DEVELOPMENT AND EXPERIMENTAL VALIDATION OF A MONTE CARLO MODELING OF THE NEUTRON EMISSION FROM A d-t GENERATOR

Authors: Romolo Remetti[1] , Luigi Lepore[1], Nadia Cherubini[2],

Affiliations:

1) Sapienza University of Rome, Dept. SBAI, Via Antonio Scarpa 14, 00161 Rome, Italy

2) ENEA CRE Casaccia, Nuclear Material Characterization Laboratory and Nuclear Waste Management, Via Anguillarese 301, 00123 Rome, Italy

KEYWORDS: Neutron Generator, Neutron Emission, MCNPX, Thermo Scientific MP320

ABSTRACT

An extensive use of Monte Carlo simulations led to the identification of a Thermo Scientific MP320 neutron generator MCNPX input deck. Such input deck is currently utilized at ENEA Casaccia Research Center for optimizing all the techniques and applications involving the device, in particular for explosives and drugs detection by fast neutrons.

The working model of the generator was obtained thanks to a detailed representation of the MP320 internal components, and to the potentialities offered by the MCNPX code.

Validation of the model was obtained by comparing simulated results vs. manufacturer's data, and vs. experimental tests.

The aim of this work is explaining all the steps that led to those results, suggesting a procedure that might be extended to different models of neutron generators.

1. INTRODUCTION

Neutron generators are one of the most widespread sources of fast neutrons for different kind of elemental analyses applications, ranging from the quantitative estimation of the fissile/fertile materials in radioactive wastes packages, to the interrogation of containers suspected of containing illicit substances, like explosives or drugs.

To put in place a particular application some quantities have to be optimized, such as the neutron energy needed, and the neutron output rate required. High energy neutrons (> 10 MeV) and high neutron emission rates (>10^8 n/s) means the use of the $^2H(t,n)^4He$ fusion reaction, that produces, theoretically, 14.1 MeV neutrons. If lower energies are required (< 3 MeV), the $^2H(d,n)^3He$ fusion reaction is available, but with lower efficiency in production rate: commercial neutron generators based on d-d reaction do not exceed $5 \cdot 10^7$ n/s, usually. Anyway, lower energies neutrons can be produced by d-t systems also, by slowing-down the 14.1 MeV neutrons. The best solution has to be evaluated case by case.

Focusing on neutron angular emission, the d-t fusion process produces neutrons with a few percent anisotropy: because of the slightness of this effect the emission could be considered without any preferential direction, but as a first approximation only. In other words, the neutron generator internal target, in which fusion reactions occur, could certainly be considered as an isotropic source, but, as soon as neutrons leave the target, their first interactions happen with the generator's internal materials, causing scattering from their original trajectories. Summarizing, the neutron distribution vs. solid angle modifies itself in a way that is peculiar to the internal layout of the specific generator.

When performing R&D activities for measurement techniques based on neutron interrogation, in order to get an accurate evaluation of the source term, and its effect on the measurement of concern, it is mandatory to perform preliminary calibrations aimed at determining neutron's average energy, and flux, at determined angular positions vs. the generator's working parameters, such as supply voltage, ion beam current, frequency of the pulses, and duty factor. Typically, this is obtained by means of costly and time-consuming experimental calibrations. Monte Carlo simulation codes, when reproducing satisfactorily the angle-energy distribution of neutrons, may represent a powerful mean for reducing the experimental efforts of the calibration phase.

Other works concerning neutron generator simulations have been published so far, for instance concerning distribution of fusion products [1], neutron fluxes calculation outside shielding/moderating structures encompassing the generator [2], and dosimetric field around the generator itself [3]. The main target of this work is obtaining a designing tool allowing to estimate what is the neutron flux impinging on a measurement system placed at a fixed point

respect to the generator, and to calculate how neutron flux changes vs. generator's working parameters.

The neutron generator considered for this work is the d-t Thermo Scientific MP-320. This system is rugged, very compact, portable and light in weight. It is based on a deuterium-tritium sealed source and it is designed for laboratory or field applications. It generates a maximum of 10^8 n/s and it is capable of continuous or pulsed output, the latter ranging from 500 Hz to 20 kHz. According to the needs of the user, the MP-320 can be provided either for d-t and d-d fusion sources. More details about the device and its capabilities can be found in [4].
In the current research, the techniques the neutron generator is involved in require the employ of fast neutrons, e.g. Differential Die Away-time Analysis (DDAA), [5], for fissile and fertile recognition and quantification, and Fast Neutron Analysis (FNA), [6], for explosive detection. It means that the generator is used mainly unshielded, inside an appropriate bunker for radiation protection issues. In these conditions, devising the neutrons behavior with materials of concern is strictly function of the source term characterization, and a classical interpretation of the neutron generator as a point source, that is usually acceptable when the neutron generator is embedded into a thermalizing moderating structure, could be unsatisfactory. Since reference data from the manufacturer, and first experimental characterizations carried out at ENEA Casaccia Research Centre, had shown significant neutron flux depressions at certain angular positions, it was decided to develop a Monte Carlo model capable both to explain such an anisotropy, and to foresee its modifications when varying generator's working parameters.

2. METHODOLOGY

Currently, the most efficient method to simulate the neutron production by a neutron generator is the Monte Carlo simulation. By this approach, a lot of 'histories' of the particle of interest are run, in order to reproduce, into a virtual world, all the possible paths traveled by the radiation, by means of random samplings from all the probability density functions ruling physical phenomena. Very complex geometries can be built, and very accurate results can be produced if the user puts into the calculation all the elements important to the transport of the analyzed radiation; moreover, a sufficient computational power should be available for running a statistical significant number of histories.

For this work, the MCNPX (Monte Carlo N-Particle eXtended) code has been used, in its version MCNPX 2.7.0. [7]. By default, the code is able to transport deuterium and tritium ions inside the matter, but it does not treat the fusion reaction between those isotopes. For overcoming such a difficulty, a subroutine based on a fusion source model developed by Pillon et al. [8] has been added to the code, by proper re-compilation of MCNPX source code.

As stated by Pillon et al., the d-t fusion neutrons, produced by the ion beam-target interaction, show angle-energy dependence, as well as spatial anisotropy. For each neutron which is sampled, MCNPX obtains from the source-model subroutine the following information: *i)* the direction cosines *(u,v,w),* which establish the neutron emission direction, *ii)* the neutron energy (E), and *iii)* the particle weight W_n, [8]. The neutrons produced show few hundred keV spread around the theoretical 14.1 MeV, and few percent anisotropy vs. the angle of emission. It's worth to note that, for its correct application, the fusion source subroutine needs input data, such as the generator's target material, its dimensions and layers, the focal spot dimension, and the ion beam acceleration voltage. In particular, it is important to remark that the MCNPX source term was not modeled with the usual SDEF card. The fusion source subroutine considers a cylindrical tritium target with four-six coaxial layers. According to geometry and density of each layer, to the acceleration voltage, and to the focal spot, the subroutine calculates the points were fusion occurs on the basis of deuteron's range within the target itself. Once the starting point is fixed, neutron direction cosines, energy, and weight, are determined. Being the generator new, a fully loaded target was considered. To account for progressive depletion, periodical recalibrations have to be foreseen.

By the way, it is interesting to note that, by the same methodology, a source model for d-d fusion reaction has been developed also, [9], with the final aim to treat all the aspects relating to fusion processes and fusion generators.

By integrating the fusion source model into MCNPX, and having considered the successive interactions of fusion neutrons inside the generator, it was possible to achieving a reliable Monte Carlo model of the Thermo Scientific MP-320 neutron generator [4], capable to reproduce the real neutron emission from the device. Apart from the use of the source model subroutine, the key issue for achieving such a result was the simulation of the generator's internal components relevant for neutron transport. The need for simulating generator's internal components was suggested by the documentation provided by the manufacturer of MP-320 [10]. A first

comparison with an equivalent 14.1 MeV point isotropic source , intended as the simplest model for a d-t neutron generator, suggested that a more reliable MCNPX model could be developed to reproduce the angular emission given by the manufacturer. It should be noted that, as declared by the manufacturer, data on angular emissions had been originated by his own MCNP calculations. If schematics of the generator's internal structure had been previously known, no particular effort would have been required, but no schematics of the internal structure were available. Because of that, information have been crossed from the device's operation manual [11] and a IAEA document on the subject, [12]. The guessed internal structure of the generator has been adjusted, during the sequence of the simulations, in order to reproduce the neutron flux vs. the azimuth angle as rated by the manufacturer. With the information crossed by the references cited previously, about 45 sequential trials have been run to achieve a model capable of reproducing the real emission by the device accurately. The final working structure is presented in Figure 1.

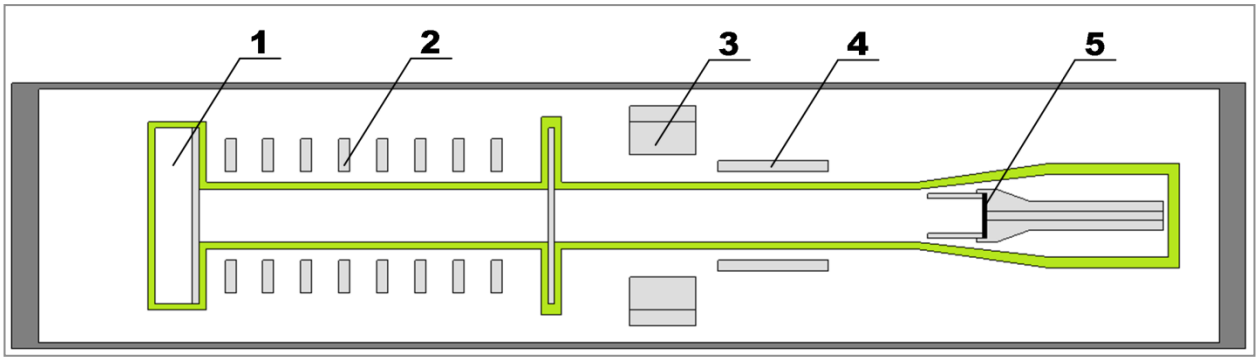


Figure 1 ThermoScientific MP-320 neutron generator guessed internal structure: 1) ion source and extractor lens; 2) pulser; 3) focus lens; 4) accelerator; 5) tritiated target.

From the figure, following the path of deuterium ions, it is possible to indentify:

1) an ion chamber for producing deuterium/tritium ions, coupled with an extractor lens for leading ions into the acceleration section;

2) the pulser;

3) a focus lens for shaping the deuterium beam;

4) the acceleration section;

5) the tritium-doped target.

The order listed before is particular of Thermo-Scientific MP320 neutron generator, and it has not to be intended as mandatory. Other instruments could use more components, or their order could be modified according to the aims to be achieved.

Once a good agreement between neutron angular flux by the MCNPX authors' model, and neutron angular flux by the MCNPX manufacturer's model, was achieved (Figure 2), the input deck retrieved has been used in field for experimental validations, as described below.

About computational power, the 45 sequential trials did not require particular efforts. For instance, 0.5 h·core per run was sufficient to get significant results with errors <10%, because generator's geometry is quite small.

Instead, high computational power may be needed when the user has to simulate the technique involving the use of neutrons. For instance, shielding, large geometries, high resolution in the energy and time domains, usually require the availability of HPC (High Parallel Computing) resources to achieve significant results in reasonably short time. For instance, the validation phase for the neutron generator model into the "Neutron Bunker" of Figure 5 needed about 500 h·core per simulation, due to the large masses of material the neutrons are interacting with.

The computing resources and the related technical support used for this work, and all future uses, are provided by CRESCO/ENEAGRID High Performance Computing infrastructure [13].

CRESCO/ENEAGRID High Performance Computing infrastructure is funded by ENEA, and by Italian and European research programmes [14].

2.1 Development of the neutron generator model

As materials are concerned, copper, iron, aluminum and aluminum oxide (alumina), especially in massive components, were assumed to be important for neutrons' scattering inside the device, while thin parts were neglected.

A first indication was given by the neutron flux angular distribution rated by manufacturer [10] via his proprietary MCNP input deck of the device: such data are reported in Figure 2. It should be noted that the angular flux has been calculated in a free-in-air condition: it means that the curve is not affected by secondary effects of interactions of the neutrons emerging from the generator with any structure. Such a condition is referred by the authors as 'uncollided flux'.

A set of 45 sequential trials was needed to develop the neutron generator model, as to reproduce, via MCNPX simulation, the neutron angular distribution given by the manufacturer in the

corresponding free-in-air conditions. The comparison of the model results vs. manufacturer's one is given in Figure 2.

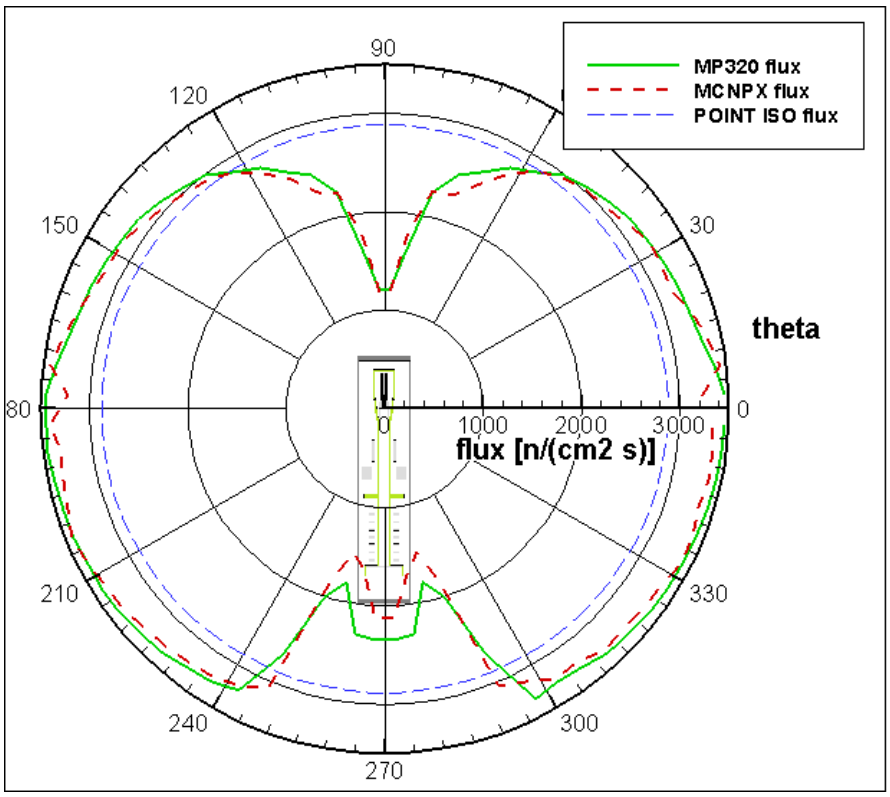


Figure 2 Comparison between the uncollided angular neutron emission, over all energies, by the MP320 neutron generator's MCNP model from the manufacturer (@ 50 cm far from the center of the target) and the angular flux calculated by the MCNPX model by the authors. For comparison, the uncollided angular neutron flux corresponding to the hypothesis of a 14.1 MeV isotropic point source is added.

With reference to Figure 2, maximum deviations from isotropy of the neutron flux vs. azimuthal angle happen at $\theta=90°$ and $\theta=270°$, in correspondence to the end-caps of the generator tube. From the shape of these neutron flux deviations, useful information about the internal layout were derived.

Referring to Figure 2, focusing on the $\theta=90°$ flux depression, it is clear that this modification is caused by:

- the alumina electric insulator of the electrode which drives the supply voltage to the doped target (the conical component sustaining item #5 in Figure 1);
- the copper electrode sustained by the alumina insulator.

It is evident from Figure 2 that the agreement between MCNPX calculations by the authors and the MCNP data given by the supplier is satisfactory.

It should be noted that an eventual hypothesis of considering the neutron generator as a point source of 14.1 MeV neutrons, without considering the interaction of fusion neutrons with generator's internals, leads to deceptive results for neutron flux. As observed in Figure 2, the generator's internals cause a deflection of neutrons emerging from the target, depleting the emission in some angles, and enriching in some others. For instance, when considering the same neutron yield, at $\theta=0°$ the neutron flux given by the accurate MCNPX model of the generator results 15% higher than that from isotropic point source assumption.

Use of moderating materials could certainly hide such a difference, but for DDAA and FNA techniques the neutron generator's use is mainly unshielded.

From Figure 2, it may be observed that around $\theta=270°$, the neutron flux behavior is more complex: two local minima can be noticed.

Such a behavior is due to:

- the copper coils of the pulser (item #2 of Figure 1) ;
- the conductive/ferromagnetic parts of the focus lens and the accelerator (item #3, #4 of Figure 1).

The maximum at $\theta=270°$ is due to neutrons that travel from the target (item #5 of Figure 1) to the deuterium source (item #1 of Figure 1) in the void central channel of the generator.

It is evident from Figure 2 that, in this case, the agreement between MCNPX calculation by the authors and the MCNP data given by the supplier is less accurate: although the trend is the same, an appreciable bias is present. Anyway, further improvements of the model were not carried out, because the $\theta=270°$ position is not involved with the current foreseen utilization of the generator.

About neutron yield, given that the manufacturer certifies the neutron output for a single setting of the generator (80 kV voltage of the accelerator, and 60 μA deuteron current, that corresponds to the maximum yield of $1.1 \cdot 10^8$ n/s $\pm$25% [15]), the Monte Carlo model allows to calculate the neutron production for every other settings by scaling down.

Current and voltage of neutron generators are the main working parameters to the neutron output rate. For the MP-320, they can be set into the range of 20-60 μA for the ion current, and 40-80 kV for the acceleration voltage. The neutron rate vs. ion current is generally quite linear. Instead, for the neutron rate vs. acceleration voltage, the manufacturer suggests a dependency

proportional to $\sim V^{1.5}$. Actually, this relationship is more complex, and can be derived by using the MCNPX modelization of the generator by combining the following data:

- the target construction data. For instance:
  - thickness and density;
  - doped depth;
  - doping concentration;
  - diffusion processes;
  - ageing.
- the penetration capabilities and interaction rate of the ion accelerated on the target.

Because of that, amid the set of MP320 generators built by the manufacturer, the neutron rate emission vs. the acceleration voltage is specific for each single item.

By means of the Monte Carlo approach used before, and thanks to the source subroutine by Pillon et al. [8], by knowing only the single emission rate from the calibration certificate by the supplier, it was assumed the possibility to calculate the emission rate corresponding to other working points, by normalizing the model results to the known point.

Because MCNPX results are normalized per source particle, for reproducing the device's output, it is necessary to adopt a normalization factor derived from manufacturer's data. The results of such a calculation are shown below.

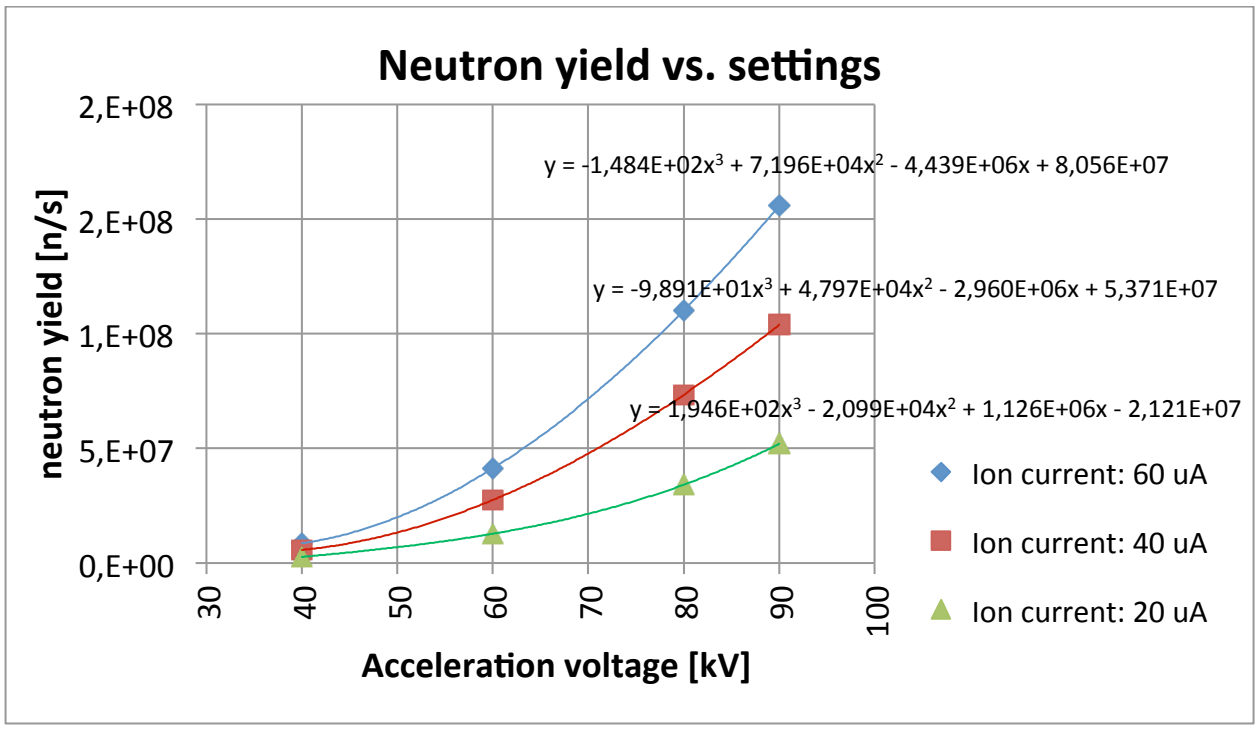


Figure 3 Thermo Scientific MP-320 neutron yields vs. accelerator voltage and deuteron beam current. Obtained via simulations calibrated with the manufacturer certificate [15]. 25% uncertainty on neutron yield should be considered, according to the uncertainty given by the manufacturer for the calibration point (60 µA @ 80 kV).

Figure 3 shows the variation of neutron yield vs. accelerator voltage for three current values. It is worth to remark that the supplier gives the output rate corresponding to the maximum yield only (60 µA @ 80 kV), with a 25% uncertainty. It should be noted that such uncertainty is propagated to the MCNPX predicted neutron yield via the normalization told before. Monte Carlo calculation errors are practically reduced to zero by running a very high number of histories (*nps* more than 10^9).

Such a procedure is to be intended as an integration of the manufacturer's information, and allows the user to determine the optimal setting parameters for obtaining the desired neutron yield. Figure 3 shows three curves but only one is really relevant, because the other ones result proportional by means of the ion beam current. Reporting all three lines into the graph is intended as a practical aid for the operator, for helping to select the appropriate neutron generator's working parameters according to the desired neutron output.

3. EXPERIMENTAL VALIDATION

An experimental setup has been realized at ENEA Casaccia "Nuclear Material Characterization Laboratory and Nuclear Waste Management" [16] in order to validate the simulated results. In particular for:

1) the emission at θ=0°, (Figure 2), with generator operated at its maximum neutron yield (60 μA @ 80 kV);
2) the emission at θ=0°, (Figure 2), with generator operated at its minimum neutron yield (20 μA @ 40 kV).

The θ=0° position, corresponding to the spot of maximum neutron flux, was used for the particular laboratory's arrangement for implementing DDAA and FNA.

The code validation has been carried out as follows:

1) accurate reproduction of the measurement condition's geometry and materials into MCNPX;
2) comparison of predicted neutron yields by the MCNPX neutron generator model vs. neutron yields measured in field at the minimum and maximum power, respectively (20 μA @ 40 kV) and (60 μA @ 80 kV).

The experimental neutron rate vs. generator's main operating parameters has been measured by means of a Berthold LB6411 neutron counter, in the geometrical configuration shown in Figure 4.

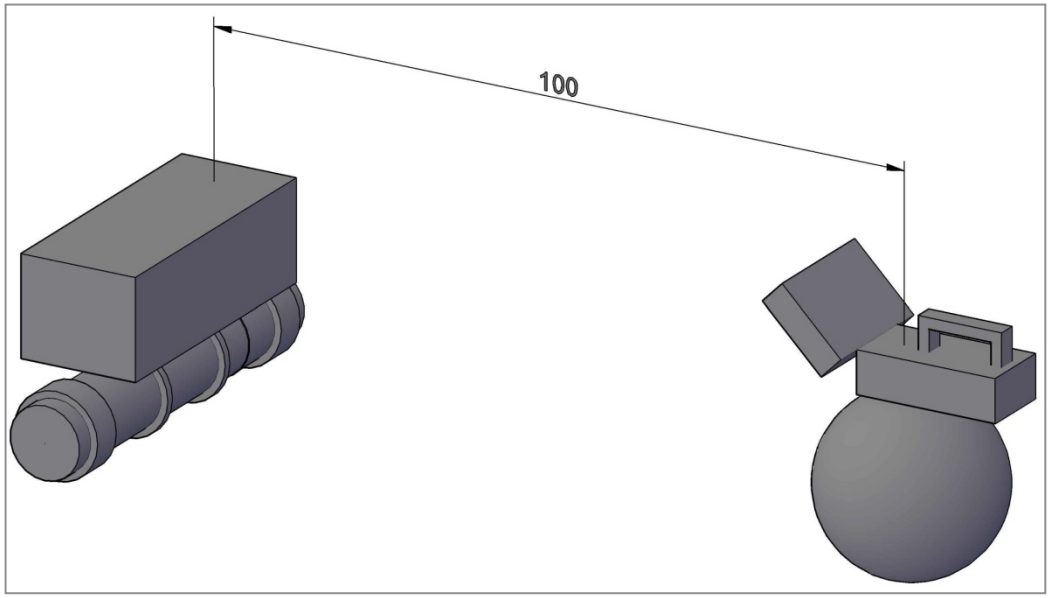


Figure 4 Experimental setup for Thermo Scientific MP-320 neutron generator and Berthold LB6411 neutron counter. Dimensions are expressed in cm.

As needed for obtaining accurate results from Monte Carlo calculations, MCNPX input included geometries and materials of the room where experimental measurements took place, as well as structural materials, *e.g*. walls, floor, table holding the generator, which could significantly affect the flux distribution. In particular, Figure 5 gives a description of the "neutron bunker" of the ENEA Casaccia TRIGA RC-1 reactor building, where measurements have been carried out. Walls and floor are made of ordinary or baritic concrete, while the ceiling is made of polyethylene sheets supported by steel beams. Other elements, *e.g.* the LB6411, other components of the neutron plug, P.E. and lead bricks present inside the bunker, were included into the simulation, but neglected in the representation given in Figure 5 for clarity of representation.

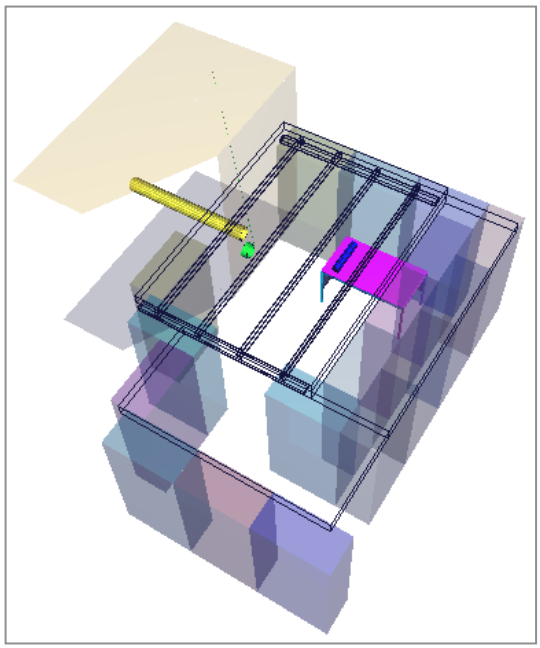


Figure 5 The "Neutron Bunker" where measurements have been carried out, as simulated in MCNPX Visual Editor. The generator (the blue cylinder) is laid on the table together with neutron counter (not represented). Together with the walls (gray boxes), and the ceiling (wireframe boxes), the reactor diametral thermal channel's water neutron plug (yellow and green cylinders) is represented.

Summarizing, the geometry of the experimental measurement has been reproduced into MCNPX very accurately, in order to get the same total detection efficiency as the real experimental test. About simulation details, 10^{10} neutron histories were considered, with computational efforts in the order of 500 h·core per simulation due to complexity and overall dimensions of the geometry. Regarding the experimental phase and focusing on the measurement of the maximum neutron yield by MP-320, having set the generator at (60 μA @ 80 kV) in a continuous irradiation mode, five count rates from Berthold LB6411 were registered. The averaged result was 1645 cps ± 2%. As known, LB6411 can be used to estimate the neutron flux, at the position where it is placed, by means of a conversion factor, called 'R' by detector's operating manual. R is function of the neutron spectrum entering the polyethylene sphere. According to the operating manual, the knowledge of the neutron spectrum is used to get a more accurate evaluation of R. It is obtained by weighting the function $R(E)$ on neutron energy spectrum, $\varphi(E)$, entering the detector,

$$R = \frac{\int R(E) \cdot \varphi(E)\, dE}{\int \varphi(E)\, dE}$$

It is worth to note that the *R* default value, stored in the detector, refers to ^{252}Cf spontaneous fission neutrons energy spectrum. *R* unit is cm^2; when count rate is divided by *R*, the local neutron flux is calculated.

As can be easily seen, while the count rate is retrieved experimentally, *R* cannot be evaluated directly by LB6411. The availability of the 'Neutron Bunker' (Figure 5) MCNPX input deck, shown before, can overcome the gap providing a computation of the required energy spectrum.

In order to evaluate the goodness of the MP-320 neutron generator model developed, two calculations have been carried out starting from the experimental count rates 1645 cps $\pm$ 2% given by LB6411:

1) calculation of factor *R*, and corresponding generator neutron yield, considering the generator as a 14.1 MeV point isotropic source;
2) calculation of factor *R*, and corresponding generator neutron yield, considering the generator model developed.

The relationship between the measurement by LB6411 and neutron yield is the following

$$measured\ neutron\ yield = \frac{measured\ count\ rate}{R \cdot F2\ tally\ on\ LB6411}$$

where *R* is the LB6411 correction factor discussed before, and the 'F2 tally' is a result of the MCNPX simulation; it physically represents the number of neutron entering the polyethylene sphere of the detector, divided by the number of started neutrons, divided by the detector's polyethylene sphere surface area.

All results are resumed in the following Table 1.

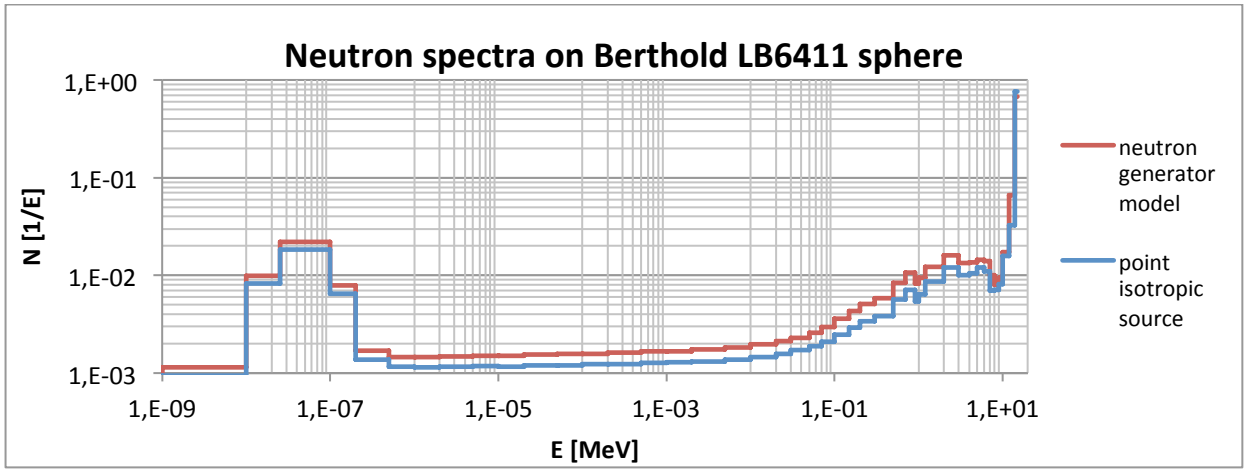
	R [cm^2]	F2 tally on LB6411 surface [cm^{-2}]		Measured neutron yield [n/s]	Δ% with manufacturer reference
14.1 MeV point isotropic source	5.627E-01	2.238E-05		1.307E+08	+18.8%
MP-320 model by the authors	5.606E-01	2.432E-05		1.207E+08	+9.7%

Table 1 Comparison of the measured neutron yield by means of Berthold LB6411 neutron detector vs. the reference $1.1 \cdot 10^8$ n/s rated by manufacturer (60 μA @ 80 kV).

330

331 Differences in *R* values and F2 tally values in Table 1 can be explained as follows.
332 Referring to Figure 2, the neutron flux angular distribution given by the accurate MCNPX model
333 of the generator differs in its shape from isotropic point source assumption: it means that also the
334 interactions inside the 'Neutron Bunker' in Figure 5 will differ. Such differences, that affect both
335 *R* and F2 tally values, can be viewed throughout the neutron spectra on LB6411 sphere surface,
336 as reported in Figure 6 and Table 2.

337



338

Figure 6 Neutron spectra entering the Berthold LB6411 polyethylene sphere inside the ENEA Casaccia TRIGA RC-1 'Neutron Bunker'. MCNPX calculations have been run in two cases: neutron generator as a point isotropic source (blue curve), neutron generator as the model proposed by the authors (red curve).

339

Neutron energy range	Neutron generator model	14.1 MeV point isotropic source
0-0.025 eV	6.85%	6.61%
0.025–1 eV	22.10%	21.31%
1–200 eV	4.64%	4.18%
200 eV–1 MeV	15.04%	12.31%
1–12 MeV	18.73%	16.81%
12–15 MeV	32.63%	38.77%

Table 2 Numerical values of the neutron spectrum energy composition entering Berthold LB6411 sphere.

340

The most significant differences in neutron energy spectra can be easily seen in Table 2. Starting from the first row, the first three energy ranges present almost the same values: multiple scattering phenomena that produce such a kind of neutrons are due to the "bunker effect", and are less dependent on the way the neutron generator output is modeled. The fourth and the fifth ranges, calculated via the neutron generator model, show larger results than the isotropic point assumption: that is the effect of the generator 'internals' whose primary interactions with 14.1 MeV fusion neutrons deplete the sixth range enriching the forth and the fifth ones. That explains also why the sixth range results larger for the isotropic point assumption.

The entire procedure shown above has been repeated also for the minimum neutron generator power. The experimental count rate of LB6411 is 35 cps $\pm$ 13.7%. Applying the appropriate conversion, measured neutron yield are reported below.

	R [cm^2]	F2 tally on LB6411 surface [cm^{-2}]		Measured neutron yield [n/s]	Δ% with MCNPX predicted value
14.1 MeV point isotropic source	5.627E-01	2.238E-05		2.78E+06	+2.7%
MP-320 model by the authors	5.606E-01	2.432E-05		2.57E+06	-5.5%

Table 3 Comparison of the measured neutron yield by means of Berthold LB6411 neutron detector vs. the value $2.71 \cdot 10^6$ n/s predicted by MCNPX model (20 μA @ 40 kV) calibrated via manufacturer data.

A comprehensive representation of results reported in Table 1 and Table 3 is shown in Figure 7.

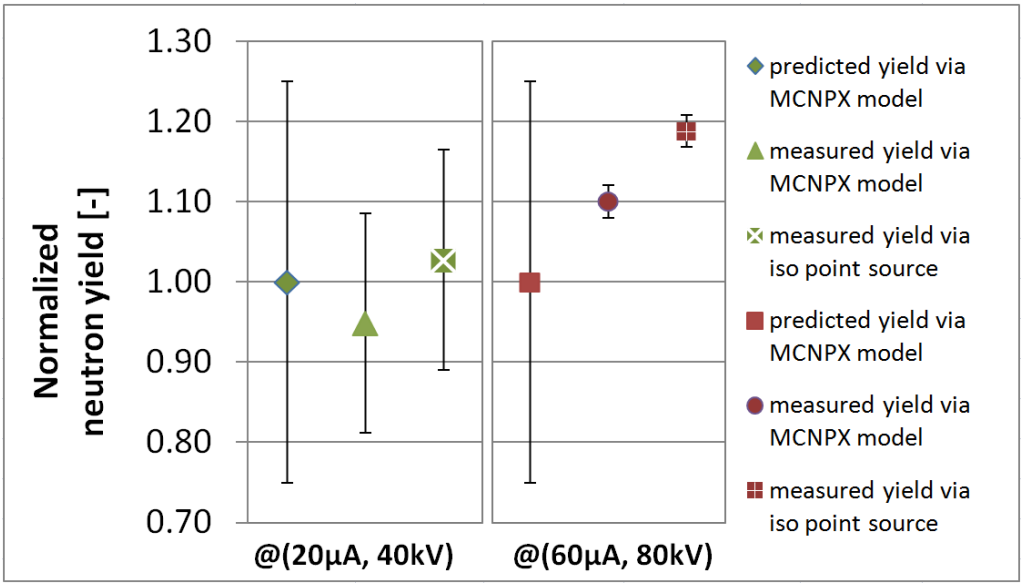


Figure 7 Comparison between the simulated emission rates and measurements at minimum and maximum neutron generator yields. Measured yields have been normalized to predicted values. Measured points are presented according to the different *R*-correction models.

At (20 μA @ 40 kV), the Δ% experimental vs. simulated, using both neutron generator model and isotropic point source assumption, are -5.5% and +2.7% respectively. At (60 μA @ 80 kV), the Δ% experimental vs. simulated, using both neutron generator model and isotropic point source assumption, are +9.7% and +18.8% respectively. However, when minimum yield is measured, counting statistics on LB6411 is poorer than measurements at maximum yields, as can be graphically seen with uncertainty bars in Figure 7.

It is worth to note that

i) the comparison of experimental vs. simulation measurements at minimum and maximum yields at θ=0°,

ii) the agreement among manufacturer angular flux distribution and the angular flux by the MCNPX model,

iii) the more likely neutron energy distribution of the MCNPX model vs. the isotropic point source assumption,

increase the confidence in using the input deck developed.

Summarizing, the MCNPX model developed, in order to optimize DDAA and FNA techniques, seems to behave better than the simple isotropic point source.

4. CONCLUSIONS

The application of the methodology previously described to the Thermo-Scientific MP-320 neutron generator led to the identification of a working MCNPX input deck of the device, capable to reproduce the neutron emission in good agreement vs. measurements.

First, a pure simulation procedure has been carried out as to retrieve the generator internal configuration capable in reproducing, in the free-in-air condition, the neutron flux angular distribution given by the manufacturer [10].

Once a sufficient agreement has been reached, the MCNPX model, with the features provided by its *ad hoc* neutron source by Pillon et al. [8], has been used to retrieve the operating curve of the device, namely neutron yield vs. settings, via a calibration of the model by means of the manufacturer neutron yield certificate at maximum power.

An experimental verification phase has been carried out, measuring the neutron yield at minimum and maximum power by means of a Berthold LB6411 neutron detector. The position selected for the measurement is $\theta=0°$ (Figure 2), being the spot where neutron flux is maximized. The real measurement geometries and materials have been reproduced very accurately into MCNPX as to provide a reliable calibration for LB6411 as to get a valid measurement of the generator neutron yields.

Experimental results vs. simulated have been compared: the MCNPX generator model provided by the authors behaves better than 14.1 MeV-isotropic point assumption:

1) giving more accurate values for the neutron flux at the angular position $\theta=0°$ (Figure 2),
2) giving more representative neutron spectra at the angular position $\theta=0°$ (Figure 2),
3) providing a set of operating curves for the device, namely neutron yields vs. settings.

All described steps are resumed into the workflow presented in Figure 8.

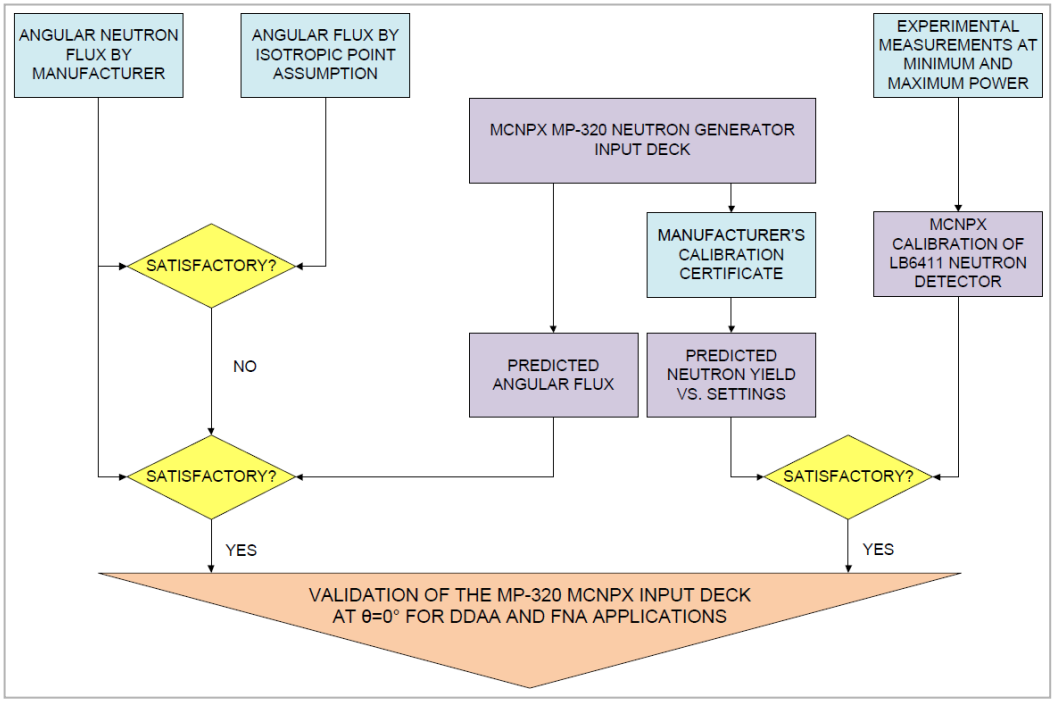


Figure 8 Workflow of the study.

At the current stage of the work, only the θ=0° geometry has been investigated because DDAA and FNA setups required the maximum neutron focal spot. Further experimental studies covering all the theta coordinates are foreseen in the future, in order to carry on another programme related to activation measurements.

The capability of the code allows the user to simulate also the behavior of the device vs. time when it is used in pulse mode, rather than in continuous neutrons emission. Moreover, it allows the user to run several trial simulations to optimize measurement technique's geometry and parameters, powering the generator only to check some correspondence between simulated and measured values at some key-stages of the technique's optimization process. Feasibility studies and sensitivity analysis can be implemented in a systematic way, saving precious device's lifetime.

The procedure shown before is very versatile. It can be applied to whatever neutron generator in order to retrieve a working MCNPX input deck to be implemented for any users' needs. Currently, at ENEA Casaccia Research Center, the realized MCNPX input deck is available for all the experiments that foresee the use of the MP-320. Further, it will be used in future works on

417 waste assessment and CBRNE (Chemical Biological Radioactive Nuclear and Explosive)
418 activities [17].
419
420
421 **REFERENCES**
422

[1] U. Wiacek and J. Dankowski, "Monte Carlo modeling of distributions of the d-d and d-t reaction products in a dedicated measuring chamber at the fast neutron generator," *Nuclear Instruments and Methods in Physics Research Section B: Beam Interactions with Materials and Atoms,* vol. 349, pp. 96-102, 2015.

[2] J. Katalenich, M. Flaska, S. A. Pozzi and M. R. Hartman, "High-fidelity MCNP modeling of a D-T neutron generator for active interrogation of special nuclear material," *Nuclear Instruments and Methods in Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipments,* vol. 652, no. 1, pp. 120-123, 2011.

[3] J. W. Hayes, E. Finn, L. Greenwood and R. Wittman, "Characterization of a Thermo Scienfitic D711 D-T neutron generator located in a low-scatter facility," *Nuclear Instruments and Methods in Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipment,* vol. 741, pp. 57-66, 2014.

[4] Thermo Scientific, "MP 320 Neutron Generators," Thermo Scientific, 2015. [Online]. Available: http://www.thermoscientific.com/content/tfs/en/product/mp-320-neutron-generators.html.

[5] A. C. Raoux, A. Lyoussi, C. Passard, C. Denis, J. Loridon, J. Misraki and P. Chany, "Transuranic waste assay by neutron interrogation and online prompt and delayed neutron measurement," *Nuclear Instruments and Methods in Physics Research Section B,* vol. 207, pp. 186-194, 2003.

[6] A. Buffler, "Crontraband detection with fast neutrons," *Radiation Physics and Chemistry,* vol. 71, no. 3-4, pp. 853-861, 2004.

[7] J. F. Briesmeister, "MCNP – A general purpose Monte Crlo code for neutron and photon transport," Nov 1993.

[8] M. Pillon, M. Angelone, M. Martone and V. Rado, "Characterization of the source neutrons

produced by the Frascati Neutron Generator," *Fusion Engineering and Design,* no. 28, pp. 683-688, 1995.

[9] A. Milocco, A. Trkov and M. Pillon, "A Monte Carlo model for low energy D-D neutron generators," *Nuclear Intruments and Methods in Physic Research Section B: Beam Interaction with Materials and Atoms,* vol. 271, pp. 6-12, 2012.

[10] F. Prezzavento, *Private communication.*

[11] Thermo Scientific, *MP320 Neutron Generator - Operation Manual.*

[12] IAEA, "Neutron Generators for Analytical Purposes," Vienna, 2012.

[13] G. Ponti, "The role of medium size facilities in the HPC ecosystem: the case of the new CRESCO4 cluster integrated in the ENEAGRID infrastructure," in *2014 International Conference on High Performance Computing and Simulation*.

[14] ENEA, "ENEA CRESCO Website," [Online]. Available: http://www.cresco.enea.it/english for information..

[15] Thermo Scientific, *Thermo Scientific MP-320 Neutron Generator calibration certificate.*

[16] "ENEA, Italian National Agency for New Technologies, Energy and Sustainable Economic Development, website," [Online]. Available: http://www.enea.it/it?set_language=it&cl=it.

[17] "End-user driven DEmo for cbrNE," [Online]. Available: https://www.eden-security-fp7.eu/.

423

424