performance of the core have to be studied. In this study, static and dynamic core performance for HEU and LEU fuels were investigated. In static conditions, two groups of neutron flux distributions in axial direction were obtained by solving diffusion equations. In transient cases, the slow and fast transient phenomena, small reactivity and large reactivity insertion effect, were investigated. To find the temperature feedback effect on core reactivity, such as coolant drainage malfunction, the effective multiplication factor was calculated for two different temperatures, namely 293° K and 373° K, using the computer code WIMS-D/4. From here, the average temperature coefficient of reactivity was obtained. It should be pointed out that the coefficient dependence on temperature is approximately a linear function. In fast transient study, for HEU fuel \$1.5/0.5 reactivity and for LEU fuel \$1.35/0.5 reactivity were considered. In slow transient study, a reactivity equivalent to \$0.0085/s for HEU fuel and \$0.0034/s and \$0.0085/s in the case of LEU fuel was introduced into the core. The result of calculations show that in fast transient in the case of LEU fuel, reactor power drops sharply before the scram time set is reached. It seems as if the overall performance of the LEU core is comparable with that of the HEU fuelled core.

Introduction

In recent years, in virtue of the non-proliferation policy, highly-enriched uranium (HEU) fuel was unavailable for refuelling the research reactor cores. Accordingly, many reactor operators with the technical and scientific assistance of IAEA [1] have planned to refuel their reactors with medium-enriched uranium (MEU) or low-enriched uranium (LEU) fuel. In the fuel replacement some changes in the thermo-kinetic behaviour of the core inevitably take place. In this study, kinetic performance of the Tehran Research Reactor (TRR) core has been investigated.

Keywords: HEU to LEU conversion; Slow and fast transients; 20% enriched fuel; Fuel replacement; Temperature coefficient of reactivity

TRR Core Specification

The existing TRR HEU fuelled core is a 5MW pool-type research reactor with MTR type fuel elements which can go critical with 19-24 standard fuel elements (SFE) and 4-5 control fuel elements (CFE) plus one regulating rod. Each SFE consists of 18 curved plates of which two end plates are dummy, all of which are affixed to two supporting side plates. The core is water reflected on all sides, and on the west side the thermal column is located with 20 cm of lead to shield it against strong core gamma-rays. The core has six beam tubes facing the reactor core and one through tube plus two pneumatic irradiation facilities. The LEU core has nearly the same configuration as the HEU core, except that it has one row of graphite blocks (8 cm thick) on the opposite side of the thermal column to enhance core reactivity. The fuel plates are assumed to be flat and all are loaded with slightly less than 20% enriched uranium of U_9O_8-Al fuel meat. Figure 1 shows the horizontal cross-sectional cut-away of the TRR core. The core is cooled by demineralized water at the rate of $500m^3/h$. In normal operations, the average inlet temperature is normally 23°C, but the designed inlet temperature is 38°C and the outlet temperature is increased by about + 1.6°C per MW power [2]. In critical cases, the temperature difference of inlet and outlet can reach as high as +12°C.

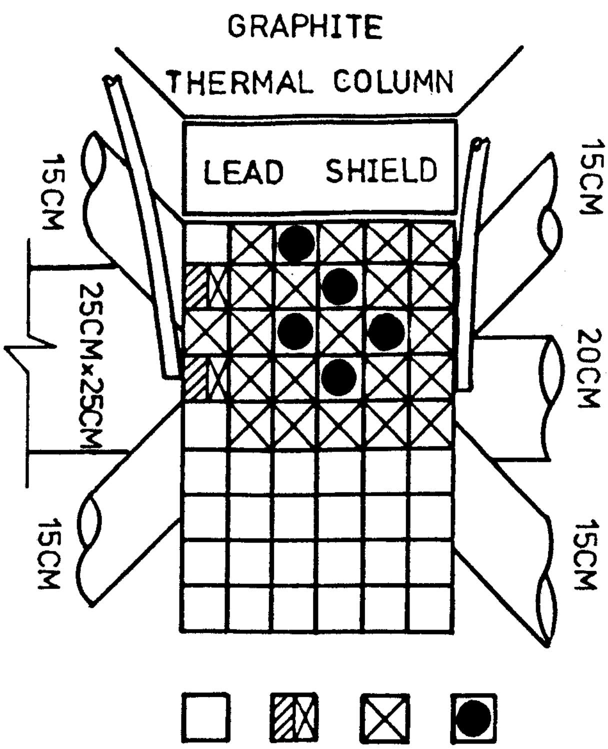

Figure 1. TRR core arrangement

Method of Analysis

The treatment of time-dependent behaviour of neutron population in the TRR core was carried out for both fuels of HEU and LEU using two group time-dependent diffusion equations [3-4]:

$$D_1\nabla^2\Phi_1 - (D_1/\tau + D_1B^2)\Phi_1 + K_{00}(1-\beta)\Sigma a_2\Phi_2 + \sum_{1}^{6}\lambda_i C_i = 1/v_1\,(\partial\Phi_1/\partial t) \quad (1)$$

$$D_2\nabla^2\Phi_2 - (\Sigma a_2 + \Sigma p + D_2B^2)\Phi_2 + pD_1\Phi_1/\tau = 1/v_2(\partial\Phi_2/\partial t) \quad (2)$$

where

D_1 and D_2	=	diffusion coefficients of groups 1 and 2, cm
Φ_1 and Φ_2	=	neutron fluxes of groups 1 and 2
v_1 and v_2	=	neutron velocities of groups 1 and 2, cm/s
B^2	=	reactor core buckling, cm^{-2}
τ	=	Fermi Age, cm^{-2}
K_{00}	=	infinite multiplication factor
Σa_2	=	absorption cross-section for group 2, cm^{-1}
Σp	=	absorption cross-section of poison, cm^{-1}
λ_i	=	decay constant of precursor i=1,6
C_i	=	concentration of precursor i
β	=	effective delayed neutron fraction
p	=	resonance escape probability

To solve the equations, they are transformed into a set of linear equations by the finite difference method [3, 5], by subdividing the dimension (radius or slab thickness) of the reactor core into suitable mesh size which must be less than the core mean free path and the time intervals into constant time steps. This concept has been employed in the computer code COSTANZA [4].

In order to explore the effect of temperature changes on reactor dynamics, the reactivity changes due to a sudden change in the reactor core environment were calculated. The effect of temperature rise on core reactivity is manifold, including Doppler broadening effect [3], moderator dilution, slight increase of Fermi Age, increase of diffusion length and increase of physical core dimensions. As a result of power surge of the reactor core, in an adiabatic phenomenon the fuel temperature rises sharply. Since nuclear fuel composite contains, except in some special reactors, a very high proportion of U-238, and since U-238 has many narrow capture resonances, in the range of 6 to 100ev, as the fuel temperature rises, U-238 resonances broaden and non-fissionable absorption increases. This effect, the so-called Doppler broadening effect, results from a negative reactivity in the reactor core. This effect is of prime importance in reactor design, but other coefficients are by no means of less importance.

Other temperature reactivity coefficients are also carefully determined. These changes reflect as a feedback effect on core reactivity. The total averaged temperature reactivity coefficient $\alpha(T)$ was calculated using transport equation code WIMS-D/4 [6, 7]. The critical parameters, namely effective multiplication factor, for two different temperatures i.e., 293°K and 373°K were calculated. Then the parameters were fed as input data to the code COSTANZA. It is seen that in transient phenomena following a power rise reactor, power drops rather sharply owing to $-\alpha(T)$. In this case, the temperature distribution along the coolant channel was obtained using the general heat conduction equation [5].

$$k\nabla^2 T + P = C_p \rho \frac{dT}{dt} \qquad (3)$$

where

- k = thermal conductivity of media W/m.°K
- P = specific power W/m^3
- Cp = specific heat J/kg°K
- ρ = density kg/m^3

The numerical solution of the equation is given in the code COSTANZA using finite difference equations. The finite difference expressions for diffusing media are set up in fuel, cladding and coolant by methods elucidated in numerical analysis text books and reference [5]. Noting that in cladding and coolant no heat is generated i. e., P = 0, therefore the time-dependent heat transfer equation is given as:

$$h.s(T_c - T_b) - C_p.G\frac{dT}{dz} = C_p.\rho A\frac{dT_c}{dt} \qquad (4)$$

where

- h = heat transfer coefficient between clad surface and coolant W/m^2°K
- S = clad surface m^2
- T_c = clad surface temperature K
- T_b = averaged coolant temperature K
- G = coolant flow rate m^3/h or kg/s
- A = sectional area of coolant channel m^2

In the computational analysis, the reactor core was axially divided into five homogenized regions from top to bottom, as shown in Figure 2. For point computational purposes the axial direction was divided into 92 mesh points and for transient analysis time steps into millisecond intervals.

Region 1	10 cm	H_2O
Region 2	6 cm	H_2O + Al
Region 3	30 cm	H_2O + Al + U-235 + U-238 + Poison
Region 4	30 cm	H_2O + Al + U-235 + U-238 + Poison
Region 5	5 cm	H_2O + Al

Control Rods

Figure 2. Computational regions of reflected core

In each region, all elements were uniformly diffused in the medium. As regards permanent fission products with reference to the operational history of the core, the absorption cross-section was estimated at 50 barns per initial U-235 atom [8]. The present control rods are oval type made of aluminium in the form of an elliptical can with a thin lining of absorber composed of Cd-In-Ag with 5w%, 15w% and 80w% respectively, see Figure 3. The total reactivity of four control rods and one regulating rod is about 12.5% $\Delta k/k$.

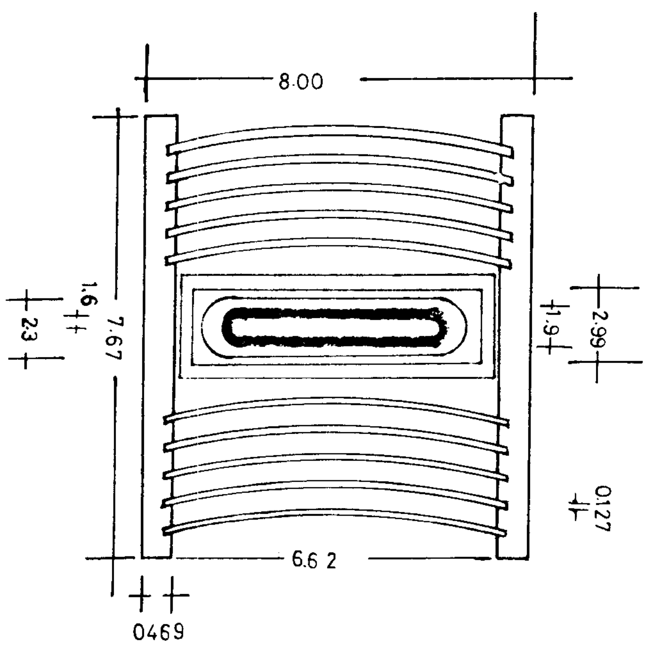

Figure 3. Horizontal cross-secti n of a control rod

Based on many years of experience in operating research reactors around the world, it is found that accidental reactivity introduction into the core, such as beam tube flooding; fissile material insertion into the core; introduction of an absorber or a reflective material in thermal neutron flux peaking; a drop in coolant flow rate resulting in fuel temperature increase or accidents in fuel loading; introduces small or large reactivities into the core. In the case of fuel element drop accident an equivalent reactivity of about \$1.5 in 0.5s is assumed. In the case of slow transients due to control rod withdrawal or other effects, \$0.0085/s of reactivity in the HEU core and \$0.0085/s and \$0.0034/s for the LEU fuel were assumed being inserted into the core. In this case the averaged two group fluxes, which are proportional to the reactor power, were obtained, see Figure 4. It was found that the LEU core is more susceptible to any reactivity change in the core than the HEU core. This reactor behaviour may be due to the fact that the LEU core has more U-235 fuel loading than the HEU core and furthermore, four times the U-235 mass. U-238 is introduced into the core to reduce the enrichment to 20%. Therefore, as a result of the control rod(s) movement, neutron spectrum is locally hardened and due to local flux depression, overall power drops. To compensate for the drop in power, neutron flux should increase in the vicinity of the control rod(s). This flux bulge leads to more fissioning in fissile material as well as resonance capture in U-238 resonance.

In transients, the clad surface temperature may pass the onset of nucleate boiling (ONB) in 0.6s. However, based on many years of operating experience of these types of reactors, it has been suggested by the experts that 0.5s is set as a scram safety margin [9]. Since there is not much experience in the world with the LEU fuelled core, the reactivity insertion has been chosen at between \$1.35 to \$1.5/0.5s for LEU fuel to be on the safe side.

Results and Discussion

The main objective of this study was to investigate time-dependent performance of the LEU core as compared to that of the HEU core in fuel conversion. In the criticality search, temperature was taken as T=20°C and the control rods inserted about 40% (6×0.40 = 24 cm) in the core which was assumed to be uniformly diffused in the upper core. The criticality was reached by automatically adjusting the absorber level in the upper core by changing rod length in the core. After the criticality was achieved, two group flux distributions in axial direction were calculated from

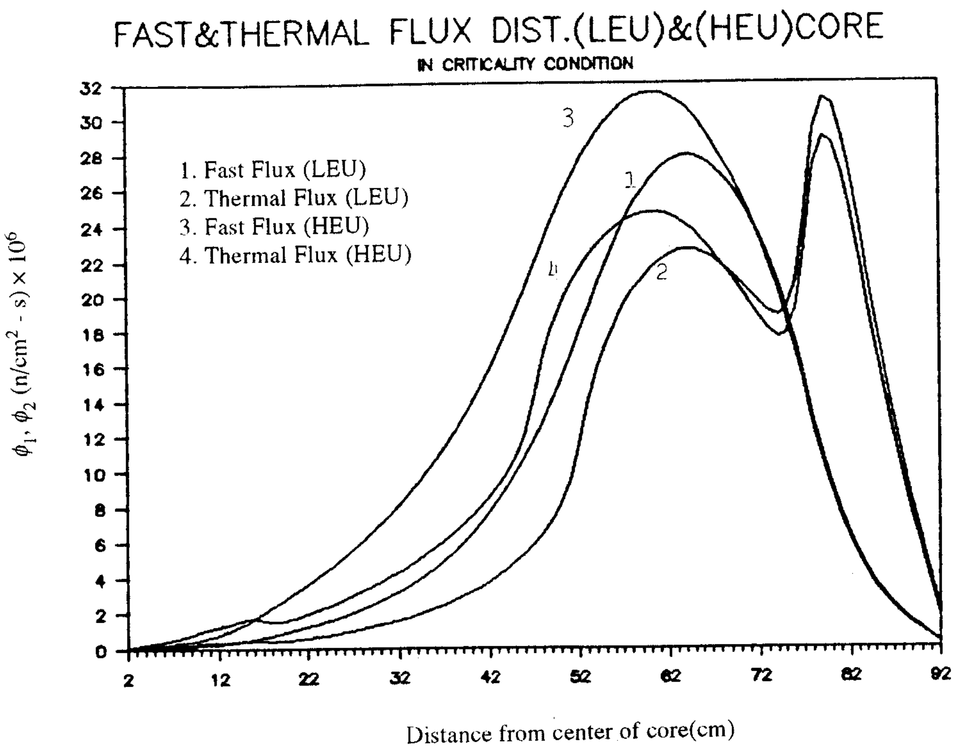

Figure 4. Two group flux distributions in axial direction for HEU and LEU fuelled core

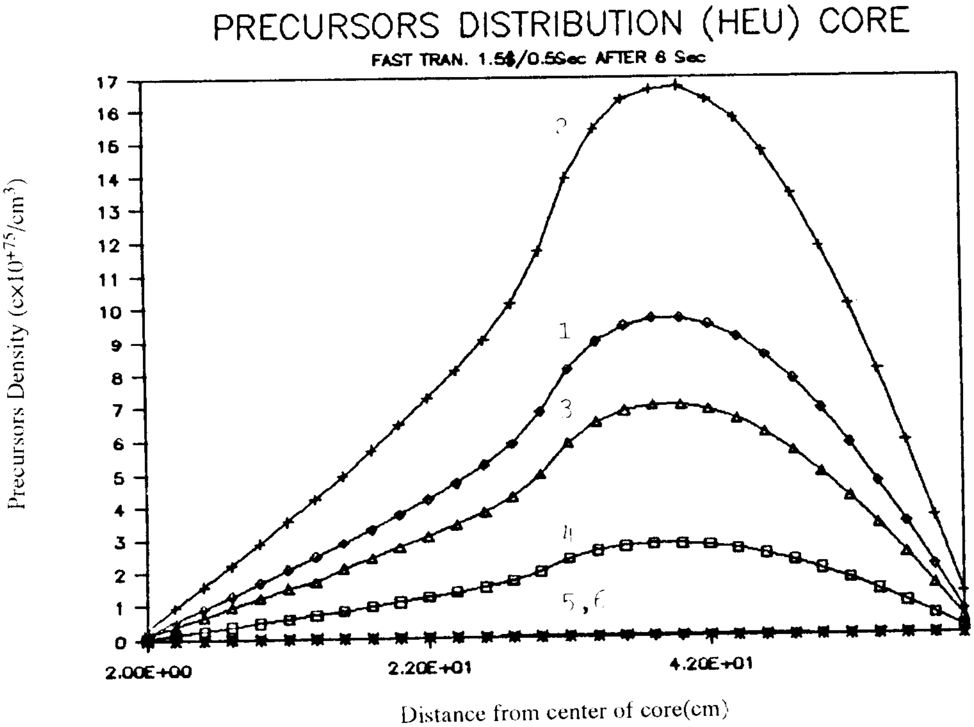

Figure 5. Six group precursor atomic densities for HEU fuelled core

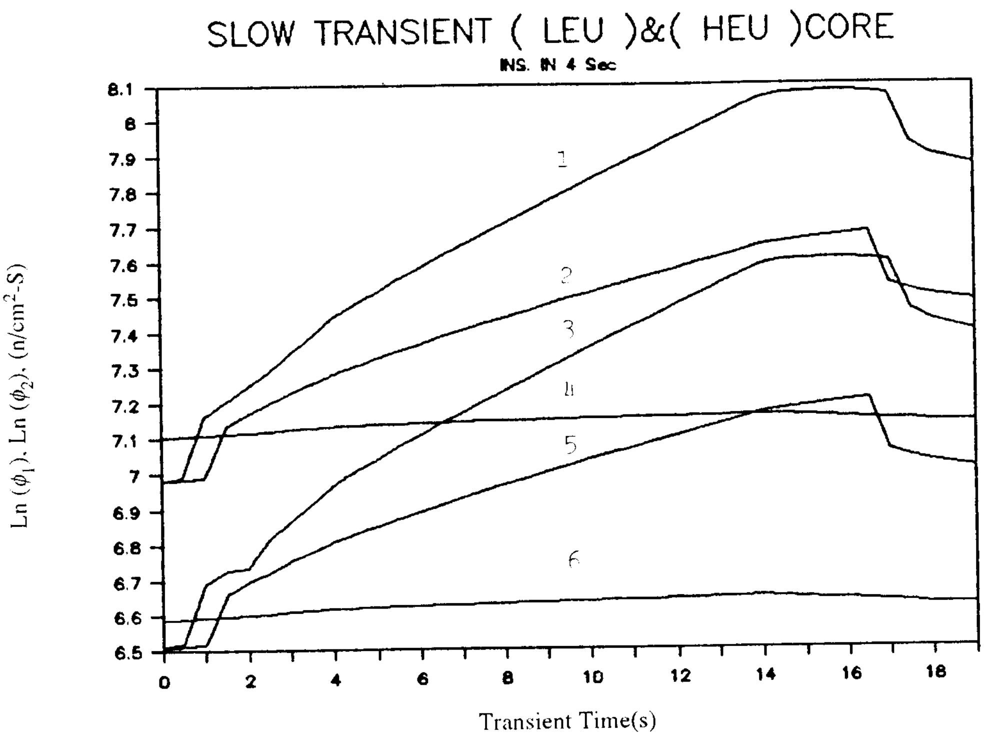

Figure 6. Slow transients for HEU and LEU fuel

1. Fast Flux (LEU, 0.0085 \$/s)
2. Thermal Flux (LEU, 0.0085 \$/s)
3. Fast Flux (LEU, 0.034 \$/s)
4. Thermal Flux (LEU, 0.0034 \$/s)
5. Fast Flux (HEU, 0.0085 \$/s)
6. Thermal Flux (HEU, 0.0085 \$/s)

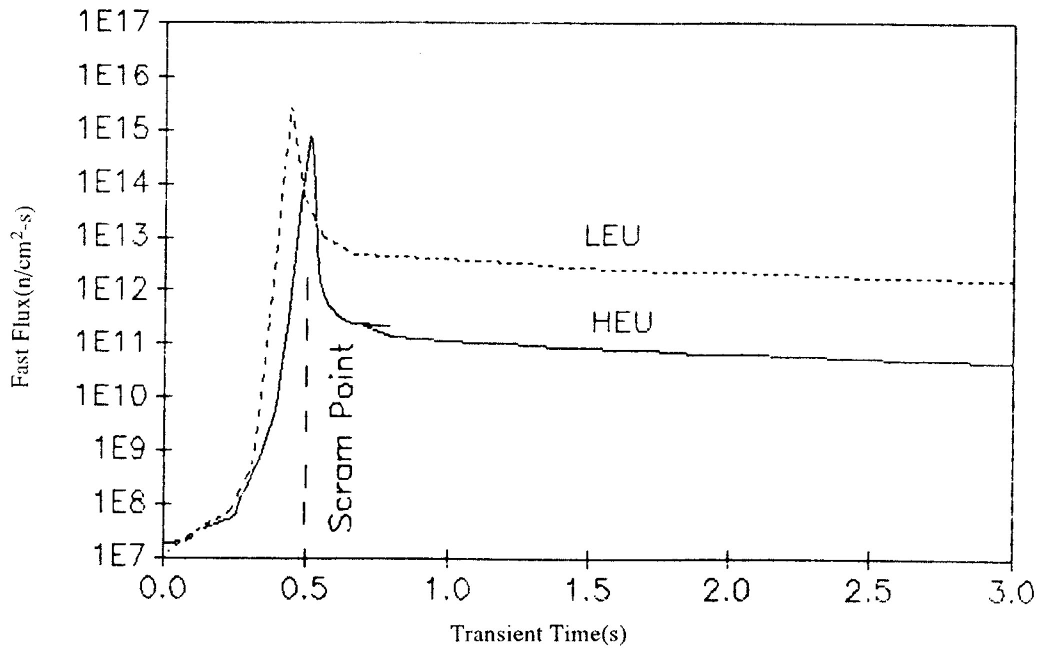

Figure 7a. Fast transients for HEU and LEU fuelled core_thermal flux distributions

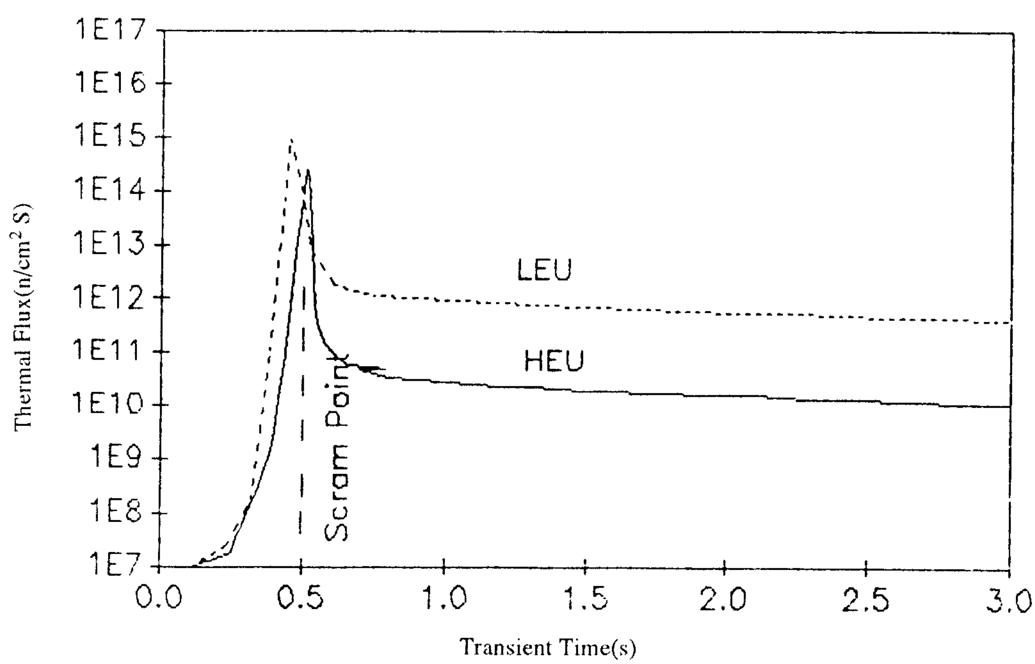

Figure 7b. Fast transients for HEU and LEU fuelled core fast flux distributions

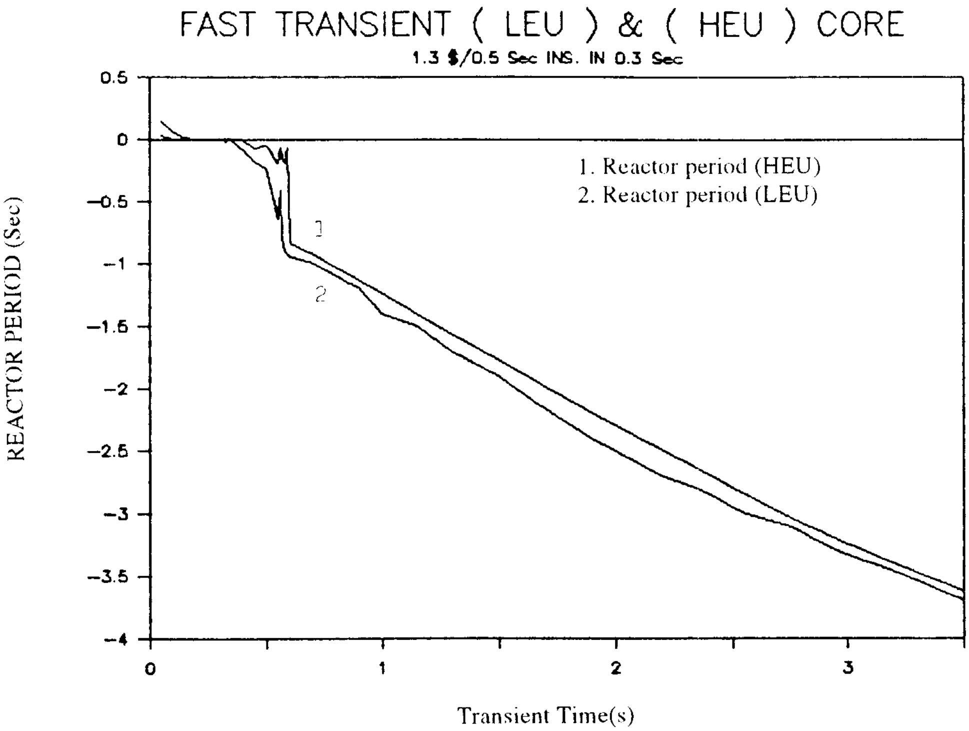

Figure 8. Reactor period in fast transient for HEU and LEU fuels

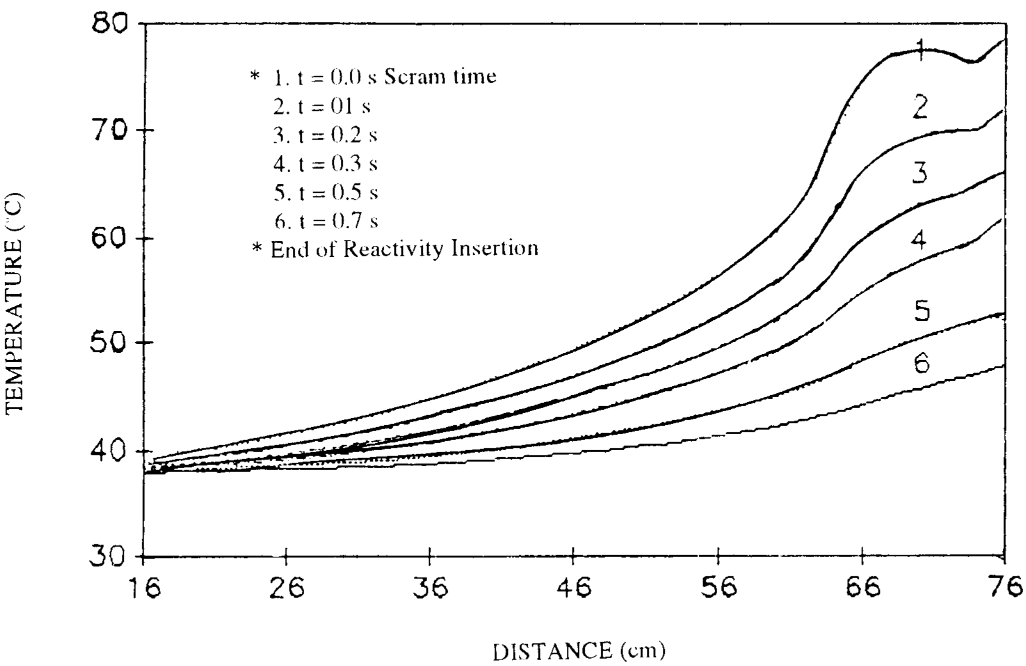

Figure 9. Temperature variations of fuel meat along the coolant channel during fast transient, HEU fuel

the solution of two group diffusion equations, see Figure 4. In order to assess the importance of delayed neutrons on reactor core kinetics the distribution of atomic density of six group precursors (delayed neutron emitters) were obtained by solving equations 1 and 2. These are shown in Figure 5.

In transient study, a series of slow and fast reactivity insertions were considered. The positive reactivity insertion leads to reactor power increase, which depends on the amount of reactivity and the rate of reactivity insertion. This leads to primarily higher temperatures in fuel meat and then spreads to cladding and the moderator with the end result of an adverse effect on reactor dynamics. This phenomenon reflects as the thermal feedback effect on temperature dependent group constants and consequently on core reactivity. The total temperature coefficient of reactivity was calculated by solving the transport equation using the WIMS-D/4 code. Then the coefficient was fed into the kinetic code COSTANZA and transients were studied. Figure 6 shows the slow and Figure 7 (a, b) the fast transient due to slow and fast reactivity insertion, for both HEU and LEU fuels. In the fast transient, two group fluxes increase sharply with time.

Normally, the reactor is designed such that at a certain power level (flux level) the reactor should automatically shut down. It is seen from Figure 6 that in the fast transient, especially for LEU fuel, the reactor power drops quite fast even before the scram set time for shut down margin is reached. This is mainly due to the Doppler broadening effect which is more effective in LEU fuel than in HEU fuel because of a higher U-238 concentration in LEU fuel. Figure 8 shows the reactor period in fast transient for both fuels, HEU and LEU, indicating the very small period at the inception of transient. It is seen that the period is shorter for LEU than for HEU fuel but drops faster than for HEU fuel and becomes zero .15s after reactivity insertion. The .3s period turns out to be negative at about 0.3s which indicates power declining. The cladding surface temperature and coolant average temperature may change as a result of abnormal operational conditions such as malfunction of water pump, water leakage or water pipe rupture, resulting in poor heat removal from the fuel elements and consequently a higher fuel temperature. A case of inefficiency of the secondary cooling system leading to a higher coolant inlet temperature - because of a lower rate of heat removal from the core - was studied and the effect on the reactivity coefficient was investigated.

The introduction of positive reactivity into the core, as a result of beam tube flooding or due to the insertion of a fissile or neutron producing reactive materials into the core during full power operation, are examples of low reactivity insertion of about 0.2% $\Delta k/k$. Insertion and removal of an experimental apparatus nearby core in the thermal flux peaking creates a reactivity impact on reactor operations whose degree of importance depends on the absorption rate and/or neutron production rate of the material inserted. Figure 9 shows the temperature variations along the coolant channel in fast transient for HEU fuel. One should bear in mind that in fuel conversion, there are some inevitable changes of core parameters as compared with those of the HEU core. The LEU core is loaded with 6 to 7 times more uranium in the meat thickness of about 0.5 to 0.7 mm. This brings about physical and thermodynamical property changes of the core. In order to have the same neutron flux as the HEU core, the core has to be more compact. This in turn creates new thermohydraulic problems. Long-standing irradiation of fuel with new compositions such as U_3O_8, USi, U_3Si_2 [10] and UO_2 may develop new metallurgical problems. Furthermore, conversion of a fraction of 80% U-238 fuel content to fissile Pu-239 nuclide, which also has positive temperature coefficients of reactivity, must be taken into consideration. Another metallurgical problem is possible brittleness of fuel as a result of fuel irradiation. At this time, it is premature to say that LEU fuel core operation is safer than HEU fuel core operation.

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