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SIMULATIONS FOR THE NEUTRON DETECTOR **TETRA** WITH **MCNP**

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Тестов Д., Кузнецова Е., Вильсон Д. E15-2013-128
Моделирование нейтронного детектора ТЕТРА
с использованием кода MCNP

Для изучения структуры нейтроноизбыточных ядер, получаемых на ускорительном комплексе АЛТО (ИЗОЛ), ИЯФ (Орсэ), в коллаборации с ОИЯИ (Дубна), был усовершенствован нейтронный детектор высокой эффективности ТЕТРА, состоящий из 90 счетчиков, заполненных ^{3}He. Для оптимизации новой конфигурации детектора ТЕТРА были проведены расчеты с использованием кода MCNP. Обсуждаются детали и результаты моделирования.

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Simulations for the Neutron Detector TETRA with MCNP

To study the nuclear structure of β-delayed neutron precursors at ALTO ISOL-facility at IPN (Orsay), the high efficiency 4π neutron detector TETRA with ^{3}He filled counters built at JINR (Dubna) was modified. The MCNP simulations to optimize the future configuration were necessary. The details of the calculations and the major results obtained are discussed.

The investigation has been performed at the Flerov Laboratory of Nuclear Reactions, JINR.

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1. INTRODUCTION

Nuclei located in the neutron-rich side of the chart of nuclides are weakly bound due to large N/Z ratios. They have relatively short half lives and β decays back to the stability. In case of large Q_β values, β-decay daughter can be left in excited state whose energy is above the neutron separation energy S_n. The nucleus further deexcites either by γ emission to lower states or by the neutron emission. The probability to occur for this process is higher for nuclei with large neutron excess in vicinity of neutron closed shells. With new opportunities offered by radioactive beam facilities the serious experimental efforts are devoted to investigate properties of such nuclei [1]. Due to typically large Q_β values, a β-decay daughter can be left in an excited state above the neutron separation energy. In such a case, a nucleus may further deexcite by neutron emission. Measurements of the β-delayed neutron emission probability P_n are important for nuclear structure studies since it directly shows the probability to populate certain level above the neutron separation energy and can serve certain model constraints [2]. Neutron-rich nuclei play therefore a key role in the astrophysical rapid capture neutron process. The r-process constitutes one of the major processes in which elements heavier than iron are formed. It consists of a series of rapid neutron captures followed by β decays and passes through a net of nuclei with large Q_β values far from stability. The position of the r-process depends on nuclear structure properties and stellar conditions under which it occurs: temperature, density, and duration of the neutron flux. Magic neutron numbers play a special role in the r-process after freeze out of the neutron flux, these nuclei β decay bringing the process back towards the line of β stability. Since just few of r-process nuclei are accessible experimentally, most of the parameters needed for r-process calculations are derived from theoretical models. Studying β-decay properties of the neutron-rich nuclei accessible nowadays at neutron closed shells $Z = 50$, $N = 82$ is highly demanded by astrophysics [3,4].

To extend investigation of neutron-rich nuclei produced at ALTO ISOL-facility [5, 6], in the framework of JINR (Dubna) – IPN (Orsay) collaboration, 4π high-efficiency neutron detector TETRA with ^{3}He filled counters was modified [7, 8]. Here we report the MCNP simulations performed to optimize the configuration of TETRA to match the new requirements.

2. TETRA NEUTRON DETECTOR

The type of neutron detector employed for the specific physical task is mostly determined by the energy range of neutrons of interest and the information on a neutron event required. Thus, for β-decay studies a ^{3}He filled detector seems to be the most preferable due to its high efficiency, though the information on neutron energy obtained by these detectors are usually poor. In ^{3}He detector a neutron is registered due to registration one or two products of the reaction: $^3\mathrm{He} + n \rightarrow {}^3\mathrm{H} + {}^1\mathrm{H} + 765$ keV. The probability of a neutron to interact with ^{3}He is much higher when a neutron has a thermal energy. That is why, a neutron detector is placed in high-density moderator in order to slow down neutrons.

The TETRA is a gas filled neutron multidetector constructed at JINR, Dubna. It consists of 90 counters 500 mm length and 32 mm in diameter filled by ^{3}He at 7 atm pressure with 1% admixture of CO_2; and built-in preamplifier at the end of each tube [9]. The optimized by MCNP configuration is presented in Fig. 1. The counters are placed in four raws in a single peace of moderators and arranged so that the distance between their centers is 5 cm. The central cavity (Fig. 1) is made to be 13 cm to accommodate a beam line with a $4\pi\,\beta$ detector inside and a germanium detector. The thickness of borated (5%) polyethylene shielding

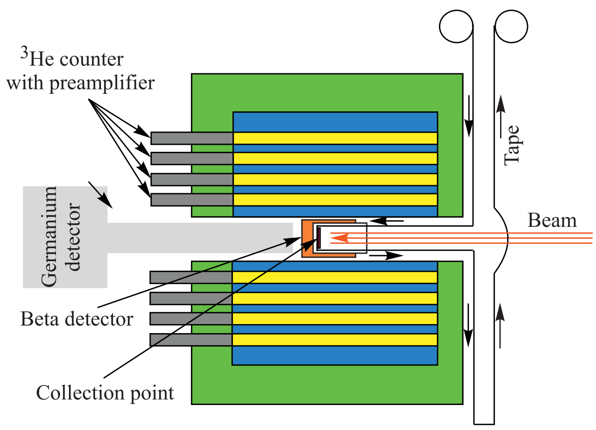

Fig. 1. The schematic view of TETRA setup

is 150 mm all around the detector. The collection point is located on the mylar tape. Since β- and $\beta{-}n$-decay daughters of the source accumulated by the beam at the collection point are beta decay further to stability, the tape is moved with a period defined by the half life of the nucleus of interest. The further subsections describe the model developed and present the major result obtained.

3. MONTE CARLO N-PARTICLE (MCNP) TRANSPORT CODE

Though the manual given in [10] is full and comprehensive, illustrated by many useful examples, from our point of view, it is challenging for new users to differentiate between information needed to learn how to use the code and the information specified for more complicated cases. We found the tutorial in the reference [11] covered better practical problems which beginners can face. Some of the notations in the MCNP documentation use original terminology. Thus, the term card, historically a punched card, should be interpreted as a line of the input file. Briefly, to prepare an input file, one should include the Geometry definition card (Subsec. 3.1), Materials cards and corresponding cross section tables (Subsec. 3.2), the Source specification card (Subsec. 3.3), the physical process to be studied — reaction $^3\mathrm{He}+n$ in our case, and the Results required — tally cards (Subsec. 3.4).

3.1. Geometry Specification. In MCNP a cell is a volume bounded by set of surfaces. However, to create too many objects specifying each time, a new volume could be very tedious that is why MCNP offers universes and lattices, which are able to repeat elements simplifying the users work. A universe is composed of all cells that are specified in it. A universe is either a lattice or an arbitrary collection of cells. Lattices are repeated structures, basically, equivalent to lattices in a reactor and can be rectangular or the hexagonal.

TETRA has the shape of hexagonal prism to minimize the corner effects. Naturally, the hexagonal repeated structure was used in building of the model. The main virtual neutron «bin» in the simulation is a hexagonal prism defined by the universe $52\,(u = 052)$ which includes a steel tube, inner gas volume filled with ^{3}He and 1% of admixture of CO_2, and a hexagonal piece of moderator so that the distance between parallel sides is 5 cm. The same bin but filled differently by other materials represents complimentary bins listed in Fig. 2.

In the model it is logical to group neutron bins in four rows and operate during calculations only by these four raws instead of 90 independent bins, which can significantly shorten the computing time. The central cavity is simply described by the empty cell 500 (see Fig. 3). The first group (525) is a lattice-2 defined geometrically by its surfaces filled with virtual neutron bins to match exactly to the first row of counters of TETRA. Analogously, the cells 535, 545, 555 match, respectively, the 2nd, 3rd and 4th layers of the detector. The cell 565

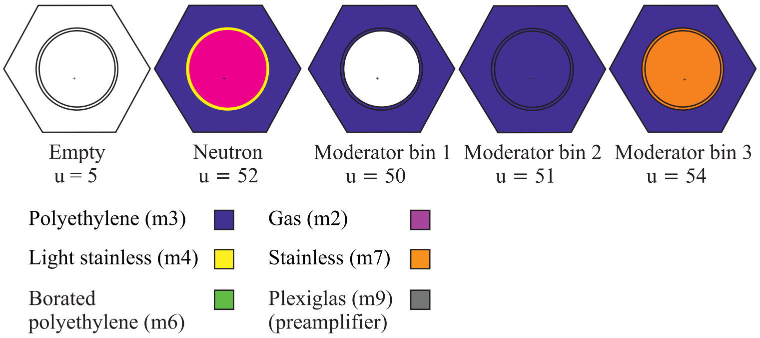

Fig. 2. Main bin filled by different materials from the Table

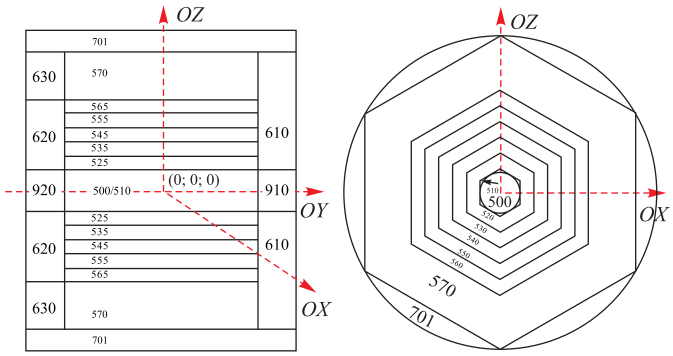

Fig. 3. Geometry cell specification

is made in the same way but contains only complementary (moderator) bins and corresponds exactly to the additional layer of moderator. The shielding all around is simply defined by cell 570 filled by borated polyethylene. During the experiments the neutron detector is put on a stainless table (cell 703) in an experimental hall with concrete walls (cell 705)*. Finally, cell 902 defines everything else. As was shown in the stimulation, the table and the walls give no noticeable effect on the results.

3.2. Material Specification. To specify materials filling, the various cells in an MCNP calculation involve: a) defining a unique material number, b) the elemental (or isotopic) composition, and c) the cross-section compilations to be used. The density is specified on the cell definition card. This permits one and the same material to appear at different densities in different cells.

*Cells 703, 705, 902 are not shown in Fig. 3.

The Table gives a list of materials used in the simulations. The isotopic composition for the most of them was found in the literature [12]. The contributions of other materials were considered as negligible and were not taken into consideration. Finally, the geometry cells presented in Fig. 3 are filled bins as is shown in Fig. 4.

The list of main materials used in the simulation. The isotopic composition can be found on cells *m2–m9*; **for (*) it was taken from [12]**

Card	Material	Density, g/cm^3
m2	^{3}He	0.00086
m3	Polyethylene	0.93*
m4	Light steel	7.93*
m6	Borated polyethylene, 5%,	1005
m7	Stainless still	7.93*
m8	Concrete LA	2.25*
m9	Plexiglas	1.18*

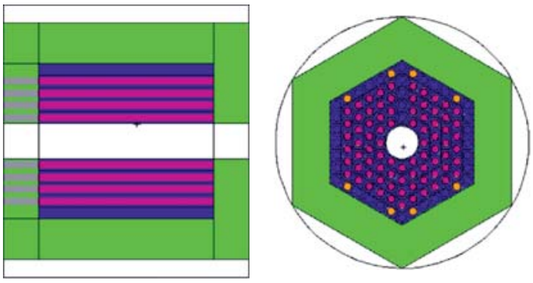

Fig. 4. Geometry cell specification filled with materials

3.3. Source Specification. The source and the type of radiation particles for an MCNP problem are specified by the SDEF card which has numerous variables or parameters to define all the characteristics of all sources in the problem. Only one SDEF card is allowed in an input file.

A point isotopic source used in the problem is defined on the SDEF card. The energies (E_n) of the neutrons emitted by spontaneous fission of ^{252}Cf (average neutron energy is 2.1 MeV) are typically described by the Maxwell–Boltzmann distribution function with an effective temperature of $kT = 1.42$ MeV.

3.4. Tally Specification. «Tally» in MCNP is a mean to specify the output results required. At least one tally card is required. The first entry on the card begins with F$id : TP$, id is the tally id number (the last digit of which determines the type of tally), and TP stands for N (neutron tally), P (photon tally), NP for joint neutron and photon tallies, and E for electron tallies.

The tally F4 (neutron fluency through a cell in n/cm^2) used with FM option is considered as the most appropriate to reconstruct the efficiency:

F4:N cell number (Cn)

FM4: C (constant), M (material), R (reaction).

MCNP multiplies the neutron fluency from the F4$_n$ tally (in n/cm^2) by the reaction cross section defined by M and R (in barns) and finally, the result is then multiplied by C. The most common use of C is to provide the atomic density of material. If it is given in atoms/barn/cm the result of the tally is reaction per source particle in 1 cm^3. With the volume of the Cn, also calculated by MCNP, the neutron flux through Cn is determined. The tally F4 is by default already normalized to the number of particles emitted by a source. There are five tallies in the solution: f524, f534, f544, f554 calculating neutron flux through 1st, 2nd, 3rd and 4th, respectively, and f54 determining the flux through the entire detector. We checked that the overall efficiency of TETRA is exactly the sum of efficiencies of each layer. All the tallies calculated passed all 10 statistical checks. Also the particle loss check was performed to avoid errors in geometry specification.

3.5. Validation of the Model. Reliability. The statistical error in MCNP is defined by the number of histories and can be as low as 1% (with 10^6 histories). However, the decisive role is played by systematic error arising from uncertainties of the model. To estimate the systematic error, the model of TETRA was validated on spontaneous ^{252}Cf fission source. In Fig. 5 is presented the efficiency of TETRA as a function of distance of ^{252}Cf source on the beam axis, from the center of the detector with zero coordinates in comparison to experimental data. The maximum efficiency is, obviously, at the center. Within the next 1–2 cm the efficiency is almost flat and gradually falls down with the increasing distance. As can be seen, the calculations regularly underestimate the experimental points.

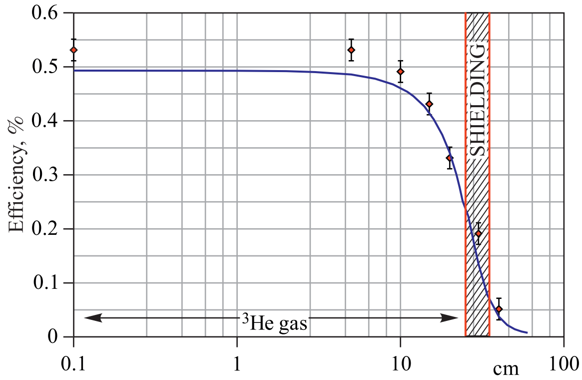

Fig. 5. Validation of the model: calculated and measured efficiency of TETRA as function of distance of the source from the center of the detector

Such a deviation originates from uncertainties in the modeling such as: neglecting minor materials and the fact that the exact composition of materials as well as exact densities can be known with certain accuracy; uncertainties in the geometrical dimension of the real parts of the detector. Nevertheless, the validation process shows that our model can be trustful within 6% of absolute error bars.

3.6. Gas Pressure and Moderator Density. Pressure of ^{3}He and density of moderator material are the most crucial parameters which influence the efficiency. As is shown in Fig. 6 an increase in the gas pressure, i.e., quantity of the gas, does not bring any significant positive effect above 10 atm. From our point of view, the saturation is reached about 7–8 atm. However, due to shortage of ^{3}He, nowadays it might be even more beneficial with the same quantity of the gas to construct more counters with 4 atm pressure of ^{3}He to make a detector bigger in size and, therefore, more sensitive to more energetic neutrons ($E_n > 1$ MeV) as is discussed in the next subsection.

The high-density polyethylene (0.93 g/cm^3) was used since it was considered as the most appropriate material for neutron moderation. Some other options could be paraffin and plexiglass, but usually their performances are slightly worse. As is illustrated in Fig. 6, presenting neutron efficiency as a function of neutron energy for the polyethylene used in a comparison to the lower density one (0.465 g/cm^3), the high-density polyethylene is a better performer.

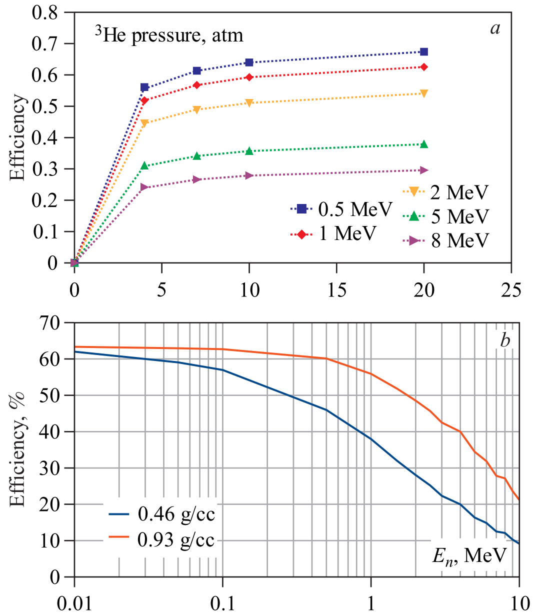

Fig. 6. Efficiency of the neutron detector as a function of: *a*) gas pressure, *b*) density of moderator

3.7. Efficiency as a Function of Neutron Energy. The efficiency has to be a constant in the energy range of neutrons of interest. Since energy of beta delayed neutrons is typically below 1 MeV [13, 14], the ^{3}He filled detectors seem to be the most suitable tool for beta delayed studies. The flatness of the efficiency can be reached by bigger number of raws with ^{3}He counters as was experimentally shown by the Dubna group [15]. Figure 7 represents the calculated efficiency of TETRA as a function of neutron energy in the range of $0.01 < E_n < 10$ MeV. The efficiency is flat almost up to 0.8 MeV. Simulations confirm that the inner row is more efficient to less energetic neutrons, while the outer one — to more energetic. The higher neutron energy is the longer path in moderator a neutron should pass to become a thermal one. In case of $E_n > 10$ MeV, neutrons can go through the detector almost without being registered.

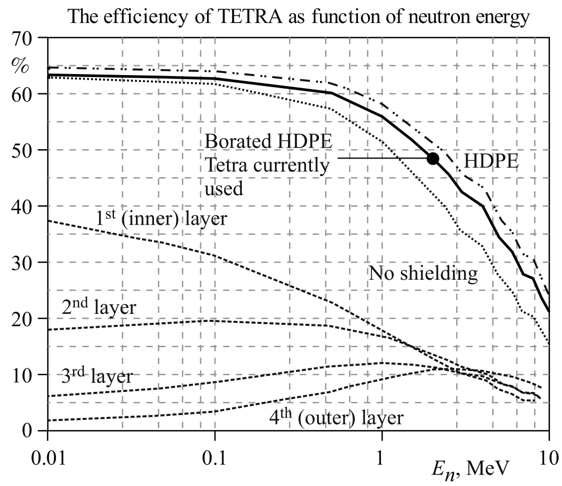

Fig. 7. Calculated efficiency of TETRA — function of neutron energy as a trend of different shielding: solid line — 15 cm of boron polyethylene (currently used); dashed line — without shielding; dash-dotted line — 15 cm of high-density polyethylene (HDPE). Dashed lines below are the efficiency of each layer

3.8. The Background Protection Qualitative Estimation. Under real experiment conditions, in the experimental room, several background sources are presented: beta delayed neutrons from decay of isotopes accumulated at the separator; neutrons from interaction of cosmic rays with material inside the room and inside the detector. Correctly defining realistic background neutron environment in MCNP is a very challenging task. Thus, it was decided to use one of the standard available MCNP sources in order to proceed the qualitative estimation of the shielding.

Cosmic rays which interact with materials inside/outside of the detector give an experimentally measured constant count rate as low as 5–10 n/s without shielding (and later measured 1 n/s with shielding installed).

Background source of neutrons from beta decay of fission fragments were approached as a neutron cloud — neutrons without dominated energy or direction. This approximation is valid due to low beam extraction potential of 30 keV, pie shield applied around the separator (polyethylene, cadmium, lead). However, the major source of neutron background during experiment was expected to come directly through the beam line together with a beam. To minimize the effect, the detection system was located at a beam line which made 90° angle with the separator. Even though, this source was believed contributing much stronger in comparison to «neutron cloud» or cosmic rays.

Under the assumptions above, a point source collimated into a cone direction, injected 60 cm from the center of the detector on its axis (OY), so that the cone passed through the edges of the detector, was considered as a background source approximation. Calculated efficiency of TETRA for such a source for different shielding configurations was normalized to the efficiency obtained with zero shielding applied (Fig. 8).

The shielding had not to exceed 15 cm width layer due to detector geometrical configuration. Although, the application of high-density polyethylene (HDPE) shielding only rejects relatively well the neutron background for $E_n < 0.5$ MeV, it increases the impact of higher energetic part due to neutron moderation and consequent increasing probability to interact with ^{3}He (Fig. 8, dotted line). In its turn, the polyethylene shelter of the same widths with 5% admixture of boron deals much properly with neutron energy range $0.01 < E_n < 10$ MeV (Fig. 8, dash-dotted line). Eventually any background shielding positively impacts the overall neutron detection efficiency of TETRA for the effect neutrons emitted

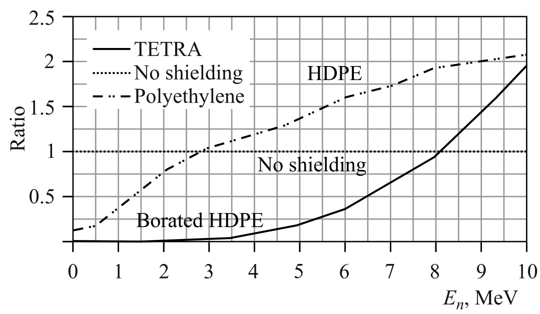

Fig. 8. Background estimation. Efficiency of TETRA as a function of neutron energy for the background source described in the text. The HDPE and borated HDPE curves are normalized to the non-shielding one

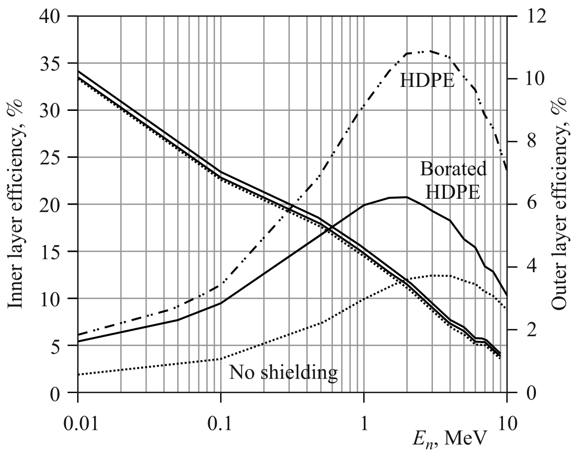

Fig. 9. Effect of the background protection on the efficiency: comparison of efficiency of inner and outer layers as a trend of different shielding, the same as in Fig. 7

from a source at the center of the detector as is shown in Fig. 9. It is due to the fact, that neutrons which already have passed the detector without interaction with ^{3}He gas, can be reflected by the shielding back and, finally, be detected. As is seen in Fig. 8, the application of shielding does not disturb much the efficiency of the first layer, whereas it kicks it up for the outer one.

4. SUMMARY

The 4π high efficiency neutron detector TETRA built at JINR, Dubna has been recently modified for on-line measurements at ALTO ISOL-facility (Orsay) to investigate β-decay properties of neutron-rich nuclear species. The unique design of the detection system performed was imposed by short half lives of the nuclei of interest, relatively low production yield and the necessity to measure simultaneously beta, gamma and neutron activity of the source accumulated by the beam on the tape at the center of the detector. The MCNP simulations were performed to find the most optimized configuration for the TETRA. Study of the TETRA efficiency was performed with ^{252}Cf spontaneous fission source based on multiplicity distribution of prompt neutrons. The specific electronic and «dead timeless» data acquisition system to register neutron events of high multiplicity were developed.

The presented experimental setup allows measuring absolute branching ratios in β decay; half lives of neutron precursors by neutron activity curve. Also it was found the practical use to serve γ-spectroscopy studies by $\gamma{-}\beta{-}n$ coincidence. Currently the setup is based at ALTO to investigate structure and absolute branching ratios of neutron rich r-process nuclei in vicinity of closed neutron shells $N = 50$, $N = 82$. The TETRA can be used at any present or future available facilities of exotic beams production, primarily at SPIRAL-2.

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