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DETERMINATION OF AVERAGE NEUTRON FLUX IN A FUEL ELEMENT OF THE NUCLEAR REACTOR RP-10

XIX MEETING OF PHYSICS

PERUVIAN INSTITUTE OF NUCLEAR ENERGY

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The efficiency of the Reactor RP-10 is a determining factor that contributes to the energy research source and country's potential. This efficiency uses certain methods to determine criticality situations, neutron distribution and determination of reactivity [1]. These results can be obtained by means of calculation or experimental measurements. Within the neutron distribution aspect [2] the determination of the peak factor of the Reactor configuration is involved. In our research, a neutron activation method will be implemented to determine the mean value of flux in a fuel element at a current of 2.10 E-09 A; using gamma spectrometry systems for the detection of neutrons [3], uranium oxide fuels of RP-10 configuration No. 46, metal flake type indicators and wires. The positions evaluated by the experimental method are the elements of positions F2; C4 and E6. Also, it was compared with the determination by calculation using the Serpent code. The results obtained are part of a hybrid methodology (experimental-calculation), for the determination of the neutron flux in a nuclear configuration.

Situation - Motivation

Nuclear technology has always been a source of permanent research in order to determine factors that contribute to a better use of Nuclear Reactors, for which the implementation of various methods is required to allow better management of nuclear configurations. The characteristic method used in the neutron distribution is the neutron activation analysis applied to metallic indicators. The determined neutron distribution is used to analyze various factors; such as: Nuclear safety, determination of neutron power, relationship with the burning of fission fuel, production of radioisotopes, tests of various materials, etc.

Problem

It requires a laborious methodology with several stages of activities, in addition to testing the reliability of the method. (A.A.N.)

Power

↓

Average core flux

↓

Objective — **Average fuel element flux**

↓

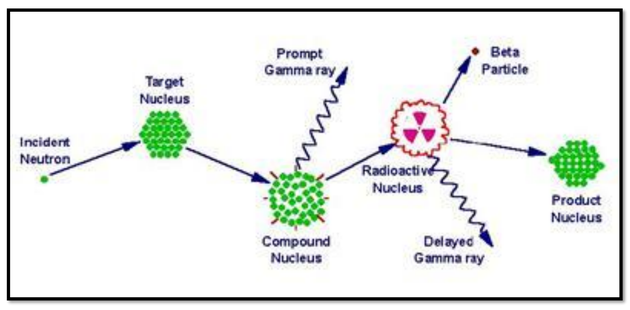
Problem — ¿Trustworthy? ¿Less laborious?

Objetive

- Determine the mean neutron flux of a fuel element at positions F2, C4, and E6.
- Compare the result using Serpent software calculation.

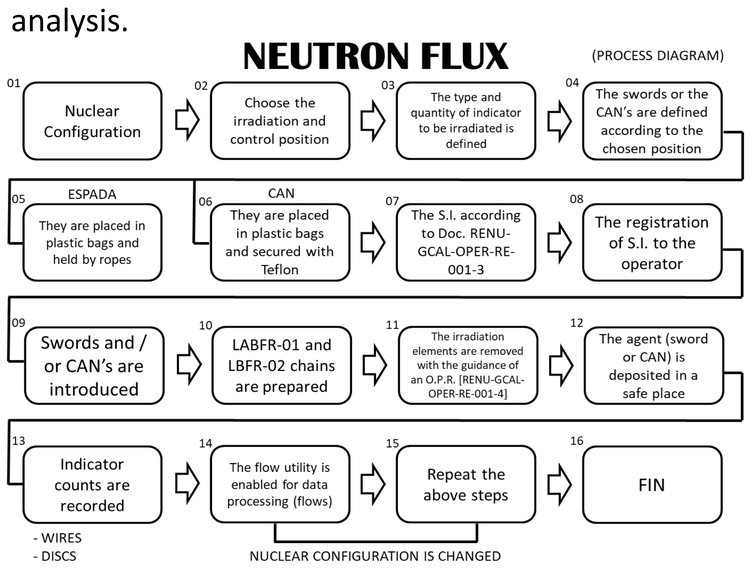
Nuclear Reactions with neutrons

The characteristic nuclear reaction is neutron capture. The indicators used are Au-198 and Cu-64.



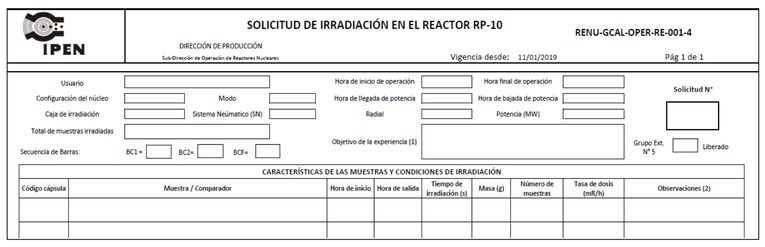
Metodology

The steps of this methodology involve: preparation of irradiation elements, request for irradiation, verification of measurement systems and data analysis.



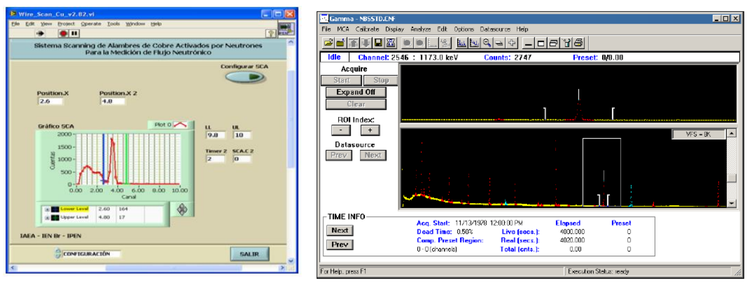
Preparation of irradiation elements

Irradiation Request



Verification of measurements systems (INa - GeHP)

Analysis of data (WSCAN – Genie 2000)



Mathematical Formulation

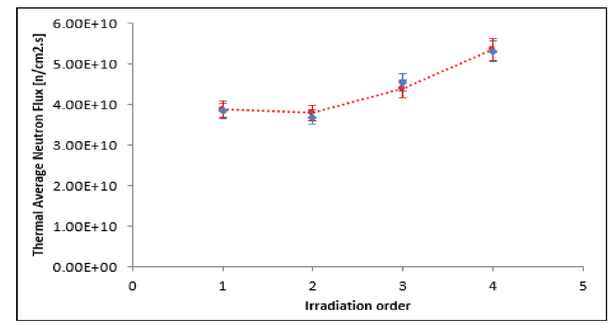
The Westcott formalism is used for data processing in the determination of mean neutron flux.

$$\Phi th = \frac{2}{\sqrt{\pi}}\sqrt{\frac{T}{T_o}}.\left(1-\frac{4}{\sqrt{\mu.\pi}}.rW\right).\Phi W$$

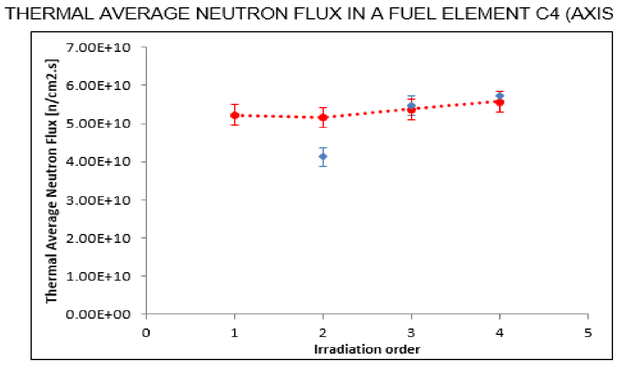
T: Neutron Temperature
μ: Factor that splices epithermal and termal regions
rW: Westcot spectral index
ϕW: Westcott neutron flux
ϕth: Thermal neutron flux

Results

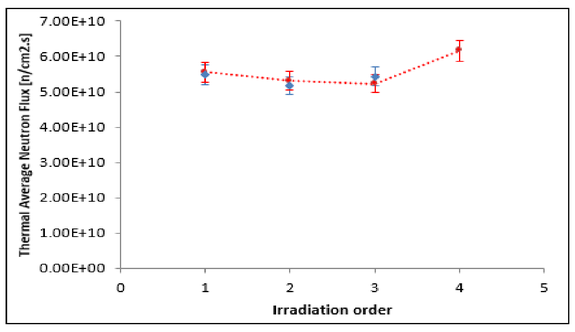
THERMAL AVERAGE NEUTRON FLUX IN A FUEL ELEMENT F2 (AXIS Z)



THERMAL AVERAGE NEUTRON FLUX IN A FUEL ELEMENT C4 (AXIS Z)



THERMAL AVERAGE NEUTRON FLUX IN A FUEL ELEMENT E6 (AXIS Z)



POSITION	THERMAL NEUTRON FLUX EXP. BY ELEMENT (MEAT) [$^{n}/_{cm^2.s}$]	THERMAL NEUTRON FLUX CAL. (MEAT) BY ELEMENT [$^{n}/_{cm^2.s}$]	RELATIVE ERROR [%]
F2	4.35E+10	4.36E+10	0.25
C4	5.12E+10	5.37E+10	5.00
E6	5.36E+10	5.37E+10	0.11

Conclusion

- At the operating current 2.10×10^{-09} A, the maximun experimental mean neutron flux was located at E6 and its value was $5{,}36 \cdot 10^{+10}$ n/cm^2.
- The mean maximum neutron flux by calculation was located at E6 and C4. Its value was $5{,}37 \cdot 10^{+10}$ n/cm^2.
- The relative error of both result does not exceed 5%

References

1. Montoya M., Rojas J. y Saetone E. Effects of neutron emission on fragment mass and kinetic energy distribution from termal neutron-induced fission of U-235, AIP Conference Proceedings 947, 326 (2007).
2. Zuñiga A, Lamas J., Cerrón E. y Huapaya I. Medición del factor de pico y potencia de un reactor nuclear, ICT (111-118),2007, IPEN, Peru.
3. Arias P., Paez J. y Vela M. Sistema de escaneado de alambrs de corbre irradiados en el reactor RP-10 usando WSCAN, ICT (57-61), 2011, IPEN, Perú.

Acknowledgments:

IAEA International Atomic Energy Agency
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