# SAFETY ANALYSIS OF NEUTRON FLUX OPTIMIZATION IN IRRADIATION CHANNELS AT THE NUR RESEARCH REACTOR

by

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**Prior to core reloading, planned power upgrading, or as a part of required analyses of past events, accurate safety evaluations should be carried out. Generally speaking, the content of a safety report has to be modified whenever a new type or design of fuel is to be used in a reactor core. As the existing plants have well established licensing procedures, including well founded analysis methods, the application of new analysis methods has to be thoroughly evaluated, with specific emphasis on their capability of producing results beneficial to reactor operation. The detailed study presented here was carried out so as to insure that the allowed operational safety limits of the NUR research reactor are not exceeded under any circumstances.**

*Key words: safety analysis, flux optimization, MTR nuclear research rector*

## INTRODUCTION

The capacity of a research reactor for utilization in terms of radioisotope production, materials testing, neutron transmutation and neutron diffraction is directly related to the magnitude of the neutron flux and to the nature of the neutron spectra present at the irradiation sites. Hence, the optimization of neutron fluxes and spectra in experimental channels is of great concern in research reactor utilization. A general safety analysis approach used at the NUR research reactor prior to neutron flux optimization in irradiation channels is presented in this paper. The approach is, essentially, based upon a judicious optimization of core configuration, combined with the improvement of reflector characteristics. In order to allow the implementation of a new core configuration into the operation scheme of the reactor, a demonstration that such a configuration is safe and in conformity with the safety standards in application at the NUR reactor [1], as well as the IAEA standard safety guide recommendation [2], is required. A standardized safety analyses for research reactors was proposed by the IAEA [3] concerning core conversion from the use of highly enriched uranium fuel to the use of low-enriched uranium fuel. Within this framework, a detailed study, including static and dynamic calculations using advanced computational tools, has been carried out.

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## DESCRIPTION OF CORE CONFIGURATIONS

The research nuclear reactor considered herein is a 1 $MW_t$, open pool, MTR-LEU fuel type reactor. The reactor core is surrounded by graphite reflector blocks and water. The latter serves as a coolant, moderator and reflector. The reactor is equipped with several horizontal and vertical irradiation channels. The reactivity control system of the reactor is made of five absorbing rods in Ag-In-Cd: four control and safety rods (CR1, CR2, CR3, and CR4) and one fine regulating rod (F). The reactor was first brought to criticality on March 23, 1989. Figure 1 shows the old configuration (Configuration IV-N), while fig. 2 shows the newly proposed configuration (Configuration X-1) for the NUR reactor core. In comparison

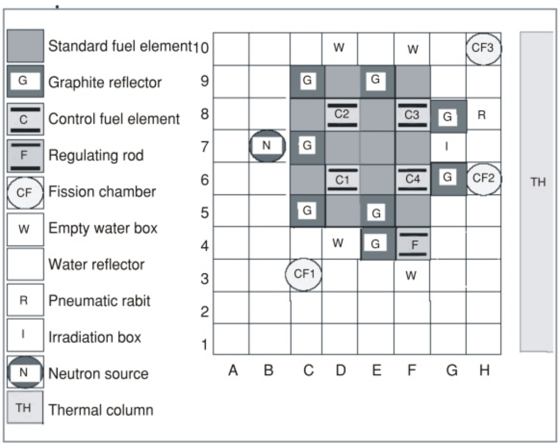


**Figure 1. Core configuration IV-N (NC = 16)**

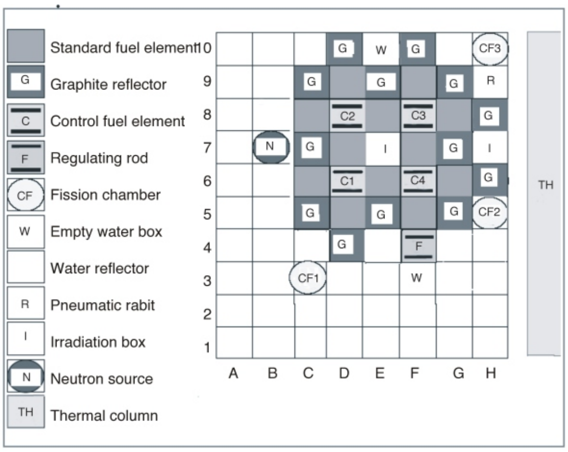


**Figure 2. Core configuration X-1 (NC = 17)**

with the old configuration, the new optimized one is characterized by:

- the presence of a central neutron trap (position E-7) with an active volume of 7 8 60 cm,
- the transfer of the fuel element, initially at position E-7, to position G-9,
- the adjunction of a new (fresh) fuel element at position G-6, and
- the increase in the number of graphite reflector blocks from 8 to 15 elements.

## PROBLEM MODELLING

Axial temperature distributions and maximum allowable heat fluxes were determined in two types of channels (average and hot), using the TERMIC code [4]. TERMIC is a program that can perform thermal-hydraulic calculations of nuclear reactor cores in pressure and temperature ranges typical of MTR reactors and has been used in the thermal hydraulic design of Argentinean reactors. On the other hand, reactivity insertion and loss of flow transients' computations were essentially performed by using the PARET code [5]. Thermal-hydraulic simulations performed by the PARET code are based on a one-dimensional thermal-hydraulic model coupled with the point reactor kinetic model. In line with the core model assumed by PARET, the core is subdivided over its radial section into numbered regions, where each one represents a single coolant channel and an associated fuel plate. For purposes of reactivity feedback calculations, provisions for weighing these regions were made, as well.

The core was modelled into two regions, *i. e.* assuming two parallel cooling channels and their associated fuel plates. The first channel represents the hottest channel in the core, the second one the remainder of the core itself. Previous simulation results confirmed that such modelling is sufficient for a good prediction of core response [3]. As for the axial direction, the two regions were subdivided into twenty sections. Hot channel factors were determined by neutron calculations using WIMS and CITVAP codes [6]. For feedback calculations, as outlined in tab. 1, uniform radial and axial weighing factors were assumed. Once the PARET input settled, an assessment of the code's response to steady state condition was carried out by confronting the calculated parameters with the measured experimental data. The objective is to show that the developed NUR core model provides acceptable results at a steady state level and can therefore offer a valid approach to transient analysis. For this purpose, the value of $\Delta T = T_{out} - T_{in}$, where $T_{out}$ is the core outlet temperature and $T_{in}$ is the core inlet temperature obtained by the PARET code (5 °C), is compared to the measured temperature (3.9 °C). The result

**Table 1. Core kinetic data**

| Parameter | Unit | Configuration IV-N | Configuration X-1 |
|---|---|---|---|
| Prompt neutron generation time ($\Lambda$) | s | 69.0 | 65.46 |
| Delayed neutron fraction ($\beta$) | pcm[(a)] | 822 | 800 |
| Radial peaking factor ($F_r$) | – | 1.456 | 2.278 |
| Axial peaking factor ($F_a$) | – | 1.388 | 1.415 |
| Temperature coefficient<br>– Doppler<br>– Moderator | <br><br>$/°C[(b)]<br>$/°C | <br><br>–2.54 $10^{-3}$<br>–1.3 $10^{-2}$ | <br><br>–2.54 $10^{-3}$<br>–2.3 $10^{-2}$ |
| Void coefficient | $/% | 0.342 | 0.342 |

(a) 1 pcm = $10^{-5}$
(b) 1 $ is the reactivity that will make a reactor prompt critical

proved to be acceptable, since the deviation is fully covered by measurement uncertainties.

The loss of shutdown heat removal accident analysis was covered by means of the LODEHR code [7], a newly developed, transient thermal-hydraulic model for predicting MTR research reactor core behaviour under loss of shutdown heat removal. The said model is well suited for fuel plate surface temperature estimation in situations where the core remains partially immersed in stagnant water, as is the case with the loss of coolant because of damage to the experimental beam tube in a MTR pool- type reactor.

A series of thermal constraints have to be observed in order to insure the safe operation of the NUR reactor. The safety of operation of the NUR reactor imposes, among other things, certain thermal limitations on the fuel cladding surface temperature, $T_w$. These limitations are:

– at normal operation, the maximum value reached by $T_w$ in the hottest channel should not exceed 90 °C; this constraint is imposed in order to minimize the rate of surface oxidation of the aluminium cladding material,

– in the case of transients, the maximum value reached by $T_w$ in the hottest channel should not exceed 120 °C; this constraint is imposed in order to avoid coolant nucleate boiling at clad surface. Flow excursion phenomenon could take place under such conditions [8], and

– in the case of severe accidents, the maximum value reached by $T_w$ should remain below 600 °C; this constraint is imposed in order to avoid the melting of the aluminium cladding of the fuel plates and consequent liberation of fission products.

The hottest channel in Configuration X-1 was determined through neutronic calculations and found to lie, when the operation of the reactor is normal, within the fuel element E-8 and characterized by an overall power peaking factor of 3.0.

At normal operation, temperature profiles within the channel were determined using the THERMIC-1H code. The results of these calculations indicate that the maximum temperature in the hottest channel stays well bellow 90 °C for a reactor operating at the most severe regime, one characterized by a thermal power level equal to 1.2 MW (*i. e.* 120% $P_n$; $P_n$ is the reactor nominal power) and an effective core coolant flow rate of 154 $m^3$ per hour.

## TRANSIENT ANALYSIS

The following trip points were considered in our transient analysis. The emergency shutdown of the NUR reactor is triggered by one of the following trip signals:

– reactor power $>1.2\ P_n$; ($P_n$ = 1 MW),
– primary coolant flow rate $<0.8\ Q_n$; ($Q_n$ = 220 $m^3$ per hour),
– reactor period <20 s,
– pool water height <9 m, and
– the natural convection valve opens when the coolant flow rate $<0.2\ Q_n$.

For all the transients, a 0.25 s delay time between the onset of the trip condition and the beginning of the effective insertion of the safety rods into the core was considered. For the computation of the decay in the coolant flow rate during a primary pump failure accident, the experimentally determined value of 2.2 s was taken as the time constant.

## IAEA suggested transient cases

As suggested by the AIEA safety guide report [2], the following cases were considered.

### *Fast and slow loss of coolant flow (FLOFA and SLOFA)*

Flow decay is governed by an exponential decrease law (exp(–t/*T*) with *T* = 1 s). As shown in fig. 3 and 4, we notice that the values of coolant and clad temperatures in the hot channel are higher in Configuration X-1. However, in comparison to the old core configuration (Configuration IV-N), a stable natural convection regime is reached more quickly.

In the case of the slow loss of flow, the decay constant is equal to *T*=25 s. Figures 5 and 6 show that the same phenomenon occurs in the case of fast LOFA. Natural convection takes place after 2 *T* = 50 s and, for Configuration X-1 fuel elements, the temperatures stabilize with a high value.

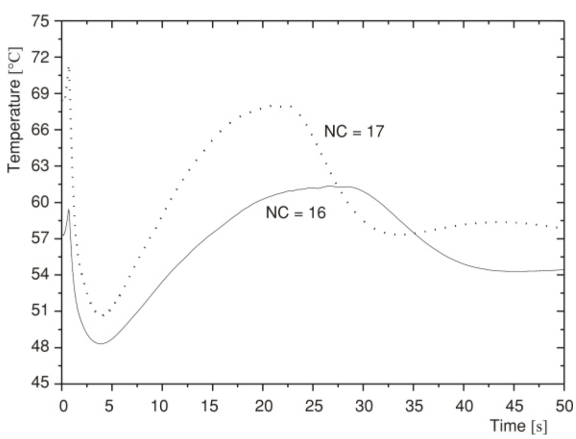


**Figure 3. Clad temperature response for a fast LOFA**

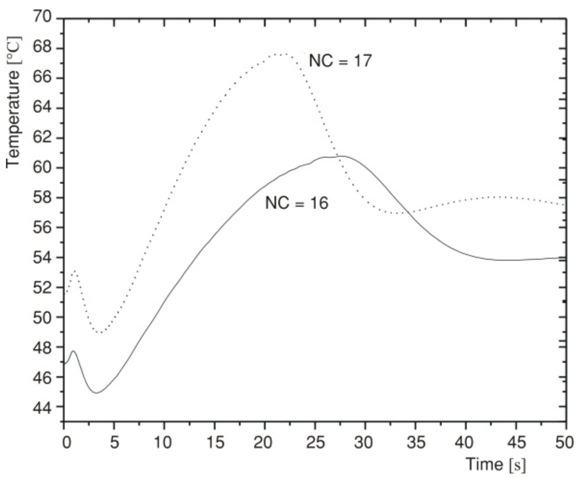


**Figure 4. Coolant temperature response for a fast LOFA**

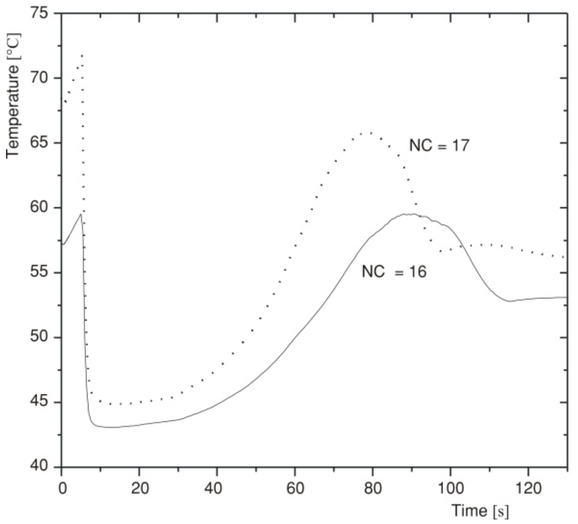


**Figure 5. Clad temperature response for a slow LOFA**

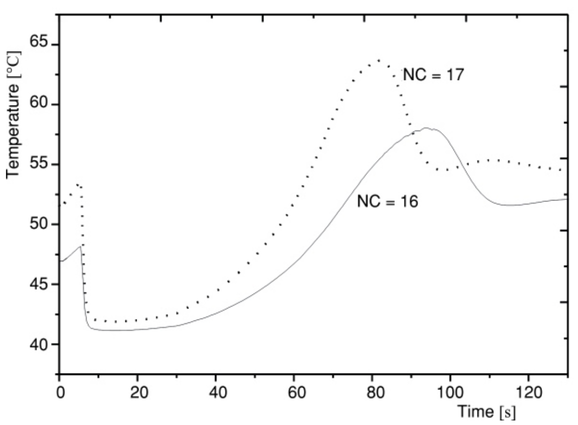


**Figure 6. Coolant temperature response for a slow LOFA**

*Fast and slow transient reactivity insertion*

A reactivity insertion of 1.5 $ within 0.5 s is considered to be positive when the reactor core is in a critical state with an initial power of 1 W. As shown in figs. 7 and 8, the peak power value is reached more or less in the same manner for the two configurations. However, the temperatures of the cladding and the coolant in the hot channel are higher in the case of Configuration X-1 elements configuration, still remaining far from their respective safety critical values (see figs.9 and 10.).

In the case of slow reactivity insertion, the insertion rate is 1.5 $ within 10 s. As above mentioned, the peak of power is reached more or less in

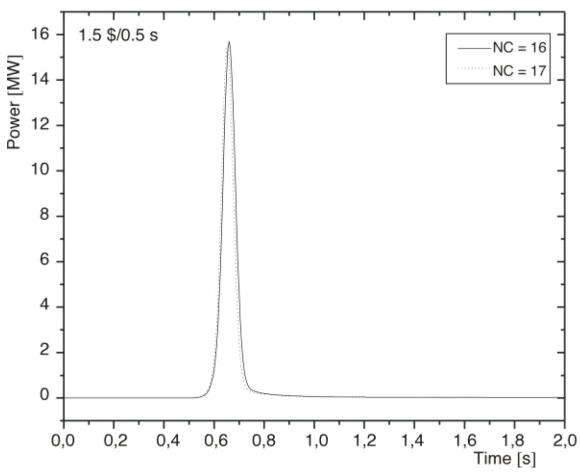


**Figure 7. Power reactor response for a fast ramp of positive reactivity insertion**

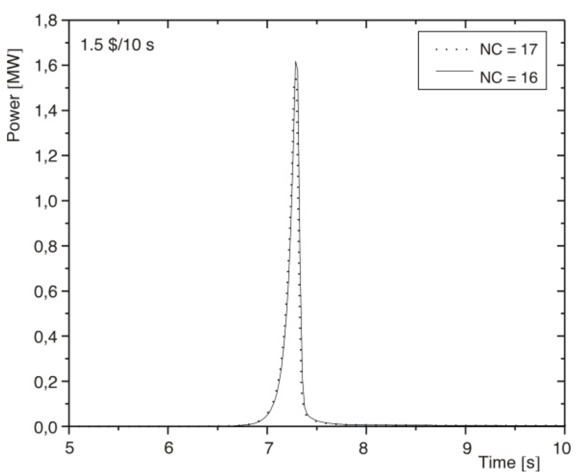


**Figure 8. Power reactor response for a slow ramp of positive reactivity insertion**

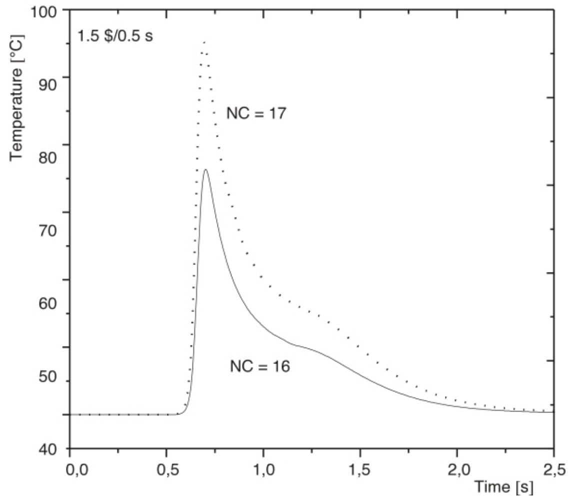


**Figure 9. Clad temperature response for a fast ramp of positive reactivity insertion**

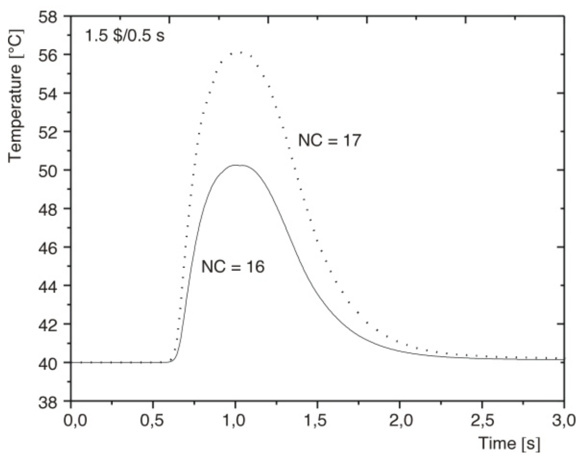


**Figure 10. Coolant temperature response for a fast ramp of positive reactivity insertion**

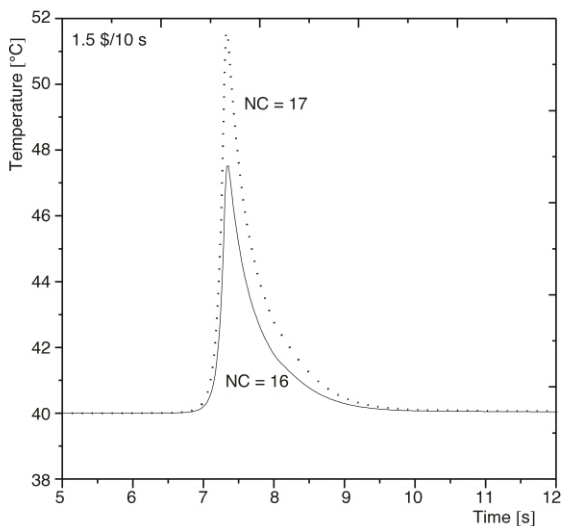


**Figure 11. Clad temperature response for a slow ramp of positive reactivity insertion**

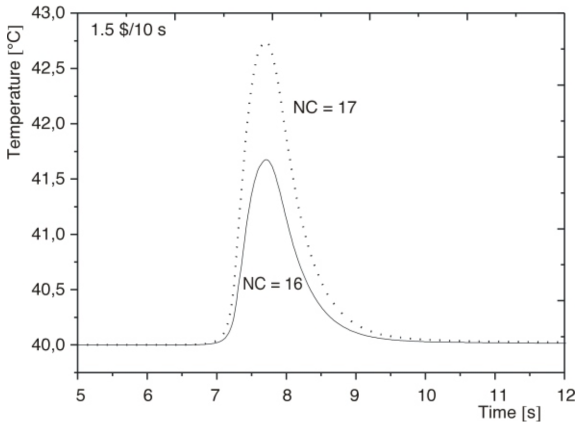


**Figure 12. Coolant temperature response for a slow ramp of positive reactivity insertion**

the same manner for the two configurations. However, the temperatures of the hot channel clad and coolant are higher in the case of Configuration X-1. (figs. 11 and 12).

## Loss of coolant accident

The case of a hypothetical accidental break in an experimental horizontal beam tube that would result in loss of water from the reactor pool and lead to a direct exposure of a large portion of the core to air was parametrically studied using the in-house LODHER code [7]. It was found that the most severe situation occurs when the height of water remaining in the pool is just enough to hinder natural air circulation through core fuel elements. In that case, the maximum temperature reached by the aluminum cladding of the fuel

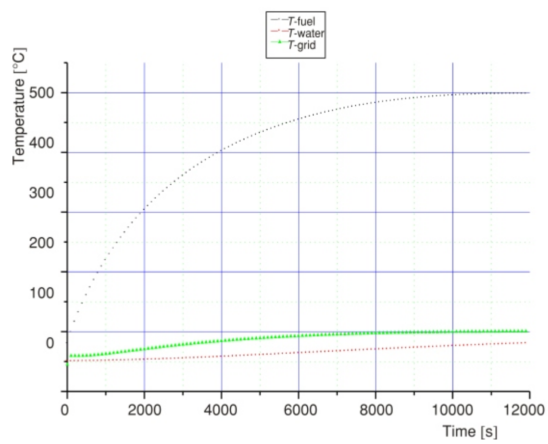


**Figure 13. Loss of decay heat removal accident: core temperatures evolution**
*(core immersion height – 5 cm, coolant leakage time – 250 s, core operation history 1 MW during 2 days, Configuration IV-N)*

plates stays below 500 °C (fig. 13), hence well below the melting point of aluminum. As the level of core immersion increases, inter-plate water temperature increases till it reaches saturation conditions. The circulation of saturated steam between fuel plates will induce a large decrease in plate temperature. Consequently, water temperature inside the inter-plate gap decreases and boiling stops. In practice, we will observe an unstable regime, where the boiling process moves from one channel to another. As shown in fig. 14, it was found that in this regime the fuel surface temperature remains below 170 °C.

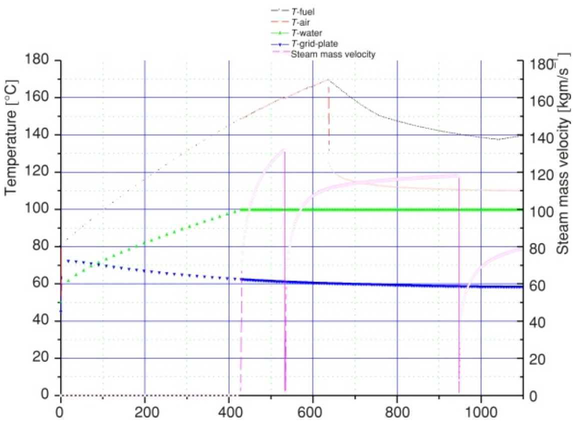


**Figure 14. Loss of decay heat removal accident: core temperatures evolution and steam mass velocity profiles**
*(core immersion height – 5 cm, coolant leakage time – 250 s, core operation history 1 MW during 2 days, Configuration IV-N)*

## CONCLUSION

The analysis performed shows that the more symmetrical nature of core geometry leads to a more adequately balanced reactivity control system and contributes quite efficiently to the operational safety of the NUR reactor.

The extensive numerical modelling and variety of measurements and experimental tests that have been performed were also of paramount importance for the introduction of a series of improvements concerning analytical tools and methods commonly used in research reactor core analysis.

According to these calculations, as predicted, temperature limits were not exceeded under severe accidental situations; neither in fast loss of flow type transients nor in fast positive reactivity insertion type transients.

The application of these improved methods will enable us to achieve a more realistic safety margin and consequently increase the lifetime and commercial productivity of research reactors.

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## СИГУРНОСНА АНАЛИЗА ОПТИМИЗАЦИЈЕ НЕУТРОНСКОГ ФЛУКСА У КАНАЛУ ЗА ОЗРАЧИВАЊЕ ИСТРАЖИВАЧКОГ РЕАКТОРА NUR

Пре обављања замене горива, планираног увећања снаге, или као део обавезних анализа догађаја који су се збили, морају се извршити поуздане сигурносне процене. Најопштије говорећи, садржај сигурносног извештаја мора да се измени кадгод се нова врста или облик горива употреби у реакторском језгру. Како постојеће електране имају добро утврђене поступке лиценцирања са ваљано заснованим аналитичким методама, то примена иновативних аналитичких метода мора бити пажљиво оцењена, са посебним нагласком на могућност постизања резултата који су на општу корист реакторског рада. У овим оквирима обављено је детаљно истраживање у намери да се обезбеди да допуштене сигурносне границе реактора у раду нису прекорачене нити у једном случају.

*Кључне речи: сигурносна анализа, оптимизација флукса, MTR нуклеарни истраживачки реактор*