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DESIGN OF A NEW WET STORAGE RACK FOR SPENT FUELS FROM IEA-R1 REACTOR

Antonio C. I. Rodrigues, Tufic Madi Filho, Paulo T. D. Siqueira and Walter Ricci Filho

Instituto de Pesquisas Energéticas e Nucleares (IPEN / CNEN - SP)
Av. Professor Lineu Prestes 2242
05508-000 São Paulo, SP, Brazil
acirodri@ipen.br, tmfilho@ipen.br, ptsiquei@ipen.br, wricci@ipen.br

ABSTRACT

The IEA-R1 research reactor operates in a regimen of 64h weekly, at the power of 4.5 MW. In these conditions, the racks of the spent fuel elements have less than half of its initial capacity. Thus, maintaining these operating conditions, the storage will have capacity for about six years. Since the estimated useful life of the IEA-R1 is about another 20 years, it will be necessary to increase the storage capacity of spent fuel. Dr. Henrik Grahn, expert of the International Atomic Energy Agency on wet storage, visiting the IEA-R1 Reactor (September/2012) made some recommendations: among them, the design and installation of racks made with borated stainless steel and internally coated with an aluminum film, so that corrosion of the fuel elements would not occur. After an extensive literature review of material options given for this type of application we got to Boral® manufactured by 3M due to numerous advantages. This paper presents studies on the analysis of criticality using the computer code MCNP 5, demonstrating the possibility of doubling the storage capacity of current racks to attend the demand of the IEA-R1 reactor while attending the safety requirements the International Atomic Energy Agency.

Key words: Research Reactor, High-Density Storage, MCNP 5

1. INTRODUCTION

Spent nuclear fuel is generated from the operation of nuclear reactors of all types and needs to be safely managed following its removal from the reactor core. Spent fuel is considered waste in some circumstances or a potential future energy resource in others and, as such, management options may involve direct disposal (as part of what is generally known as the 'once through fuel cycle') or reprocessing (as part of what is generally known as the 'closed fuel cycle'). Either management option will involve a number of steps, which will necessarily include storage of the spent fuel for some period. This period of storage can differ - depending on the management strategy adopted - from a few months to several decades. The period of storage will be a significant factor in determining the storage arrangements adopted. The final management option may not have been determined at the time of design of the storage facility, leading to some uncertainty in the storage period that will be necessary, a factor that needs to be considered in the adoption of a storage option and the design of the facility. Storage options include wet storage in some form of storage pool, dry storage in a facility or storage casks built for this purpose. Storage casks can be located in a designated area on a site or in a designated storage building. A number of different designs for both wet and dry storage have been developed and used in different States [1]. The basic safety aspects of the storage of spent fuel from power reactors are applicable for the storage of spent fuel from research reactors. A proper graded approach - which takes the differences between the fuel types into account - should be applied. Issues relating specifically to the storage of research reactor fuel as the lower heat generation, higher enrichment and use of cladding materials that are less corrosion resistant, should be given particular consideration. Fuel composition, cladding material, shapes and sizes of fuel assemblies differ significantly in research reactors. For example, different fuel elements can be loaded into the research reactor and thus a variety of spent fuel can be generated. This may comprise, for example, fuel assemblies with different cladding materials (e.g. Al, stainless steel, Zr) or with different fuel compositions. In certain research reactors, reconstitution of an irradiated fuel assembly (e.g. by replacement of pins) is carried out [2]. Currently, the IEA-R1 reactor operates at full power of 4.5 MW during 32h weekly, and under these conditions the racks used in the storage of spent fuel elements are less than half of its initial capacity. What indicates an autonomy of approximately 6 years of operation under these conditions. The figures 1 and 2 show the photo with the current status of the storage of the reactor pool.

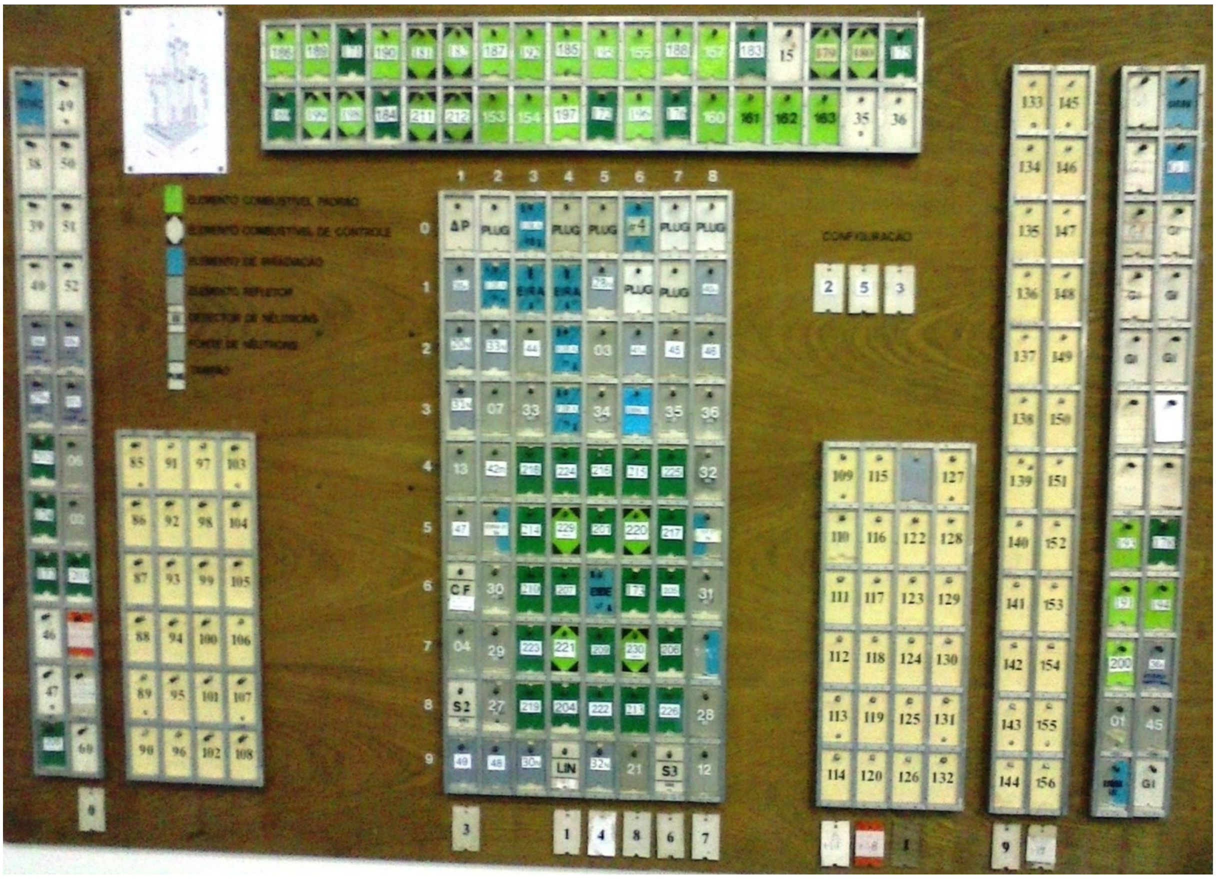

Figure 1: Current photo of the map plate matrix and racks of the IEA-R1.

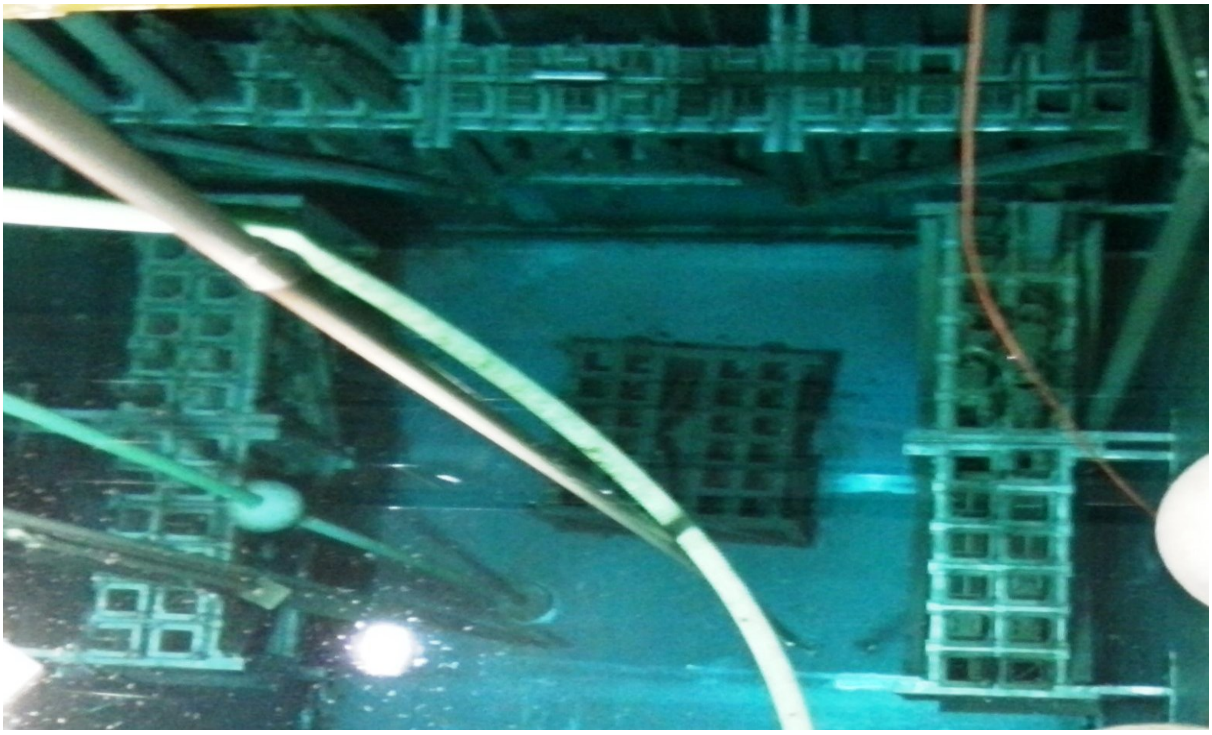

Figure 2: Current photo of the racks for spent fuel elements of the IEA-R1.

After extensive literature research about high-density racks for storage of spent fuel elements, the Boral ® was found and it has optimum cost / benefit.
BORAL® Composite the long-term integrity of a thermal neutron poison is essential for effective criticality control of spent nuclear fuel. In the nuclear industry worldwide, BORAL® has demonstrated long-term integrity when used in the design and fabrication of spent fuel pool storage racks and dual-purpose (storage/transportation) canisters and casks.
BORAL® is a precision hot-rolled composite plate material consisting of a core of mixed aluminum and boron carbide particles with an 1100 Series aluminum cladding on both external surfaces. The cladding forms a solid and effective barrier against the environment. BORAL® is produced over a wide range of surface dimensions, areal densities and thicknesses. BORAL® is manufactured in flat sheets that can be cut, punched, bored and formed into shapes. The physical properties of BORAL® allow it to be designed into fabricated structures as necessary [3].
Boron Products manufactures a standard grade of enriched boric acid to satisfy most nuclear applications. For those applications where standard product characteristics are not suitable or where an alternate purity is required, custom materials are also available.
BORAL® has the longest continuous service history of any neutron absorbing material and is currently used at seventy nuclear power plants and eleven research reactors worldwide. Decades of successful performance clearly demonstrate its effectiveness in high gamma and neutron radiation fields [4].

2. CALCULATION METHODOLOGY

For the purpose of comparison of the current and the study situation, each actual rack (6 X 2 matrix) was replaced by the rack under study (8 X 3 matrix) to the same dimensions as above.

This made it possible to double the storage capacity using the same area. In order to simulate the case the MCNP computer code version 5 was used [5].

2.1. The Code MCNP-5

Monte Carlo N–Particle is a general-purpose transport code which considers continuous-energy, generalized-geometry, time-dependent, coupled neutron/photon/electron. It can be used in several transport modes: neutron only, photon only, electron only, combined neutron/photon transport where the photons are produced by neutron interactions, neutron/photon/electron, photon/electron, or electron/photon. The neutron energy regime is from 10^{-11} MeV to 20 MeV for all isotopes and up to 150 MeV for some isotopes. The photon energy regime is from 1 keV to 100 GeV, and the electron energy regime is from 1 keV to 1 GeV. The capability to calculate keff eigenvalues for fissile systems is also a standard feature for neutron, photon, electron, or coupled neutron/photon/electron transport, including the capability to calculate eigenvalues for critical systems. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori.
Pointwise cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation (such as ENDF/B-VI) are accounted for. Thermal neutrons are described by both the free gas and S(α,β) models. For photons, the code accounts for incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption and absorption in electron positron pair production. Electron/positron transport processes account for angular deflection through multiple Coulomb scattering, collisional energy loss with optional straggling and the production of secondary particles including K x-rays, knock-on and Auger electrons, bremsstrahlung and annihilation gamma rays from positron annihilation at rest. Electron transport does not include the effects of external or self-induced electromagnetic fields. Photonuclear physics is available for a limited number of isotopes.
Important standard features that make MCNP very versatile and easy to use include a powerful general source, criticality source, and surface source; both geometry and output tally plotters; a rich collection of variance reduction techniques; a flexible tally structure; and an extensive collection of cross-section data.

2.2. Applied Methodology

The design basis and evaluation of rack criticality safety are consistent with the contents described in US-APWR Design Control Document (DCD) [6, 7]. Specifically, of the 10 CFR 50.68 (b), the item (2) and (3) for new fuel storage rack and item (4) for spent fuel storage rack are applied as the criticality safety design criteria. In addition, the analysis results were evaluated referring to ANSI/ANS-8.17-2004.
For new fuel storage racks, the maximum keff value including all biases and uncertainties must be less than or equal to 0.95 for the flooded condition with unborated water; and less than or equal to 0.98 for optimum moderation at a 95 percent probability, 95 percent confidence level (95/95). Rack cells are assumed loaded with fuel of the maximum fuel assembly reactivity [7]. Under the design criteria mentioned above, evaluations were conducted referring to the equation (1) described in the most recent ANSI/ANS-8.17-2004. More specifically, Section 5 of ANSI/ANS-8.17-2004 states that the calculated multiplication factor kp shall be equal to or less than an established allowable neutron multiplication factor; i.e.

$$k_p \leq k_c - \Delta k_p - \Delta k_c - \Delta k_m \quad (1)$$

If the various uncertainties are independent, we will have the equation (2)

$$k_p \leq k_c - (\Delta k_p^2 + \Delta k_c^2)^{1/2} - \Delta k_m \quad (2)$$

Where:
kp is the calculated keff
kc is the mean keff derived from the code validation
Δkp is the allowance for convergence*, tolerances, and modeling limitations
Δkc is a bias uncertainty derived from the code validation
Δkm is an arbitrary margin to ensure the subcriticality of kp
(* The 2σ value of MCNP output is applied according to the 95/95 rule.)

2.3. Modeling & Results

The simulated computationally arrangement with the MCNP consisted of a basket of 16 X 3 fuel elements totaling 48 elements. The basket was filled with new fuel elements (0% burning) of U_3Si_2-Al of 3.0 gU/cm^3, corresponding to condition with increased risk of criticality.

Table 1: Show the geometric specifications of standard fuel element IEA-R1 reactor

Parameter	cm
Nozzle Length	18.500
Space between nozzle and plates	2.400
Active board length	60.000
Above the active region to the top	6.400
Total length of the element	87.300
Water channel between the plates	0.289
"Pitch" between the plates	0.441
Board external fuel (2 units)	
length	71.440
active length	60.000
width	6.790
active width	6.260
thickness	0.152
active thickness	0.076
Internal fuel plate (16 units)	
length	62.500
active length	60.000
width	6.790
active width	6.260
thickness	0.152
active thickness	0.076

Table 2: show the geometric data of the basket

Parameter	cm
Overall length (x)	183.085
Overall width (y)	34.248
Overall height (z)	85.000
Number of cells ("rack")	48 unid.
Geometry of the cell (48 units)	
Length internal (x)	11.038
Internal width (y)	10.908
Length (x) - with half thickness	11.419
Width (y) - with half thickness	11.289
Overall height (z)	85.000
Sheet thickness	0.381
Pitch (x)	11.419
Pitch (y)	11.289

From these data and the MCNP Visual Editor Computer Code [8] aid could generate the input data required for the simulation, as can be seen in the next figures.

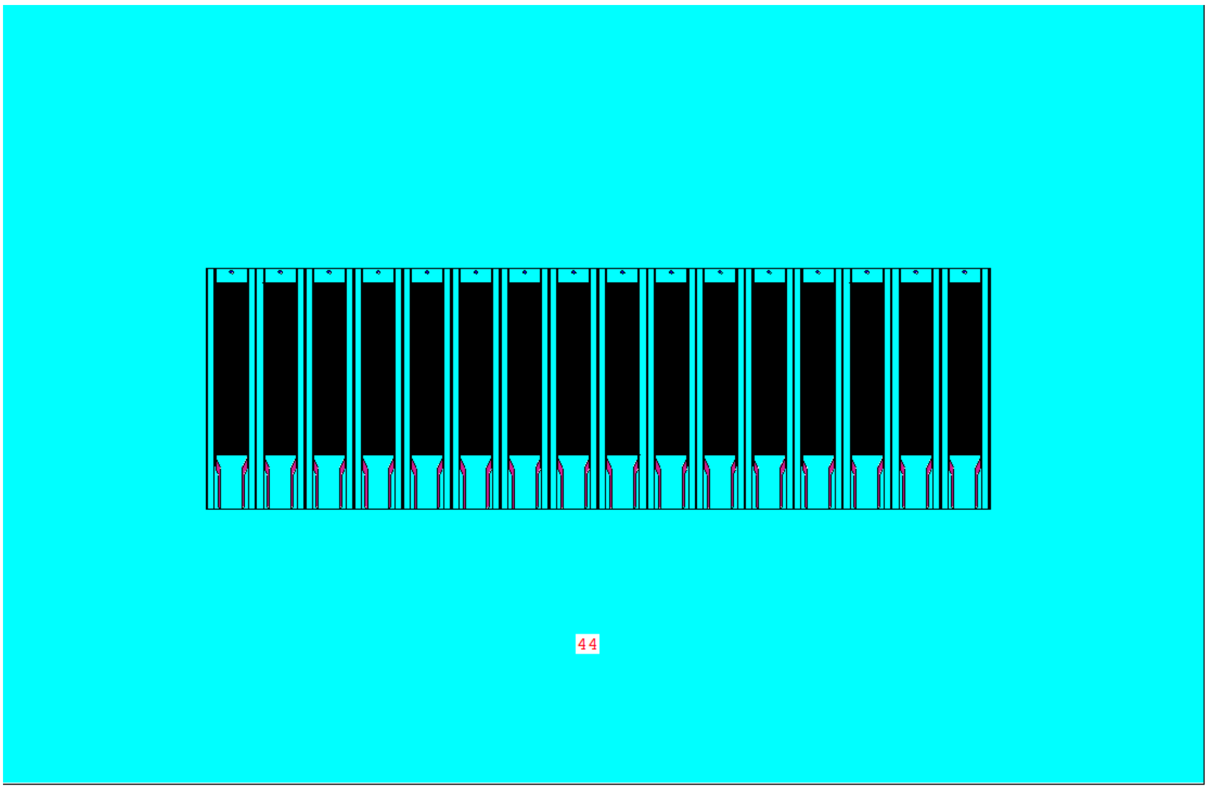

Figure 3: Simulated basket front view obtained with MCNP Visual Editor.

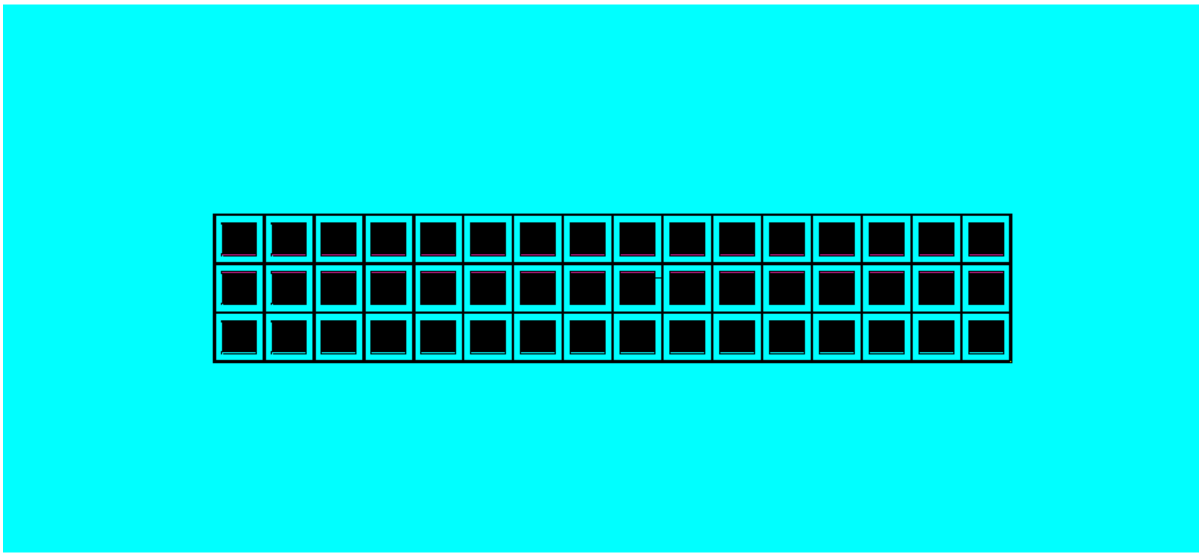

Figure 4: Top view of the simulated basket obtained with MCNP Visual Editor.

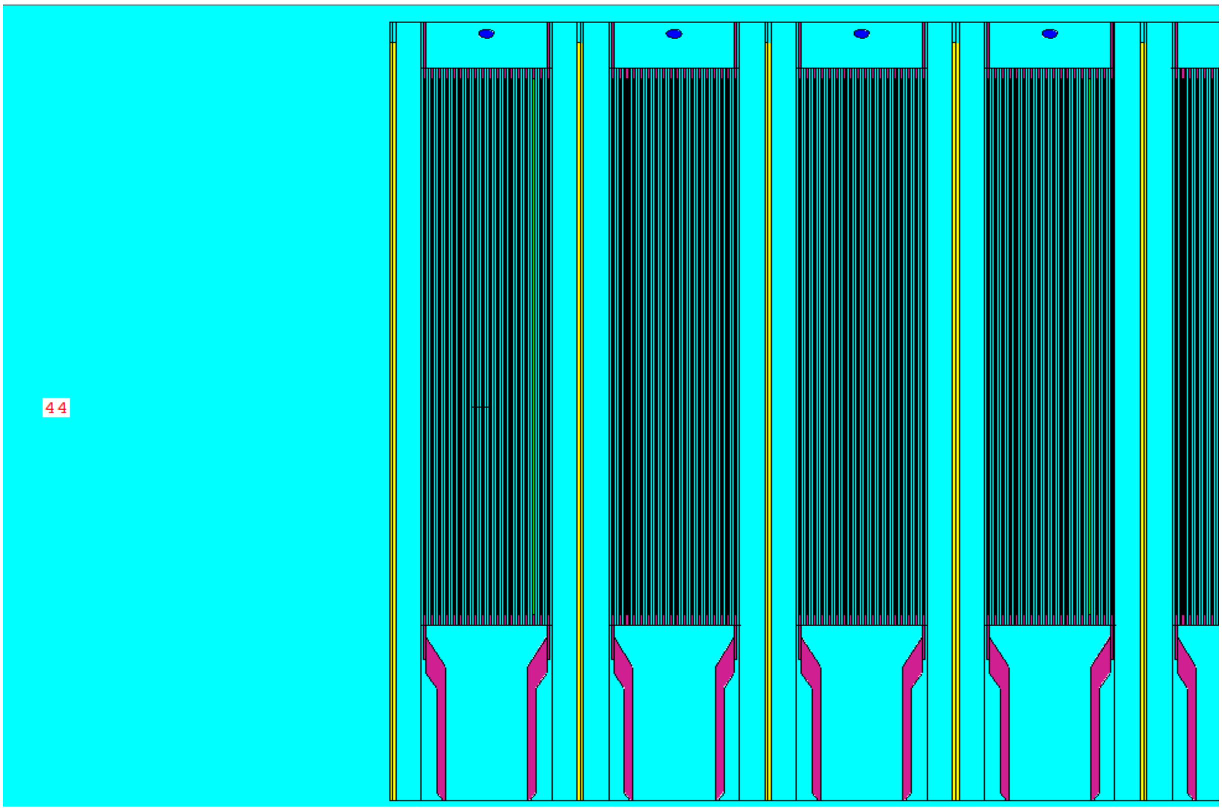

Figure 5: Zoom of the front view of the basket with fuel element obtained with MCNP Visual Editor.

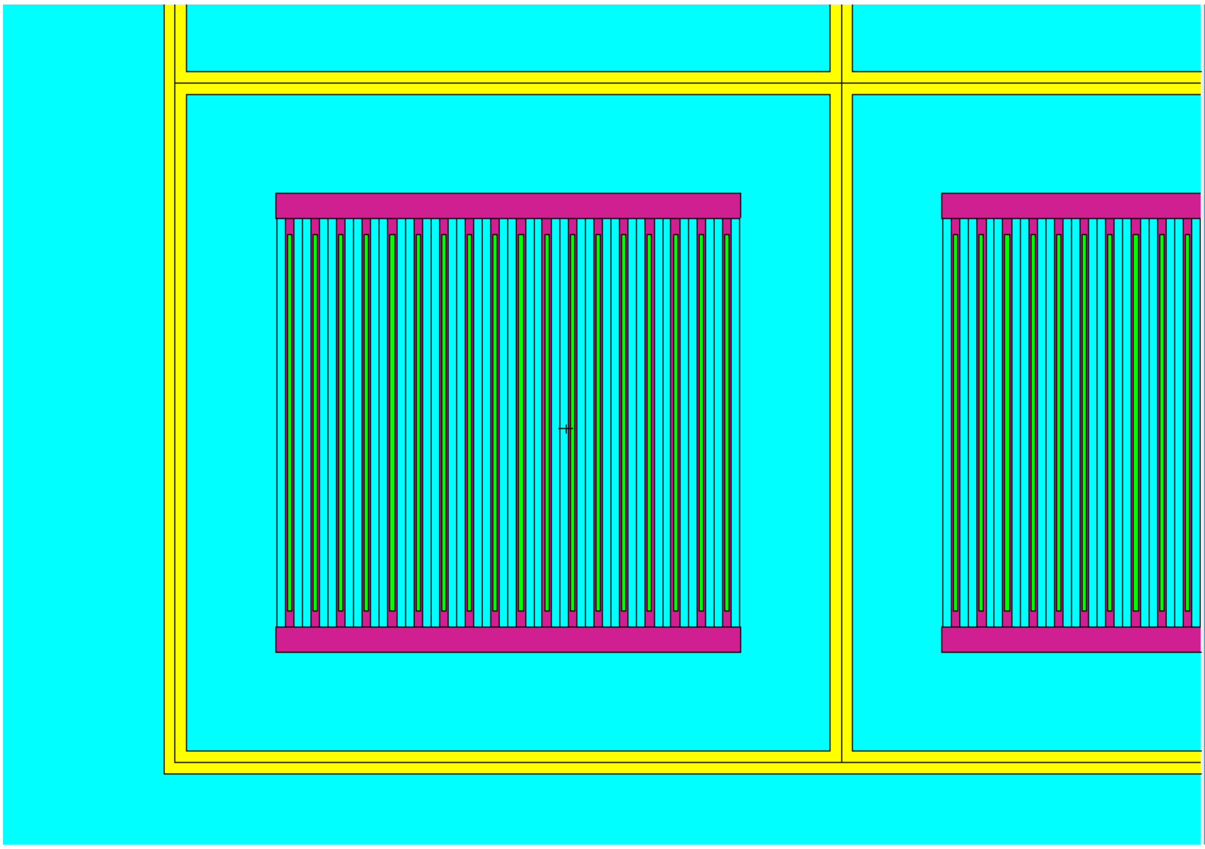

Figure 6: Zoom of the top view of the basket with fuel element obtained with MCNP Visual Editor.

After running the code MCNP 5 with 500 cycles and 100,000 histories per cycle the following results were obtained:

Table 3: It shows the results obtained

Problem	keff	Standard Deviation	68% Confidence	95% Confidence	99% Confidence
First Half	0.50926	0.00016	0.50911 to 0.50942	0.50895 to 0.50958	0.50885 to 0.50968
Second Half	0.50909	0.00016	0.50893 to 0.50925	0.50878 to 0.50940	0.50868 to 0.50950
Final Result	0.50918	0.00011	0.50907 to 0.50929	0.50896 to 0.50940	0.50889 to 0.50948

3. CONCLUSIONS

Based on the results obtained (keff=0.50918) and considering that all fuel elements loaded in the basket are new (maximum reactivity possible), we can conclude that we are comfortably

away from the safety limit is 0.95 as standard 10 CFR 50.68, thereby the sub criticality is maintained. The ENDF/B-VII libraries were used to be the most appropriate to this type of reactivity calculations.
With this project, it is possible to double the storage capacity of the racks to the spent fuel elements of the IEA-R1, and thus, increase the operational autonomy of the reactor around 15 years. The reactor core and the element of spent fuel storage compartment are interconnected; the same demineralized water passes through the two. Thus, the cooling heat of decay of fission products and the treatment of water for storage racks will be made while facilitating the operation.

4. FUTURE WORK

As a suggestion for future work, designing an arrangement "honeycomb" to better, take advantage of the bottom of the storage compartment of the reactor pool that is currently underutilized.
We also intend to perform the analysis and determination of bias (deltaKc).

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REFERENCES

1. Antonio C. I. Rodrigues, Tufic Madi Filho and Walter Ricci Filho. "Borated Stainless Steel Storage Project to the Spent Fuel of the IEA-R1 Reactor" *International Nuclear Atlantic Conference - INAC 2013*. Recife. PE. Brazil. November 24-29. 2013. IS*BN:* 978-85-99141-05-2 (2013).
2. INTERNATIONAL ATOMIC ENERGY AGENCY. Storage of Spent Nuclear Fuel. IAEA Safety Standards Series No. **SSG-15**. IAEA. Vienna (2012).
3. "Ceradyne. Inc Boron Products/ Nuclear Power/ Boral®" http://www.ceradyneboron.com/products/nuclear-power/neutron-absorbers/boral/ (2013).
4. "BORAL® Composite/ STANDARD SPECIFICATIONS" http://www.ceradyneboron.com/uploads/specSheets/SP-BORA-001en%20Specification%20Boral%20rev2.%2024-01-2011.pdf
5. X-5 Monte Carlo Team. *MCNP: A General Monte Carlo N-Particle Transport Code (Version–5)*. Los Alamos National Laboratory. LA-UR-03-1987 (2003).
6. Prevention of Criticality in Fuel Storage and Handling. 'General Design Criteria for Nucle ar Power Plants.' "Domestic Licensing of Production and Utilization Facilities." Energy. T itle 10. Code of Federal Regulations. Part 50. Appendix A. Criterion 62. U.S. Nuclear Reg ulatory Commission. Washington. DC.
7. 'Criticality Accident Requirements.' "Domestic Licensing of Production and Utilization Facilities." Energy. Title 10. Code of Federal Regulations. Part 50.68. U.S. Nuclear Regulatory Commission. Washington. DC.
8. L.L. Carter and R.A. Schwarz. *MCNP Visual Editor Computer Code*. Version 19L (2005)