A PRELIMINARY STUDY FOR THE UTILIZATION OF THE TRIGA RC-1 RESEARCH REACTOR AS A FACILITY FOR RADIATION DAMAGE

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ABSTRACT

TRIGA RC-1 Mark II reactor of ENEA's Casaccia Research Center reached its first criticality in 1960, with a maximum thermal power of 100 kW. In 1967 it was upgraded at the thermal power of 1 MW. Currently the core, fully reflected by graphite, contains 111 TRIGA standard SS cladded fuel elements (235U enrichment 19.90%, uranium weight fraction 8.5% of the UHZr alloy). The reactor is moderated also by demineralized light water, serving as first biological shield and coolant too. TRIGA RC-1 is equipped with various experimental channels and irradiation positions in-core and out of the core, providing a wide range of neutron and gamma fluxes and spectra useful for diverse applications.
During 2017 an agreement was signed between ENEA and the Italian Spatial Agency to cooperate in the field of neutron/gamma radiation damage analysis on electronic components to be used in future space-crafts. This agreement provides for use ENEA TRIGA RC-1 (and TAPIRO) research reactors as tools to perform neutron/gamma irradiation on such electronic components. In the meantime, in the frame of other activities focused on the evaluation of the current TRIGA RC-1 fuel burn-up level, a MCNPX model of the reactor has been implemented and validated by means of a comparison between experimental and calculated neutron flux spectra for different core positions, starting from the first core loading in 1967.
This paper describes the main steps moved up to now to characterize the facility neutron field and to evaluate some key ASTM(American Society for Testing and Materials)standard damage parameters, such as 1 MeV neutron equivalent flux and hardness parameter, using the MCNPX TRIGA RC-1 model. The description of the neutronic fields present in the available irradiation channels and facilities of TRIGA RC-1 to be used in the future experimental campaigns devoted to radiation damage analysis, always based on the results from the MCNPX model, completes the work described in this paper.

1. Introduction

During last years the necessity of a new justification for aged research reactors forces operators, with the support of the international community[1],to investigate the possibility of using such facilities, including also many low power installations, in a wide range of research fields. Since their licensing, occurred in many cases during the 60 and 70th, the original mission of these facilities has been abandoned, as in the case of Italy because the nuclear program was dismissed. Nevertheless, such research reactors represent a unique occasion for young generation scientists to take advantage of the wide possibilities of utilization of research reactors as neutron and gamma radiation fields sources. In many cases, research reactors can be useful also for radioisotope production or, as described in this paper, for investigations in the field of neutron/gamma radiation damage analysis. In particular, TRIGA type reactors show a great versatility also because quite often they are operated inside research centres.
This paper is focused on a feasibility study on the utilization of TRIGA RC-1 as a facility for neutron radiation damage analysis. At first, a full description of the first core configuration, labelled 38, is provided. For this configuration a wide range of neutron flux measurements was performed in the past. Then the TRIGA RC-1 MCNPX model is described together with its validation by means of different comparisons with experimental data concerning criticality calculations, control rods calibration curves and neutron flux profiles. The second part of the

paper focuses on the evaluation by the MCNPX model of some ASTM [8] (American Society for Testing and Materials) standard damage parameters, like 1 MeV equivalent neutron flux and hardness parameter, for a position in the central thimble.

2. TRIGA RC-1 Research reactor

RC-1 is a thermal pool reactor, based on the General Atomic (GA) TRIGA Mark II reactor design, operating at the thermal power of 1MW [2]. The core, in the current configuration, is loaded with 111 standard TRIGA fuel elements, it is contained in an aluminium vessel, seven meters deep, filled with demineralised water. A cylindrical graphite structure around the core is the lateral reflector of the reactor. The biological shield is provided by concrete with an average thickness of 2.2 meters. The water inside the vessel provides the first biological shield, neutron moderation and core cooling. Thermal power is removed from the core by natural convection, and exchanged with the environment through two thermohydraulic loops, coupled by two heat exchangers and two cooling towers. In Fig 1 the horizontal and vertical sections of the reactor are shown, together with a 3D section of the reactor with neutron channels.

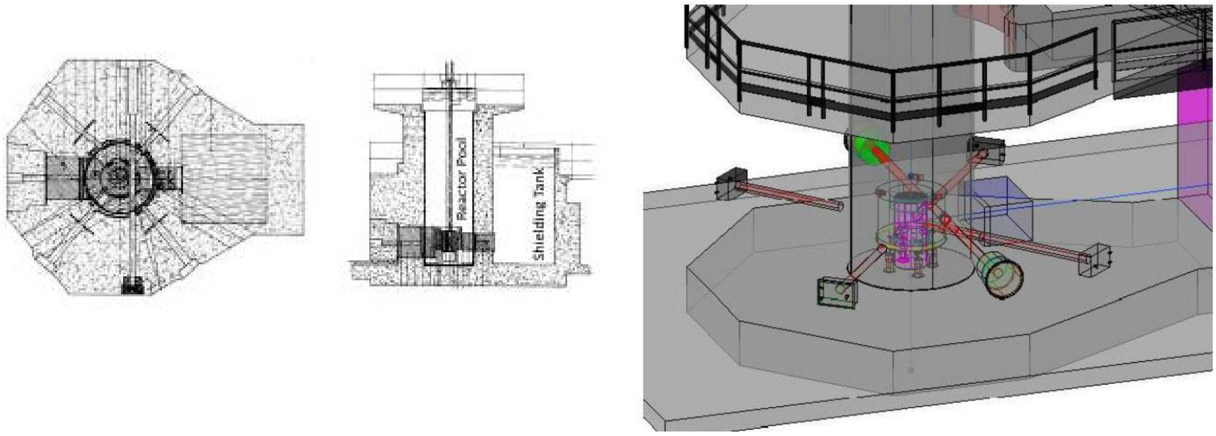

Fig 1 Horizontal and vertical sections of RC-1 research reactor and neutron channels

The RC-1 core, surrounded by a graphite reflector, consists of a lattice of TRIGA standard fuel elements, graphite dummies elements, control and regulating rods. There are 127 channels on the upper grid plate available for these core components and the grid itself is divided into seven concentric rings. One channel houses the start-up source (Am-Be) while two fixed channels are available for irradiation (central channel and *rabbit*).
The TRIGA fuel elements, cylindrical shaped and stainless steel cladded (AISI 304 - thickness 0.5 mm) consist of a ternary alloy of H-Zr-U. The Uranium is 20% enriched in ^{235}U, and represents the 8.5% of the total fuel weight. Two graphite cylinders at the top and at the bottom of the fuel rod ensure upper and lower neutron reflection. The fuel element is provided externally with two fittings in order to allow the remote movements and the correct placements into the grid plates. The metallurgic alloy stability is related to a variation of the total number of atoms less than 1% [3]. Another feature regards the prescription that forces the removal of elements from the core if their burn up is higher than 35%: this is a condition linked to the U-ZR-H lattice properties. From the point of view of the utilization, the reactor is mainly utilized for training, flux measurements and irradiation of neutron detectors.
The reactor is controlled by four boron carbide rods: three, stainless steel cladded, are *fuel follower* type (two shims and the safety rods) whereas the last, aluminium cladded, is the

regulation rod [2].In Table 1 the principal irradiation channels with associated neutron fluxare shown.

Description	Neutron flux($n \cdot cm^{-2} \cdot s^{-1}$)
Lazy Susan	2.00 10^{12}
Pneumatic transfer system(rabbit)	1.25 10^{13}
Central Thimble	2.68 10^{13}
Thermal column collimator	~1 10^{6}
Tangential piercing channel	~1 10^{8}

Tab 1 TRIGA RC-1 irradiation facility features in terms of neutron flux

3. TRIGA RC-1 1 MW core characterization

TRIGA RC-1 1 MW core has been loaded 22 times since the first criticality in 1967, corresponding to different configurations obtained by fuel shuffling or fresh fuel utilization (that corresponds to the removal of burned fuel from the core and its storage into the TRIGA RC-1 storage facilities).The first core configuration operating at 1 MW was obtained in 1967 and it's showed in Fig. 2: 76 fuel elements are arranged into 6 rings around the central thimble, filled with air. It represents the reference configuration for a MCNP model validation, since it is a 'zero burn-up' configuration (all fresh fuel loaded). A complete set of data have been measured such as control rods calibrations, neutron flux evaluation in various core positions and control rods calibration curves [5].

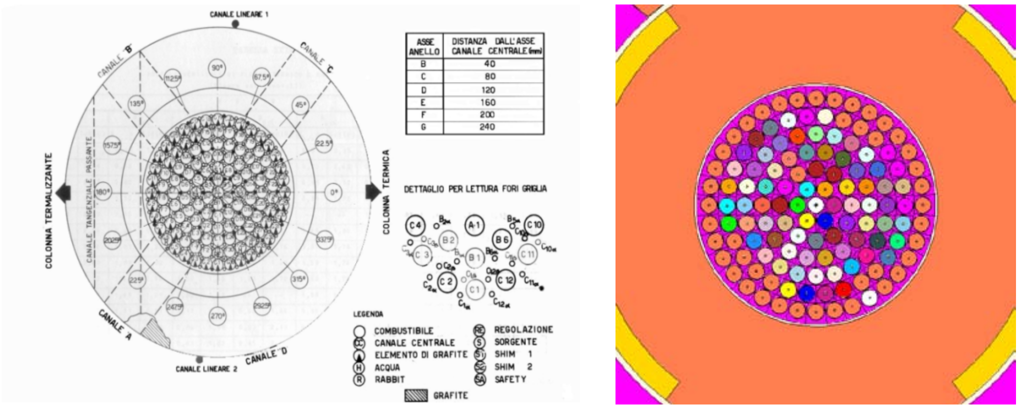

Fig 2 TRIGA RC-1 configuration #38 and corresponding MCNPX model

3.1. TRIGA RC-1 MCNPX model

The MCNPX [7] model described in this paper is based on detailed material compositions retrieved from plant documentation [3][4][5]. The considered reactor core configuration consists of 76 fuel elements at nominal zero burn up corresponding to the first historical core configuration at 1 MW [3], labelled #38. This choice is related to the necessity of validating the MCNPX model by means of criticality calculations, flux calculations and control rod calibration curves [4].As nuclear data the model uses ENDF/B-VII cross sections evaluated at 20 °C (together with the corresponding S(α,β) matrices for light nuclei),neglecting in this way the fuel temperature coefficient feedback (experimental value not less than -10

pcm/°C)for the steady operative condition at 1 MW. On the other hand, the validation process can be considered correct because some experimental data are evaluated at the thermal power of 20 W, corresponding to all core components in equilibrium at room temperature, 20 °C (isothermal condition). The last consideration is about the current geometrical model adopted in the calculations which does not include the thermalizing column and the outer part of the thermal column. This design choice is justified by the observation that the influence on k_{eff} evaluations by thermal and thermalizing columns can be neglected.
The model has been carefully analyzed from the point of view of materials: in particular it has been proved [7] that a correct zirconium content is crucial to obtain a good reference model regarding criticality. In the current MCNPX model zirconium has been considered with natural isotopic abundances.
Another point of interest is the fuel composition taken from the fuel materials sheets belonging to the shipment documentation from GA. Finally some important considerations have been done about the fuel cladding composition: two hypothesis have been considered, both generating credible MCNP models. The current MCNPX implemented model uses data for AISI 204 taken from a material compendium [9]. This last AISI composition shows sensible differences, as weight concentrations, respect to those adopted in the previous MCNP models. The results, mainly for criticality, seem to confirm that AISI composition from [9] is to be preferred.

3.2. MCNPXTRIGA RC-1 model validation

MNCPX [5]has been used to provided various evaluations on the TRIGA RC-1 1 MW core:

- criticality
- neutron flux
- control rods calibration curves

Criticality has been evaluated considering the control rods position reported in the reactor operation book. At 20 °C, or in other words at very low power, about 20W of thermal power, there is a good agreement between the MCNPX model and the experimental value (ρ=0):

$$(\rho \pm \Delta\rho)_{MCNPX} = (54 \pm 31)\ \text{pcm}$$

Even though there is enough agreement between experimental and evaluated data, it is important to note that the uncertainty on the cross sections has not been considered in this paper.

The MCNPX model has been used for neutron flux evaluations, by means of tally F4, in various core positions. The MCNP flux per unit neutron source has been normalized to a 1 MW thermal power for the comparison with experimental measurements.
Experimental data provided by TRIGA RC-1 documentation does not take into consideration errors and uncertainties on the procedure. Measurement can be affected by errors provided by the uncertainty in the positioning of the gold foils into grid positions and the uncertainty in the reactor steady state condition. Some fluctuations in the criticality are possible affecting the experimental data on the foil activation. At last, there is the uncertainty in the activation level of the foils, or better the uncertainty in the counting rate of the activated foils. At least a 5% of error affecting neutron flux experimental evaluations has to be taken into account. Experimental measurements and MCNPX evaluations are shown in Table 2 and Fig. 3, just for the thermal component of the neutron flux evaluated by using the cadmium threshold E_{cd} = 0.55 eV. Evaluated data by means of MCNPX are affected by an uncertainty derived from the F4 tally uncertainty and the error on the normalization factor, depending on the reactor

power level. The linear channel used by operators cannot allow a precision better than 10% on the power level measurement.

Ring	**Polar radius** (cm)	$(\Phi\pm\Delta\Phi)_{exp}$ $(n\cdot cm^{-2}\cdot s^{-1})*10^{13}$	$(\Phi\pm\Delta\Phi)_{mcnp}$ $(n\cdot cm^{-2}\cdot s^{-1})*10^{13}$
A	0	1.77±0.09	1.44±0.20
		2.68±0.13	2.20±0.30
		1.77±0.09	1.32±0.30
B[1]	5.5	0.53±0.04	0.83±0.15
		1.15±0.04	1.43±0.30
		0.74±0.04	0.64±0.12
C	6.2	1.60±0.04	1.51±0.24
		2.31±0.12	3.22±0.50
		1.17±0.04	1.34±0.20
D	13.4	0.55±0.03	0.77±0.10
		0.89±0.04	1.39±0.20
		0.62±0.03	0.71±0.10
E	14.5	0.85±0.04	0.80±0.10
		1.32±0.04	1.84±0.30
		0.69±0.03	0.88±0.13
F	21	0.57±0.03	0.87±0.13
		0.86±0.03	1.59±0.22
		0.57±0.03	0.93±0.14
G	25	0.80±0.04	0.69±0.10
		1.09±0.04	1.32±0.18
		0.75±0.03	0.72±0.11

Tab. 2 Experimental and evaluated neutron fluxes in various position in the core

For every polar radius in Tab.2 , corresponding to the rings filled with fuel elements (Fig.2), three evaluations and measurements have been performed at three different positions along the z-axis of the TRIGA RC-1 core and of the corresponding model: the first row in Tab. 2 represents a position very close to the lower graphite plug plane (red line on Fig. 3) , the second row corresponds to a position placed on the middle plane of the core (yellow line in Fig. 3), and the third (green line in Fig. 3) is relative to a point very close to the upper graphite plug plane.

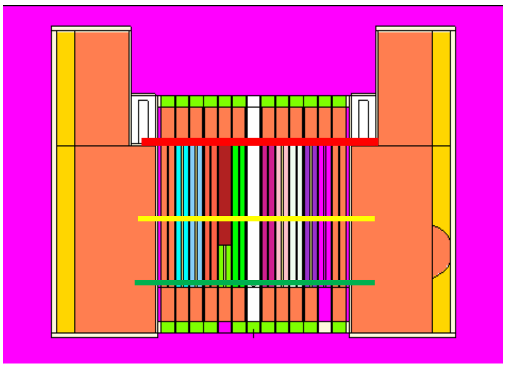

Fig. 3 neutron flux measurement positions

[1]The measurement and the calculated values refer to the closest position to the control rod fully inserted.

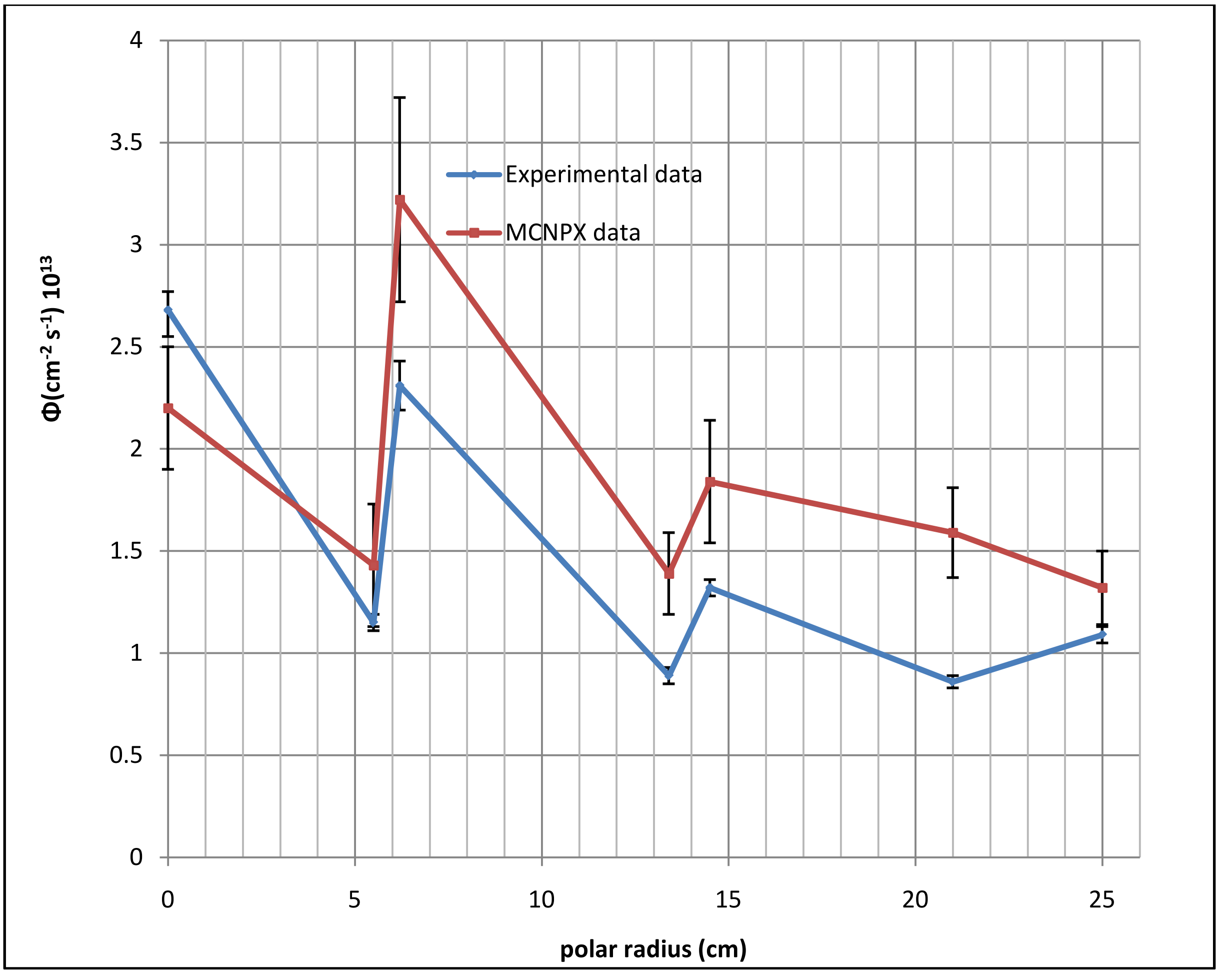

Fig. 4 Comparison between experimental and MCNPX evaluated neutron flux

It's worth of notice the flux depression (Fig. 4) in correspondence of rings B (at about 6 cm) and D (at about 14 cm). It's well understood from TRIGA RC-1 experience that neutron flux has its maximum value in correspondence of the ring B, but in our case the chosen position for ring D corresponds to a measurement point near the control rod fully or partially inserted.

The MCNP model has also been used to evaluate the control rods calibration curves. Results, regarding the safety control rod, are shown in Fig. 5 indicating a good agreement between experimental and evaluated data.

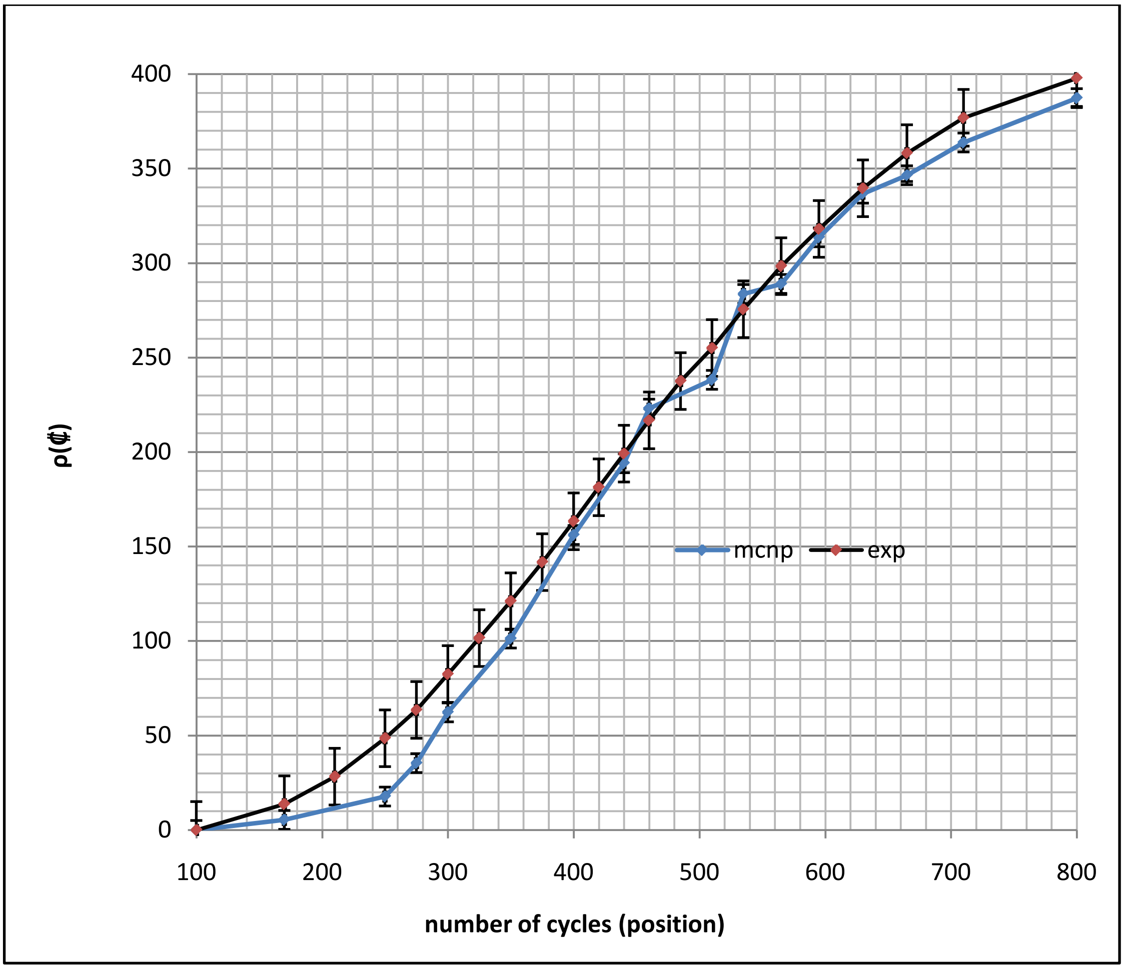

Fig. 5 Safety rod calibration curves: experimental and MCNPX results comparison

The comparison between experimental and evaluated data shows an average difference of about 20%, that it seems to be acceptable for the adoption of the present MCNP model in future evaluations. In particular, some measurement points are better reproduced by the MCNPX model, and in such cases the differences are less than 5%.

4. The ASTM standard damage parameters

In [8] are provided the definitions of the 1 MeV equivalent neutron flux and hardness parameter damage functions. Such functions are connected with quantities like the displacement KERMA (Kinetic Energy Released in Materials) functions (units [barneV]) for neutron collisions, which are provided for ^{28}Si and GaAs in a 640 energy groups SAND-II structure [10]. For each energy group g the damage KERMA functions are defined, for a given material m, as:

$$F_{D,g}^{(m)} = \sum_{\alpha} \sigma_{g,\alpha}^{(m)} < T_{g,\alpha}^{(m)} \cdot L(T_{g,\alpha}^{(m)}) > \quad [\mathrm{barn} \cdot \mathrm{eV}]$$

where the summation is over the α reaction channels (elastic scattering, inelastic scattering,...), <> indicate suitable average over the energy group g, σ are the microscopic cross sections, T are the energies of the PKAs (Primary Knock-on Atoms) and L(T) are the

Lindhard partition functions, providing the fraction of energy deposited in the lattice by the recoil atom cascade generated by a PKA of energy T.

As an example, in Fig. 6 is shown a comparison among the damage KERMA functions for ^{28}Si provided in [8] and provided by JANIS nuclear data base viewer [12] for JEFF 3.1 and ENDF BVII.1 nuclear data libraries. All the data are in the SAND II 640 energy groups structure [10]used in [8].

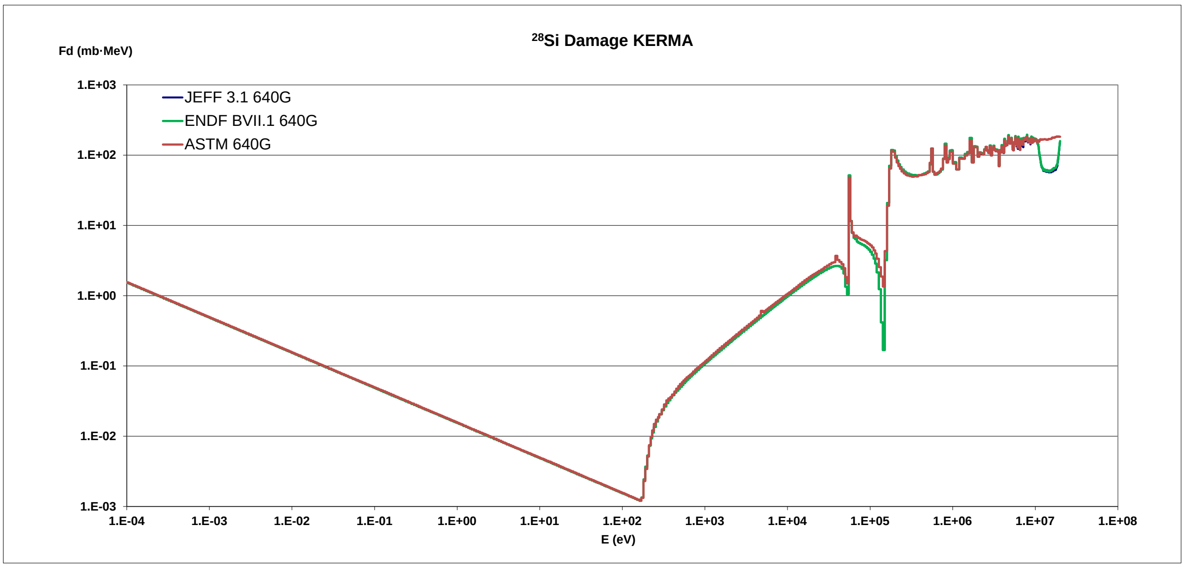

Fig 6. Comparison among damage KERMA functions for ^{28}Si provided in [8] (labeled as ASTM),JEFF 3.1 and ENDF BVII.1

In Fig. 7 are shown the JEFF 3.1 and ENDF BVII.1 relative differences respect to ASTM data.

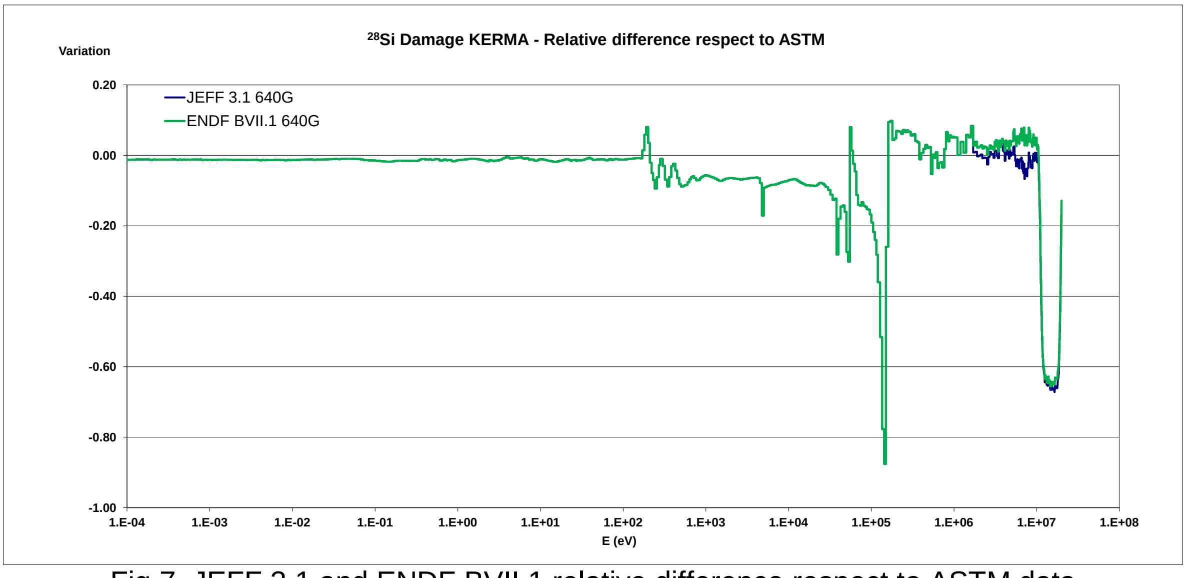

Fig.7. JEFF 3.1 and ENDF BVII.1 relative difference respect to ASTM data

From Fig. 7 it can be noticed that in some energy regions there are large discrepancies between JEFF 3.1 and ENDF BVII.1 respect to ASTM data (in particular above 10 MeV there are clearly different treatments of nuclear data). For these reasons it was decided in this work to take as reference ASTM data by using JANIS platform [11] to collapse these data (by using a so-called in JANIS “General spectrum” weighting function) from 640 energy groups to a 21 energy groups structure used to evaluate the damage parameters described in the following. In Fig. 8 the two KERMA data set are shown for both the two energy grids at 640 (original) and 21 energy groups.

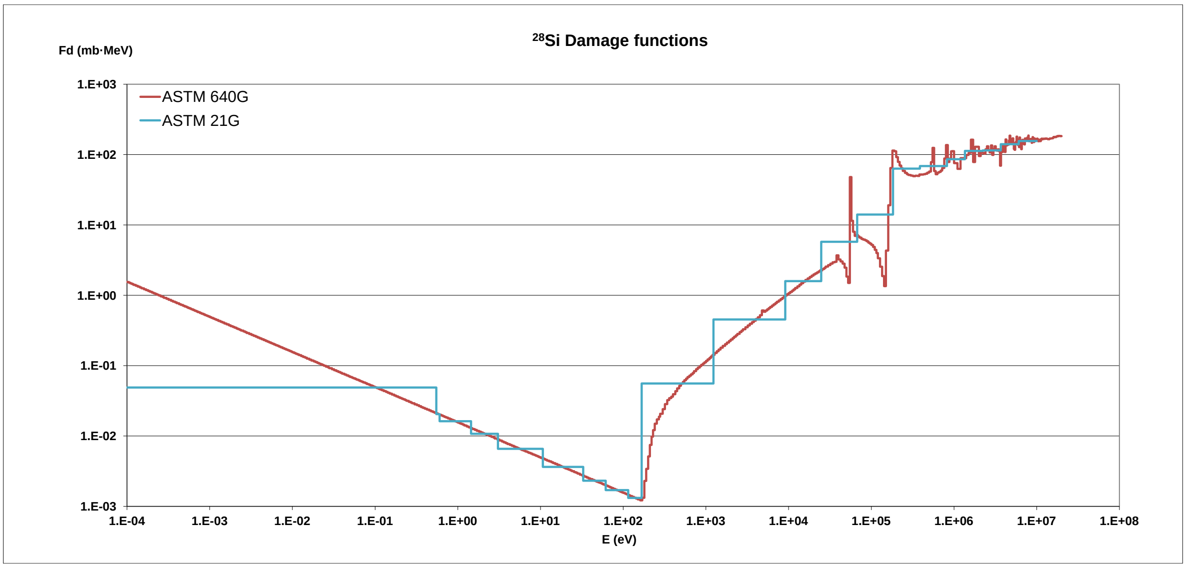

Figure 8 ASTM KERMA data set at 640 (original) and 21 energy groups

For a given position **r** in the facility the (monochromatic) 1 MeV equivalent neutron flux satisfies (by definition) the relation:

$$F^{(m)}_{D,1\ MeV} \cdot \phi_{eq}(\mathbf{r},1\,MeV) = \sum_g F^{(m)}_{D,g} \cdot \phi_g(\mathbf{r}) \qquad [eV \cdot s^{-1}]$$

i.e. this equivalent neutron flux has the property to produce the same damage power produced by the facility neutron flux at the same position **r** of the system. Therefore:

$$\phi_{eq}(\mathbf{r},1\,MeV) = \frac{\sum_g F^{(m)}_{D,g} \cdot \phi_g(\mathbf{r})}{F^{(m)}_{D,1\ MeV}} \tag{1}$$

In practice the denominator in (1) acts as a sort of normalization factor (for example, for ^{28}Si it is equal to 95 MeV·mbarn [8]). The hardness parameter H is defined as:

$$H(\mathbf{r}) = \frac{\phi_{eq}(\mathbf{r},1\,MeV)}{\sum_g \phi_g(\mathbf{r})} = \frac{\sum_g F^{(m)}_{D,g} \cdot \phi_g(\mathbf{r})}{F^{(m)}_{D,1\ MeV} \sum_g \phi_g(\mathbf{r})} \tag{2}$$

It can be seen from Eqs. (1) and (2) that to accurately evaluate these damage parameters we have accurately to know the reactor neutron flux intensity and spectrum in different positions, which in turns depend on reactor materials and reactor geometrical complexity, plus of course nuclear data.

5. Evaluations on the Central Thimble of the TRIGA RC-1 based on ASTM standard damage functions

Calculation of the 1 MeV equivalent neutron flux (Eq. 1) and the hardness parameter (Eq. 2) have been performed using the validated MCNPX model of TRIGA RC-1 applying the F4 tally in the central thimble (highlighted in red if Fig.9) at various positions along the z-axis corresponding to the fuel pellets described in Fig 9 with blue line box. The middle core plane is highlighted in yellow. Configuration #38, considered in this paper, has a central thimble filled with air. The neutron spectrum has been evaluated at each position using the 21 energy bins structure above described.

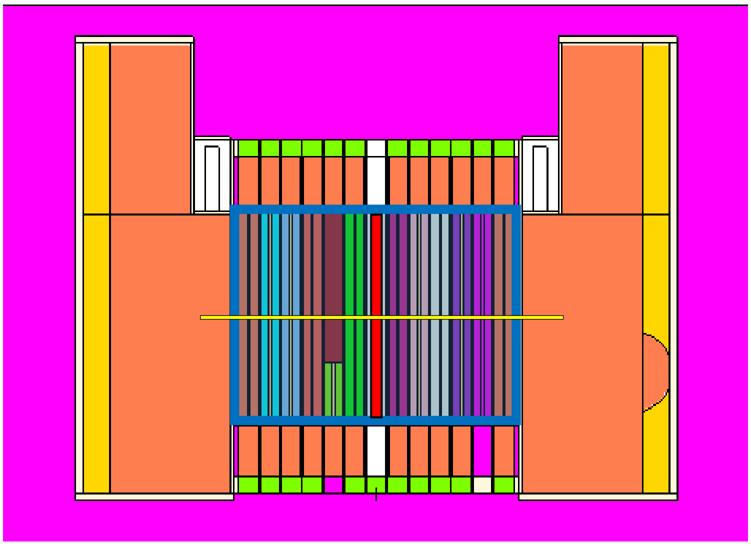

Fig. 9 Location of the fuel pellets (blue box), central thimble (red line) and middle core plane (yellow line)

The hardness parameter (Fig. 10) shows an interesting behaviour for positions from z=-19cm up to z=19cm (respect to the middle plane, highlighted in yellow in Fig. 9). The volume of the central thimble including z=-12 cm and z=12 cm is characterized by a plateau in the hardness parameter with an average value of 0.35, while it decreases smoothly going outside of this interval on the z -axis, with a relative decrease of about 10%.This behavior suggests that there is a wide zone in the central thimble around the middle core plane in which samples can be irradiated in similar hardness parameter conditions.

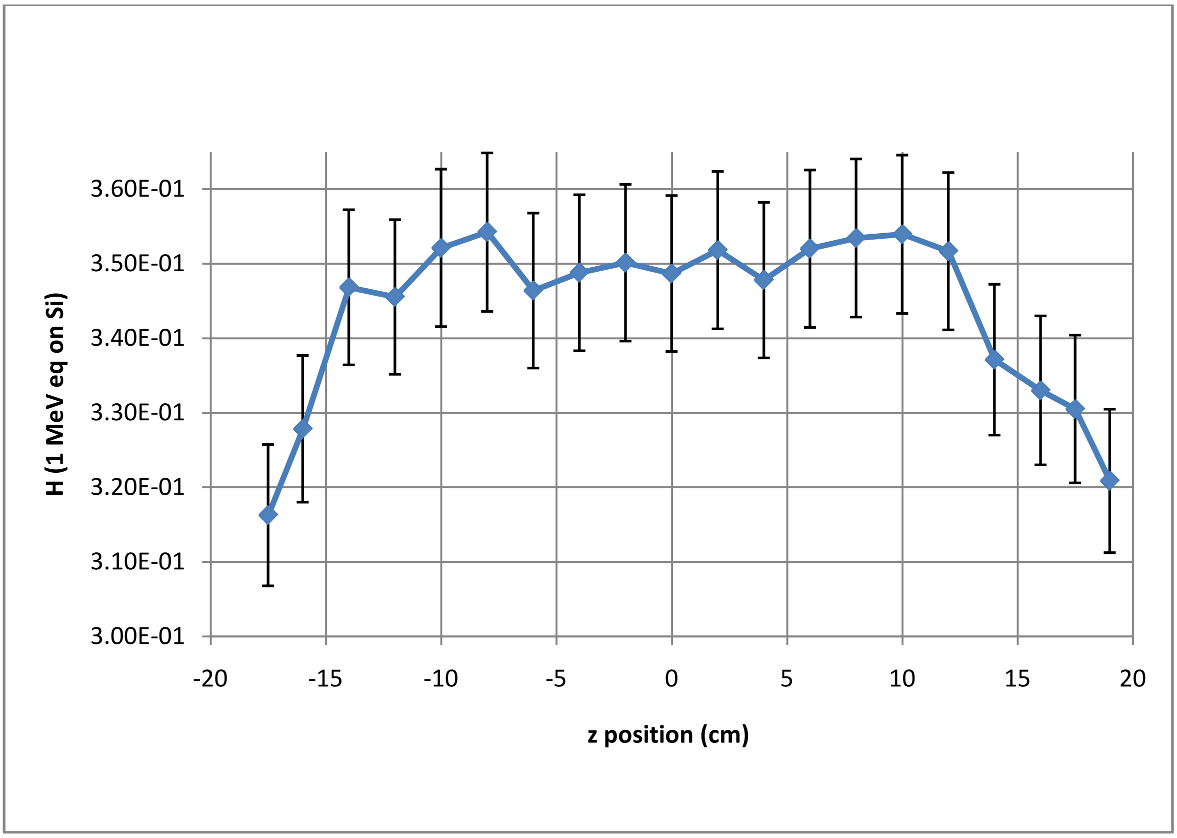

Fig. 10 Hardness parameter evaluation into the central thimble

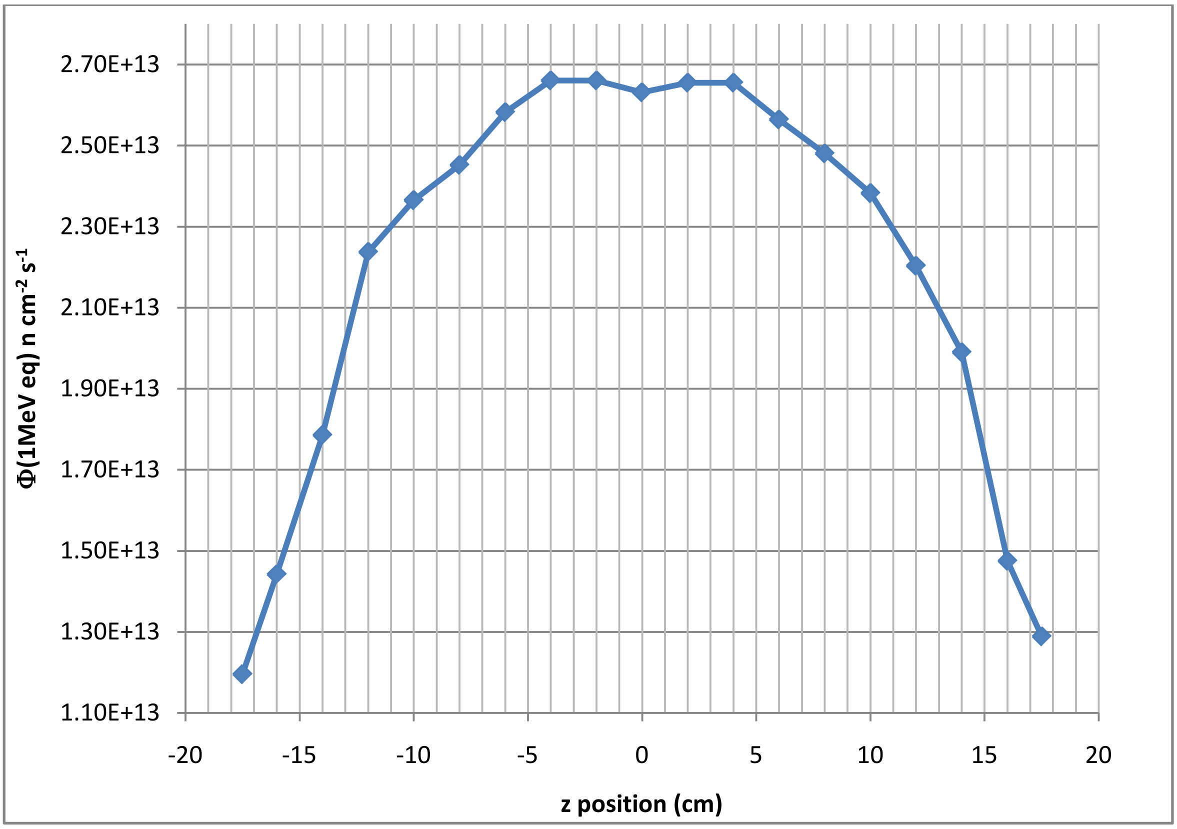

Fig. 11 1 MeV equivalent neutron flux for various positions in the central thimble

The 1 MeV equivalent neutron flux behavior (Fig. 11), evaluated at the same positions as in Fig. 10, is useful to optimize sample irradiations times and positions into the central thimble in order to obtain the desired damage level. Fig.11 shows that the relative variation of the 1 MeV equivalent neutron flux is about 5% in the interval from z =-12 cm to z = 12 cm, it is about 30% in the interval from z=-14 cm up to z=14 cm while it increase considerably at the edge of fuel active zone, close to the graphite plugs.

The hardness parameter for TRIGA RC-1, compared with those relative to a pure fission spectrum and the ENEA Fast Source Reactor TAPIRO [12], it's shown in Tab. 3.

Fission Spectrum	**0.85**
RSV TAPIRO	**0.55**
TRIGA RC-1	**0.35**

Tab. 3 Comparison of TRIGA RC-1 hardness parameter with fast spectra values.

This comparison suggests that, despite the thermal "nature" of the TRIGA RC-1 reactor, a not negligible hard component is present in the neutron spectrum in correspondence of the central thimble, and this feature candidates TRIGA RC-1 as a useful tool for neutron damage analyses.

6. Conclusions

The results obtained in this work, obtained by means of a validated TRIGA RC-1 MCNP model, demonstrates that this research reactor can be used as radiation damage facility, providing not only a good value of the hardness parameter but also a wide range of quite homogeneous conditions in the central thimble for samples irradiation. In particular a vertical zone around the mid-plane of the central thimble (about 30 cm) is characterized by the same value of the hardness parameter(0.35). These preliminary results are the first step towards a full characterization of the TRIGA RC-1 reactor in view of its utilization as a facility for neutron radiation damage analysis.

7. References

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