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Criticality Benchmark Calculation and Comparison for a Fuel Storage Rack in a TRIGA Reactor

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Abstract

In this work, computational models of criticality simulation were developed and executed for the TRIGA IPR-R1 nuclear reactor, present in CDTN, according with the specifications and implementations of a fuel storage rack. Computer simulations were performed using the MCNP6 tool and the results were compared with a similar benchmark, developed for the TRIGA MARK I.

1. Introduction

As stated in [5], storage of irradiated nuclear fuel in water pools has been standard practice since the primordial of nuclear reactor operations. Pool storage is the starting point for all other fuel storage candidate processes and is a candidate for extended interim fuel storage until policy questions regarding reprocessing and ultimate disposal have been resolved.

Moreover, several works about nuclear fuel storage processes have highlighted the utility of the usage of nuclear spent fuels on criticality control and computational criticality models for fuel storage were developed over the different aspects of transport theory models [4].

Extending the importance of the calculations done in this work, a fuel storage rack, intended to accommodate fuel-moderator elements or graphite dummy was fabricated for the TRIGA IPR-R1 nuclear research reactor in CDTN [3]. This storage system, is supposed not only to temporarily host nuclear spent fuel in order to offer an viable trustworthy storage slot that can be readily available, but also to provide new irradiation channels, in addition to the already implemented ones.

Increasing the amount of spots where irradiation targets can be placed is desired not only because this optimizes the usage efficiency of the reactor, but also because it allows users to have more control over incident dosage, as radial geometrical position and distance over the main core greatly interferes in the neutron and photon population reaching the goal.

The computational simulation was done using the MCNP6 tool, which is based in the application of the Monte Carlo Method [1] in accordance with the Neutron Transport Theory. Through this approach, MCNP6 has a consistent statistical method that aims to the estimation of the parameters needed for safety control and reactor dynamics.

The main safety parameter in this work is the k_{eff}, known as criticality coefficient, since it represents the ration between different generations of neutrons histories, and is supposed to be less than 1 (subcritical) for security purposes.

Beyond this, the MCNP6 model created in this work is mainly based in the one which was used for the commissioning of TRIGA IPR-R1 and for validation purposes, the results reached here were compared with a similar calculation done by the Department of Nuclear Engineering of the Oregon State University for the TRIGA MARK I reactor [8].

This is due to the requisitions written in [7], which states that the computational method shall be evaluated, in order validate and to verify that the study is thorough and uses benchmark critical experiments that are similar to the normal-conditions and credible-abnormal conditions models. By doing this, is expected for this study to be the base for future experimental measurements in the storage rack.

2. Methodology

According to [8], a viable method of criticality evaluation consists in measuring the neutron flux of the subcritical system at increasing levels of reactivity insertion.

This is because the dynamics of the TRIGA nuclear reactor implies in no abrupt change in neutron flux abroad the pool while k_{eff} increases, while the extrapolated data over the state of criticality tends to be conservative in matters of security.

In the efforts to simulate the behaviour of the criticality coefficient for the TRIGA reactor, several computational models were developed using MCNP6. Those models, represents different steps of increase in the value of k_{eff}, where progressive quantities of nuclear fuel were inserted inside the storage rack.

The concept of the subcritical multiplication factor (M), is defined in [2] as the ratio between the neutron flux (ϕ) and the initial neutron flux (ϕ_0):

$$M = \frac{\phi}{\phi_0} \tag{1}$$

This relation is important in this approach, because [2] states that the coefficient M can be related to the coefficient of reactivity by the Equation 2 in systems where the neutron source is uniform.

$$k_{eff} = 1 - \frac{1}{M} \tag{2}$$

The goal here, is that when the reactor approaches to the condition of criticality, the value of M increases to the point where $\frac{1}{M}$ tends to zero. Therefore, future experimental measurements won't risk to be made in supercritical conditions.

2.1. The Storage Rack

The fuel storage rack is described in [3], as consisted of two 6.2 mm thick aluminum plates, 6.7 cm wide by 30.5 cm long. The upper plate has six 4.1 cm diameter holes, while the lower plate has six 2.54 cm diameter holes. These plates are welded to four aluminum angles 61 cm long, constituting the side of the rack.

Also, the rack is supported by two 9.5 mm diameter by 2.5 m long aluminum rods, which are bolted through an aluminum bracket to the aluminum channel at the top of the tank. The installation of the rack, stays under approximately 2.1 meters of water for purposes of shielding.

The Figure 1 is a representative picture of the storage rack, to be coupled in the aluminum rods.

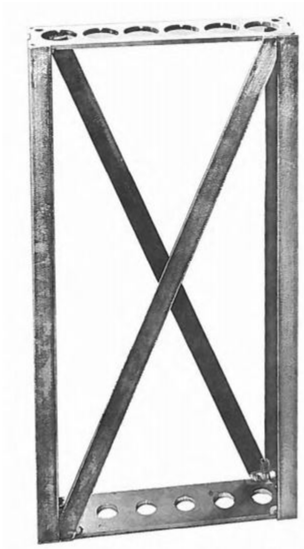

Figure 1: Picture of the fuel storage rack

2.2. Fuel Element

The source term consists of 63 Uraniun based fuel elements, self moderated by Zirconium Hydride.

For influence of the asymmetry of the core, plus the radial distance and positioning changes over the years, each fuel ring around the central tube presents a different distribution in enriched Uranium concentrations.

Also, the non homogeneous burnup through the core along the years of operation resulted in different rates of poisoning around the fuel elements.

Adopting conservative purposes, this work considers for each ring, the fuel element that has the highest concentration of U-235, as showed in Table I.

Table I: Fuel element composition

Fuel by ring	Mass of HZr (g)	Mass of Uranium (g)	Mass of U-235 (g)
B	2252.84	195.1	38.65
C	2299	193	38
D	2250.17	188.34	37.31
E	2256.61	186.62	36.97
F	2281.06	190.47	37.73

2.3. Temperature effects

The working temperature along the rings in the source term is important because of the doopler broadening effect.

Summarily, the basic physical processes that occur when the fuel elements are heated are: increasing the temperature of the Zirconium Hydride increases the probability of thermal neutrons in the fuel to gain energy from the hydrogen atoms. As neutrons gain energy from the Hydride, their average free path increases considerably, increasing their likelihood of fuel leakage.

Therefore, heating the U-ZrH alloy causes a greater hardening of the neutron spectrum in fuel than in water. This effect reduces the likelihood of neutron absorption reactions in the fuel, resulting in a loss of reactivity.

The process to adapt ENDF libraries to the characteristic temperatures of the rings in order to be used in MCNP6 are described in [9]. The respective temperatures are showed in Table II

Table II: Average temperature by ring

Ring	Average Temperature (K)
B	493
C	473
D	453
E	433
F	413

2.4. MCNP6 model

The MCNP6 tally used for computing neutron flux over the reactor is the card F4 [1]. This card, quantifies the neutron flux passing through the respective cell averaged along all the active cycles the simulation takes, as described in Equation 3

$$F4 = \frac{1}{V}\int dV \int dE \int d\Omega \phi(r, E, \Omega) \tag{3}$$

The Neutron flux, was calculated in three separated spots, where they can be measured in future: The central tube, the measurement table and in the vicinity of the storage rack.

3. Normalization

According with [10], the neutron fluxes and reaction rates in MCNP output tallies are normalized per source neutron. In order to achieve the real value of number of neutrons passing through the tally, we need to normalize the number of particles by the power of the reactor, as stated in the Equation 4.

$$\phi = \frac{P\nu}{w_f k_{eff}} \phi_{MC} \tag{4}$$

In which:

- ϕ_{MC} is the flux calculated by Monte Carlo method
- P is the nominal power of the reactor (100 kW)

- ν is the average number of neutrons released by fission
- w_f is the power per fission rate

4. Results

The model of the reactor with the storage rack was implemented using MCNP6, using the worst possible scenario, where the rack is filled with six of the B ring fuel elements.

The values of k_{eff}, neutron flux in different parts of the reactor and dosage on top of the pool were calculated and comparated with the reactor absent of storage rack, as shown in Table III

Table III: Main parameters of the simulations

	Rack filled	Rack empty
k_{eff}	0.99217 ± 23pcm	0.99248 ± 22pcm
Flux in central tube n/m^3 s	1.552e+13	1.5429e+13
Flux in extractor	7.8052e+05	7.8028e+05
Vicinity of the Rack	0	0
Dosage on top of pool	0	0

Also, another set of simulations was performed to predict the reactivity coefficient of the filled rack alone in the pool. This was done through the performing of three MCNP6 simulations.

The first case includes the rack alone inserted in the pool, with no reactor core. The second and third cases represent a simplification of the reactor's geometry, in which a cylindrical pool spaced respectively 20 cm and 25 cm from the storage rack's edges was inserted. The goal here is to reduce the characteristic standard deviation if compared to the first case, due to the difference in the pool size and neutron distribution. The numerical results of k_{eff} are shown in table IV

Table IV: Reactivity coefficient in five different configurations

Pool Configuration	k_{eff}	σ (pcm)
Rack + real size pool	0.31135	235
Rack + 20cm spaced cylindrical pool	0.39731	12
Rack + 25cm spaced cylindrical pool	0.39729	12

5. Discussion

The results presented in section 4 show a small variation in the main parameter of the reactor, as all the parameters evaluated variate only within the uncertainty.

The two last items of Table III (neutron flux in the rack and dosage on top) were not satisfactorily estimated by MCNP6, since the size of the neutron sample used in this simulation wasn't big enough, which means that further approaches need to be done, in order to achieve realistic results.

The small variation in the reactivity, can be explained by the small characteristic mean free path [6], which makes that almost all of the neutrons expelled by the core are absorbed in water before reach the storage rack

Beyond this, the results displayed in Table IV are enough to conclude that the worst scenario in which the storage rack can be filled doesn't lead the system to the condition of supercriticality.

6. Conclusion

The most important conclusion in this work is that filling the rack with fuel elements won't lead to a supercritical system, since it induces only small variations in the value of k_{eff}.

Beyond that, it was shown that no major change over the irradiation channels were verified, meaning that the regular experiments can be done regardless of the fuel configuration in the rack.

Further calculations need to be done in order to determine the real influence of the storage rack on gamma radiation over the reactor pool and the possibilities of doing sample irradiation amidst of fuels in the rack.

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