WHAT THE DIFFERENCE TO USE LEU AND HEU FUEL ELEMENTS SEPERATELY OR TOGETHER IN A RESEARCH REACTOR

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ABSTRACT

Concerning of nuclear material safety, most of the research reactors are advised to shift from HEU (high enriched-%93 U235) to LEU (low enriched-%20 U235) fuel elements. When LEU and HEU fuel elements are to be used together in a research reactor, some design and safety problems are encountered. According to use of the reactor, some research reactors such as MTR type may not show any considerable difference for HEU or LEU fuel elements, but the efficiency of radio-isotop production generated by thermal neutron interaction may decrease about twenty-thirty percent when LEU fuel elements are used. Here, fine mesh-sized 3D neutronic analysis of TR-2 research reactor is presented to indicate the arising problem when LEU end HEU fuel elements are used together in a research reactor. Partial thermohydraulic analysis of the reactor is also given to show the betterness of the LEU fuel element design. However, there might be some points that should be noticed for safer operation of plate type fuelled research reactors.

1. Introduction

Pool type TR-2 reactor was planned to be loaded with LEU and HEU fuel elements. Before the first loading of the reactor with only HEU, a detailed 3D neutronic analysis was done. But, this analysis has not been repeated for each new core configuration. Moreover, before loading the core with LEU and HEU fuel elements, a 3D neutronic analysis should have been done to check the operating safety margin accurately. After performing thin mesh size 3D and two group neutronic calculation of the present reactor core by the CITATION [1] code, it was found out that the intended design of the core has too large fuel plate power peaking factor and the peaking factor can be tremendously lowered by changing the loading configuration. Here, two different core designs are compared to emphasize how important the detailed neutronic analysis is when LEU and HEU fuel elements are used together in a core. Some calculated thermohydraulic parameters such as flow velocities in different channels are given to show the variation in calculating of the safety parameters.

2. Theory

Neutron fluxes and power distribution in the core were calculated with the CITATION code which is designed to solve problems using the finite difference representation of neutron diffusion theory. For a slab geometry, the multigroup diffusion theory with some simplification can be written as follows :

$$-D_{r,g}\nabla^2\phi_{r,g}+\left(\Sigma_{a,r,g}+\sum_{n=1}^{G}\Sigma_{s,r,g\to n}+D_{r,g}B_g^2\right)\phi_{r,g}=\sum_{n=1}^{G}\left(\Sigma_{s,r,n\to g}+\frac{\chi_g(\nu\Sigma)_{f,r,n}}{k_e}\right)\phi_{r,n}$$

Where, $D_{r,g}$: the diffusion coefficient (cm^2), ∇^2 : the Laplacian operator ( ), $\frac{\partial^2}{\partial x^2}+\frac{\partial^2}{\partial y^2}+\frac{\partial^2}{\partial z^2}$ $\Sigma_{a,r,g}$, $\Sigma_{s,r,g}$, $\Sigma_{f,r,g}$: the macroscopic absorption, scattering and fission cross sections for group-g (cm^{-2}), χ_g : source neutron fraction burn in group-g, k_e : the effective multiplication factor, ϕ_g : neutron flux for group-g (n/cm^2-s), B_g^2 : the buckling term to account for the effect of the Laplacian operator (cm^{-2}) and G is the total number of energy groups. Thermoyhdraulic calculation is performed by the NOKTA-RC [2] code. The thermohydraulic model is based on the one used in the NOKTA [3] code which solves mass, energy and momentum equations by using appropriate average values of enthalpy, pressure and velocity in a computaion cell as done in the COBRA-I∨-I [4] code.

3. Results and conclusions

After computing two group macroscopic x-sections [5] for two different parts (side plate and homogenized fuel region) of LEU and HEU fuel elements, 3D two group fluxes and power density in thin computation cells were determined with using the CITATION code for two different core configurations shown in Fig. 1. Because an LEU fuel element has much more U-235 mass (416 gr) than an HEU one (280.6gr), LUE fuel elements are more critical than HEU ones for the power density which limits the operating power level of the reactor. It might be thought that the lowest neutron fluxes should be at the outest region of the core. But, a core designer must also concern of fuel plate power peaking factor (the hot channel for thermohydraulic analysis). However, this was missed in the intended core design (the first one given in Fig. 1) of TR-2 research reactor. After checking different core configurations by the CITATION code, a better fuel element arrangement (the second one given in Fig. 1) was obtained considerably to lower the hot channel power peaking factor from 2.25 to 1.55. In the first configuration, LEU fuel elements are placed at the outest parts

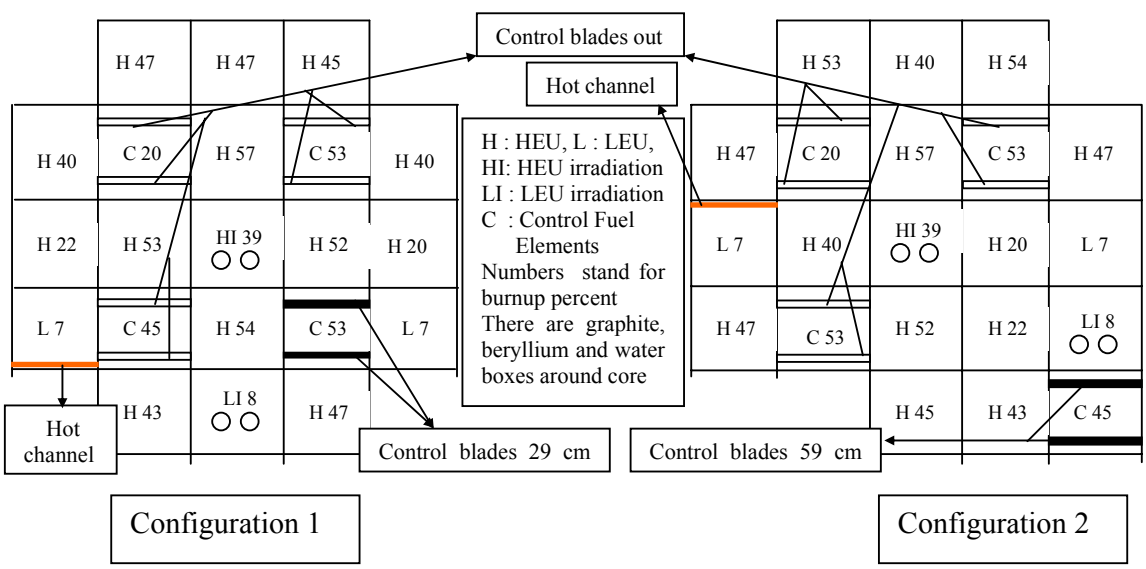

Fig. 1. Two different core configurations for TR-2 research reactor loading

of the core where the thermal flux is larger and the orientation of the fuel plates is parallel to thermalization region (water region). Although the LEU fuel elements are again located at the outside region of the core, the orientation of the fuel plates is prevented to be parallel to the thermalization region as shown in the second configuration. Thus, the hot channel power peaking factor was tremendously decreased as seeing from Figs. 2 and 3. In comparing the core loadings for the hot channel factor, the second one is 45 percent better than the first one. In Fig. 2, the axial average homogen power density profiles are given for the hot channels, the hot fuel elements and the control fuel elements. This figure also shows the effect of the control plates on power shape in the core. In Fig. 3, the homogen power distributions of the loadings given in Fig.1 are compared for the hot fuel elements at axial levels, z = 32.5 and 30.5 cm (respectively). As shown in the figure, the power density gradient sharply increases to the outward (next to thermalization region) of the fuel element in Conf. 1. For the configurations, the homogen power density profiles of the core at axial

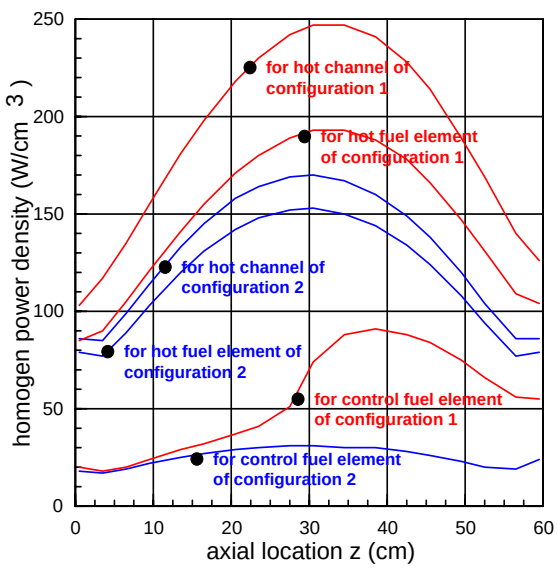

Fig. 2. Axial average homogen power density profiles for the hot channels, the hot and the control fuel elements.

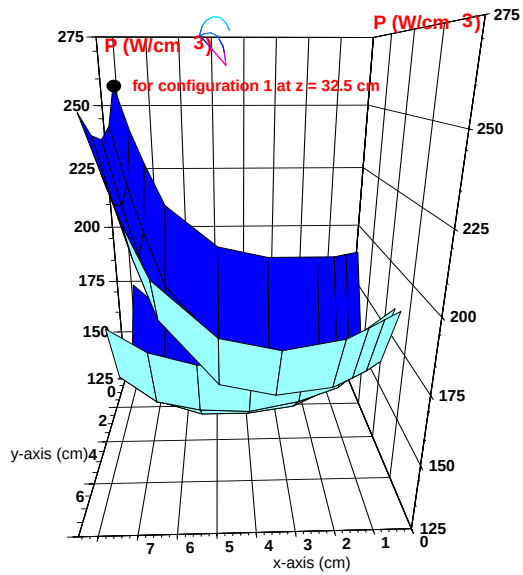

Fig. 3. Homogen power distributions for the hot fuel elements at z = 32.5 cm and 30.5 cm.

levels (maximum power density planes, z = 32.5 and 30.5 cm) are given in Figs. 4 and 5. In these figures, the effect of the the LEU fuel elements placement (core design) on the hot channel factor and the power distribution can be clearly seen. These differences between the loadings are very significiant and so important when using LEU and HEU fuel elements together in a research reactor. Another important aim of the research reactor is to have as high as possiple thermal neutron flux in the irradiation region for radio-isotope production. In Fig. 6, the axial average fast and thermal neutron flux distributions at the HEU fuel element irradiation region (HI 39 in Fig. 1) are given for the loadings. Both thermal and fast neutron fluxes are considerably higher for configuration 2 at the upper region of the core because of the control rod effect. In respect of thermalhydraulic design of the fuel elements, LEUs are better than HEUs because the thickness of the outest fuel plates of LEU is 0.75 mm thinner than those of the HEU which means that the cooling of these LEU fuel plates is considerably better. Flow velocities for 1.0, 1.32 and 2.1 mm gaps were calculated as 127, 153 and 214 cm/s respectively. As conclusion, even though for research reactor, detailed 3D

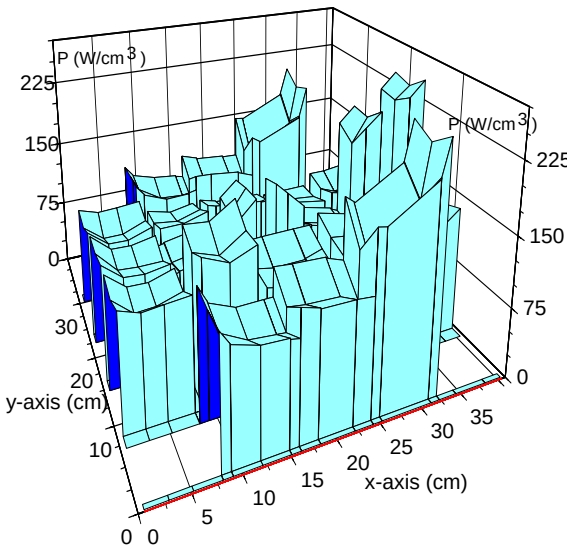

Fig. 4. The homogen power distribution at z = 32.5 cm for configuration 1.

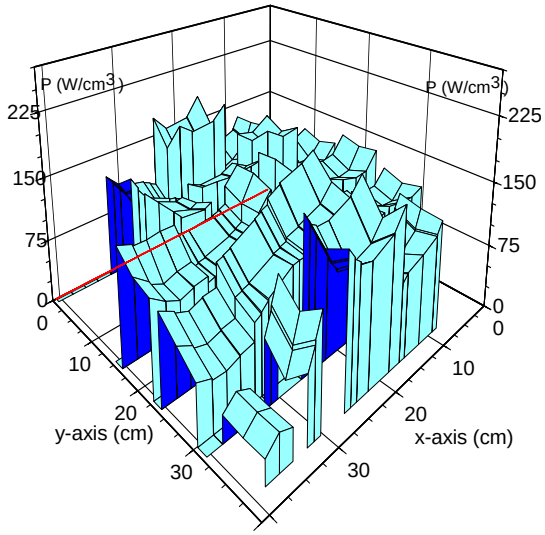

Fig. 5. The homogen power distribution at z = 30.5 cm for configuration 2.

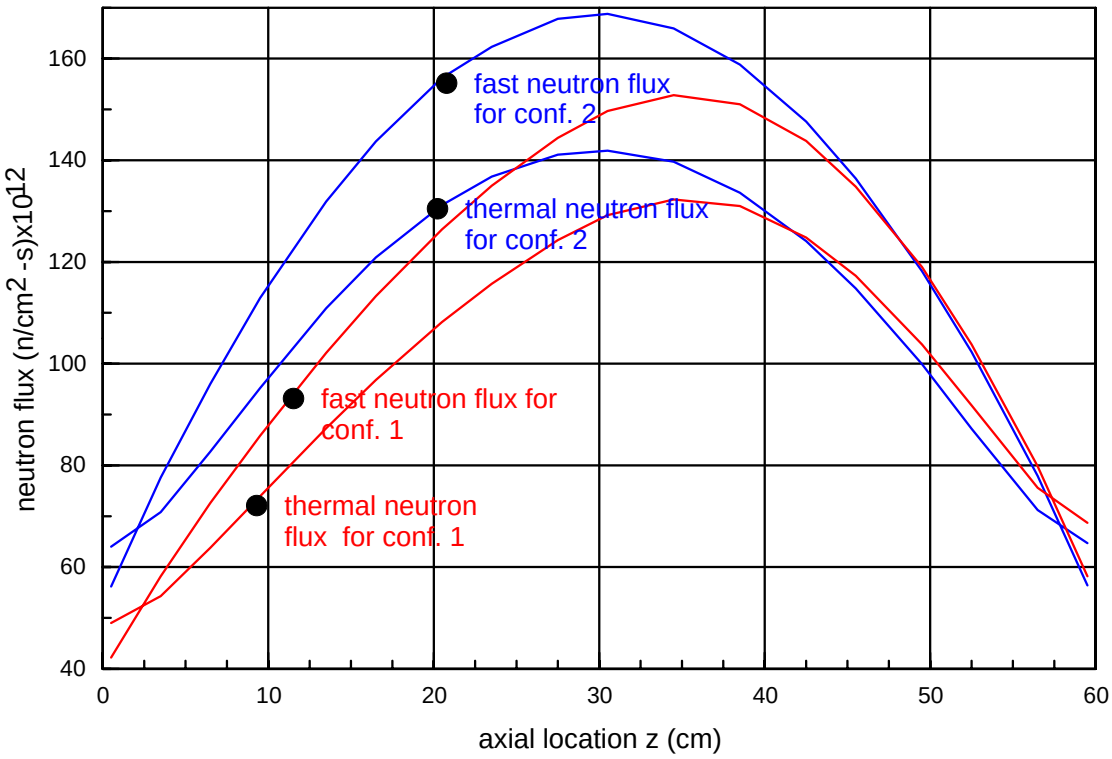

Fig. 6. Average thermal and fast neutron flux profiles at HEU irradiation region (HI 39 in Fig.1) for the core configurations.

neutronic analysis should be always done before loading the reactor or making considerable change in core design or control rod positions. Further, the analysis becomes more necessary when using LEU and HEU fuel elements together in a research reactor core. If LEU (416 gr U-35) and HEU (280.6 gr U-35) fuel elements are to be used together in a core loading, LEU fuel elements have to be placed somewhere in the core that the thermal neutron flux is lower and the orientation of the fuel plates of the element must not be parallel to thermalization region (water region) as shown in Fig. 1. In general, the most critical fuel plate for the power peaking is the one next to irradiation region. In the present design of the irradiation fuel element, the flow gap next to the irradiation region, 1.32 mm, is considerably narrower than other flow gap of 2.1 mm. Therefore, the design of the irradiation fuel elements should be reviewed for better cooling.

References

[1] T. B. FOWLER, D. R. VONDY and G. W. CUNNINGHAM, "Nuclear Reactor Core Analysis Code," Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1969.

[2] Ş. KAYA and H. YAVUZ, "NOKTA-RC: Nodal Kinetic and Thermohydraulic Analysis code for Research and Commercial nuclear reactors," not published.

[3] Ş. KAYA and H. YAVUZ, "A Nodal Kinetics and Thermohydraulics Analysis (NOKTA) Code for Analyzing Rod-Ejection Accidents and Other Transients in Nuclear Reactor Cores," *Nucl. Technol.,* **129**, 26, 2000.

[4] C. L. WHEELER, C. W. STEWART, R. J. CENA, D. S. ROWE and A. M. SUTEY, "COBRA-IV-I: An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores," Pacific Northwest Laboratories, 1976

[5] M. H. TURGUT and T. TÜRKER, "Core Conversion Calculations for the TR-2 Reactor," Çekmece Nuclear Research and Training Center, ÇNAEM-AR-308, 1993.