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NEUTRONIC DESIGN OF AN ACCELERATOR DRIVEN SUB-CRITICAL RESEARCH REACTOR

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ABSTRACT

Conceptual design of an accelerator driven sub-critical research reactor (ADSRR), as a new project in the Vinča Institute of Nuclear Sciences, is suggested for support to the Ministry of science, technologies and development of Republic Serbia, Yugoslavia. This paper show initial results of neutronic analyses of the proposed ADSRR carried out by Monte Carlo based MCNP and SHIELD codes. According to the proposal, the ADSRR would be constructed, in a later phase, at high-energy channel H5B of the VINCY cyclotron of the TESLA Accelerator Installation, that is under completion in the Vinča Institute. The fuel elements of 80%-enriched uranium dioxide dispersed in aluminium matrix, available in the Vinča Institute, are proposed for the ADSRR core design. The HEU fuel elements are placed in aluminium tubes filled by the "primary moderator" – light water. These "fuel tubes" are placed in a square lattice within lead matrix in a stainless steel tank. The lead is used as a "secondary moderator" in the core and as the axial and radial reflector. Such design of the ADSRR shows that this small low neutron flux system can be used as an experimental "demonstration" ADS with some neutron characteristics similar to proposed well-known lead moderated and cooled power sub-critical ADS with intermediate or fast neutron spectrum. The proposed experimental ADSRR, beside usage as a valuable research machine in reactor and neutron physics, will contribute to following and developing new nuclear technologies in the country, useful for eventual nuclear power option and nuclear waste incineration in future.

1. INTRODUCTION

Proposal for new research project for design of an ADSRR [1] in the Vinča Institute of Nuclear Sciences in Yugoslavia is based on the TESLA Accelerator Installation, abundant experience accumulated during past 40 years from design, operation and maintenance of research reactors and at stock of available fresh HEU uranium fuel elements. The basic aim of the project is study of modern reactor physics and development of the technologies needed to design a small, fast or thermal, sub-critical low neutron flux reactor system driven by neutrons generated in a target by an accelerator beam. This type of small demonstration ADSRR will be designed in such way to simulate some of neutronic characteristics of world known proposed high-power ADS with intermediate and fast neutron spectrum (e.g. Energy Amplifier, by Carlo Rubia) and so respond to world-wide requests for urgent needs for experimental verification of nuclear parameters of proposed ADS. This new inherently safe sub-critical reactor could be also a focal point to concentrate and safe for future, now already very small, group of researchers in the field of the reactor physics in the country.

Construction of nuclear power plants is forbidden in Yugoslavia by federal "nuclear" law, brought in the Parliament in 1989 and 1995. Due to the law, it is not possible even to design a full scale ADS for power production or incineration of radioactive waste and trans-uranium nuclides. Total quantity of spent nuclear fuel is low (about 2.5 tons of LEU fuel). This spent nuclear fuel is entirely resulting of operation of the RA research reactor in the Vinča Institute of Nuclear Sciences during period of 1960-1984. Thus, economy reasons do not support design and construction an ADS, either for partitioning and transmutation of radioactive waste and trans-uranium nuclides, or a plant, even on a laboratory scale, for processing of spent nuclear fuel. According to the nuclear law, such activities are also forbidden in the country; only research in nuclear field is allowed.

However, actual unfavourable economic situation in Yugoslavia and current shortage of about 30% of electrical energy, open the question of realistic sources of electrical energy for future. In that view, nuclear power plants for electricity production may become, beside import of electrical energy, a real option in prospective of the country.

Activities at the conceptual design of the proposed ADSRR are seen as a three-phase project in three years with participation of experts from the Vinča Institute and various scientific and R&D organisation from country and abroad. Purpose of this paper is to give some results of neutronics design parameters of the proposed conceptual design of the ADSRR obtained through the initial feasibility study [2].

2. INITIAL CONCEPTUAL DESIGN

Preliminary initial conceptual design of ADSRR is based on neutronic calculations of a sub-critical system with HEU fuel elements, moderated by light water in square lattice within lead matrix [2, 3]. Neutron source is obtained from interaction of beam of charged particles extracted from H5B high-energy channel of the TESLA Accelerator cyclotron.

The TESLA Accelerator Installation in the Vinča Institute is in the final phase of construction. It is a multi-purpose facility for production, acceleration and use of ions. Construction of the installation began in 1989 and comprises a compact isochronous cyclotron - the VINCY Cyclotron; an electron cyclotron resonance heavy ion source the - mVINIS Ion Source; a volume light ion source - the pVINIS Ion Source, and several low and high-energy experimental channels. The ion sources are already in operation, while the first beam extraction from the VINCY cyclotron is scheduled in next 2.5 years.

Designed parameters of the VINCY cyclotron of the TESLA accelerator are set more than 10 years ago. These parameters were optimised for extraction of deuteron beam. These parameters are not favourable in respect to driving an ADS for energy production or transmutation of trans-uranium nuclides. The cyclotron can deliver either protons with maximum energy of 75 MeV and current of 5 μA, or deuterons with maximum energy of 73 MeV and current of 50 μA. Thus, one of the main tasks in the project will be theoretical and experimental examination of interaction of the beam particles with different materials in order to choose and design an optimal target in respect to escaping neutron spectrum and neutron yield.

The available HEU fuel elements in the Vinča Institute of Nuclear Sciences, in a form of the ‘slugs’, are produced in ex-USSR in 1975. They are in current utilisation at both research reactors in the Vinča Institute, since 1976. A view of the HEU fuel slug is given in Figure 1. Geometry shape and material composition of the HEU fuel slug are described elsewhere in details, e.g., [4]. The fuel layer (2 mm thick) has inner/outer diameter (ID/OD) 31/35 mm and its total length is 100 mm. It is manufactured as 80%-enriched UO_2 dispersed in aluminium matrix. The fuel layer is covered by aluminium cladding on both sides, 1.0 mm thick on its inner side and 1.1 mm thick on its outer side. Top and bottom of the slug are covered by the 'stars' so that the total length of the slug is 113 mm. Inside the slug, an 'Al expeller' is designed as a hollow cylinder tube that acts as a coolant flow adapter during forced cooling (used, e.g., at the RA reactor). The aluminium, used in the HEU fuel slugs, is known in Russia as the SAV-1 alloy (0.985 weight fraction is aluminium with very low contents of neutron high-absorbing impurities, e.g., boron or cadmium).

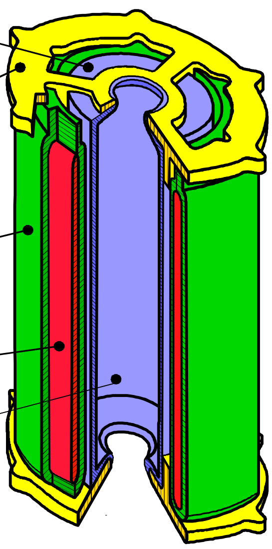

Figure 1 HEU Fuel Slug

In order to form a ‘fuel element’ of the ADSRR, six HEU slugs are placed one above the other in an aluminium tube (ID/OD 41/43 mm) filled by ordinary (light) water. The fuel elements are placed in holes (45 mm diameter) drilled in a lead matrix of the sub-critical reactor core. The holes form a regular square lattice (11 x 11 matrix) with 50 mm pitch. Design is carried out in such a way that any fuel element could be replaced, if necessary, either by a vertical experimental channel (aluminium tube, ID/OD 41/43 mm), or by a reflector element designed as a cylindrical lead rod (OD 43 mm). Light water is proposed for the primary moderator inside the fuel elements in aim to achieve dominant thermal energy neutron fission rate in the core.

The lead matrix is used as moderator and reflector of the core in aim to accomplish simulation of a intermediate energy neutron spectrum similar to ones in a high-power ADS. Radial reflector has average thickness about 210 mm. Top and bottom axial reflectors are either 113 mm or 226 mm thick, each, simulating height of one or two HEU fuel slugs, respectively. Total height of the core with both (113 mm thick each) axial reflectors is

924 mm. In case of usage of axial lead reflectors with the total thickness equivalent to height of two HEU fuel slug, the total height of the reflected core is 1140 mm. The proposed core configuration consists of 97 fuel elements, each filled with six HEU fuel slugs, except the central one that has only three lower HEU fuel slugs. Entire core is designed in a stainless steel cylindrical tank that has inner diameter 1000 mm and 1200 mm total height. Cylindrical wall and bottom of the stainless steel tank have thickness of 25 mm. The total mass of the system is between 7.5 tons and 11.0 tons, depending on thickness of the axial lead reflectors.

Particle beam, extracted from the H5B channel of the VINCY cyclotron, is introduced to the sub-critical core by a separate stainless steel tube (ID/OD 27/29 mm) under high vacuum. It is supposed that beam tube enters the sub-critical system through the top surface. An idealised target, a metal cylinder (25 mm high and 25 mm diameter), is placed at the window of the beam tube, near the centre of the sub-critical core. Neutron yield and neutron spectra from different target materials are calculated for various energies (10 MeV - 75 MeV) of an incident beam of protons or deuterons [5]. For that purpose Monte Carlo based codes, the SHIELD [6], developed in Russia, and the Lahet Code System – LCS [7], developed in USA, are used.

3. ADSRR MODEL AND CALCULATION

In aim to estimate escape neutron spectrum and neutron yield for this design option of the ADSRR, an incident infinite thin ('pencil') proton or deuteron beam, with energy of 75 MeV, is assumed to hit the top surface of the lead cylindrical target along the cylinder axis. It is obvious that neutron yield is the highest for the highest beam energy. Results obtained from calculation of interaction of the beam with the target material in idealised conditions (surrounded by vacuum), by the SHIELD code are used in this study. The SHIELD code is validated against many experimental benchmarks and in several inter-comparisons of high-energy hadron codes. Hadron-nucleus interactions inside the target material are simulated on a basis of the known Russian models of nuclear reactions, jointed in the generator of nuclear reactions - the Multi Stage Dynamical Model (MSDM). The transport of neutrons in material of the target with energies less than 14.5 MeV is simulated using 28 multi-group ABBN neutron data system.

It is shown [3] that, for a number of protons with initial energy of 75 MeV that enter to the selected Pb target, near 95% would avoid inelastic interaction inside material of the target. Neutron spectra for incident proton and deuteron beam, leaking from the Pb target, after one million incident particles, are very similar [2, 5]. Total yield of escaping neutrons from the target surfaces is about 0.19 neutron/proton in case of incident 75 MeV proton and about 0.15 neutron/deuteron in case of incident deuteron with 75 MeV energy [2, 5]. Only about 5% - 6% of these outgoing neutrons have energy in range 14.5 MeV - 75 MeV. In the code, neutrons are collected in energy groups with equal bin width of 3.75 MeV, for 75 MeV incident protons. Below energy of 14.5 MeV, neutrons are collected in 28 energy bins with energy borders equal to the structure of the ABBN-78 multi-group data library. Peak of the outgoing neutron spectrum is in energy range from 0.2 MeV to 0.8 MeV. It was shown that escaping neutron spectrum does not contain neutrons with energies less than 0.465 keV, i.e., there is no thermalisation of neutrons within the target material. This is consequence of a simplification made in the simulation. The idealised cylindrical target is placed in the vacuum and return effect of slowing-down and thermalised neutrons in surrounding media of the sub-critical core of the proposed ADSRR is not encountered in the interactions within target and in outgoing source spectrum.

The well-known MCNP code is used for preliminary criticality calculations of the ADSRR with neutron data library developed in the Vinča Institute, based primary on the ENDF/B-VI.2 evaluation file. Scattering of thermal neutrons at hydrogen atoms connected in light water molecules is encountered by the $S(\alpha,\beta)$ scattering law, according to the standard TMCCS library of the MCNP code. Three-dimensional model of the proposed ADSRR for the MCNP code is developed. The HEU fuel slugs are introduced as the 3D model of the real HEU slug developed and verified [4] for use with the MCNP or SCALE code.

To specify the neutron source (SDEF), in the first step of the calculations carried out by the MCNP code, neutrons escaping the target volume with group spectra obtained in the SHIELD code are used. Minor modifications are done due to constrain in the highest neutron energy (20 MeV) used in the MCNP neutron data library. All neutrons above 20 MeV (less than 5.0 % of total yield from protons are added to the two the highest neutron energy groups (with boundaries: 14.5 MeV – 18.75 MeV – 20 MeV). The code is run for five thousands neutron histories and the four-group spatial distribution of neutron flux and fission rate source, in cells with HEU fuel slugs of the core, is determined with 1σ error between 7% (near centre) and 20% (system edge). Energy boundaries of these four 'macro' groups for neutron distributions are defined as following: (1) thermal energy group for $E_n < 0.465$ eV; (2) epithermal energy group for 0.465 eV $< E_n <$ 1 keV; (3) intermediate energy group for 1 keV $< E_n <$ 0.8 MeV, and (4) fast energy group for 0.8 MeV $< E_n <$ 20 MeV. For determination spatial distribution of the fission rates in the four energy groups, the F4 tally with integration over fission cross-section by appropriate cell volume and neutron energy is used in the MCNP code.

The spatial distribution of fission neutron source, obtained by the method described, is used for the subsequent MCNP calculations of the neutron effective multiplication factor (KCODE option) in the ADSRR. In this (second) step of calculation by the MCNP code, 1500 neutron active cycles are run with 1,000 neutrons per cycle, after 20 initial ones. Simultaneously, the neutron spectra in various cells of the ADSRR lattice are calculated.

4. RESULTS OF NEUTRONIC STUDY AND DISCUSSION

Values obtained for the average effective neutron multiplication factor (k_{eff}) and the prompt removal neutron lifetime (l_p) in the ADSRR system, for two cases of thickness of axial (top and bottom) Pb reflectors, are shown in Table 1 including statistical uncertainty (1σ) for 0.67 probability. The result for the second case is not acceptable due to safety reasons, because maximum recommended *keff* value for an ADS is 0.98. This value could be achieved in the second case of the proposed ADSRR system by removing one layer of the top Pb reflector.

Table 1. MCNP code: Values for k_{eff} and l_{p} in ADSRR
for different thickness of axial (top and bottom) Pb reflectors

Axial Pb reflectors thickness [cm]	$k_{\mathrm{eff}} \pm 1\sigma$	$l\mathrm{p} \pm 1\sigma$ [μ s]
2 x 12.3	0.9709 ± 0.0006	84.9 ± 0.1
2 x 24.6	0.9865 ± 0.0006	91.5 ± 0.1

Multi-group neutron spectra calculated in few various lattice cells of the proposed ADSRR are shown in Figure 2. It could be easily seen that request for dominant intermediate neutron spectrum in the core was met by proposed design.

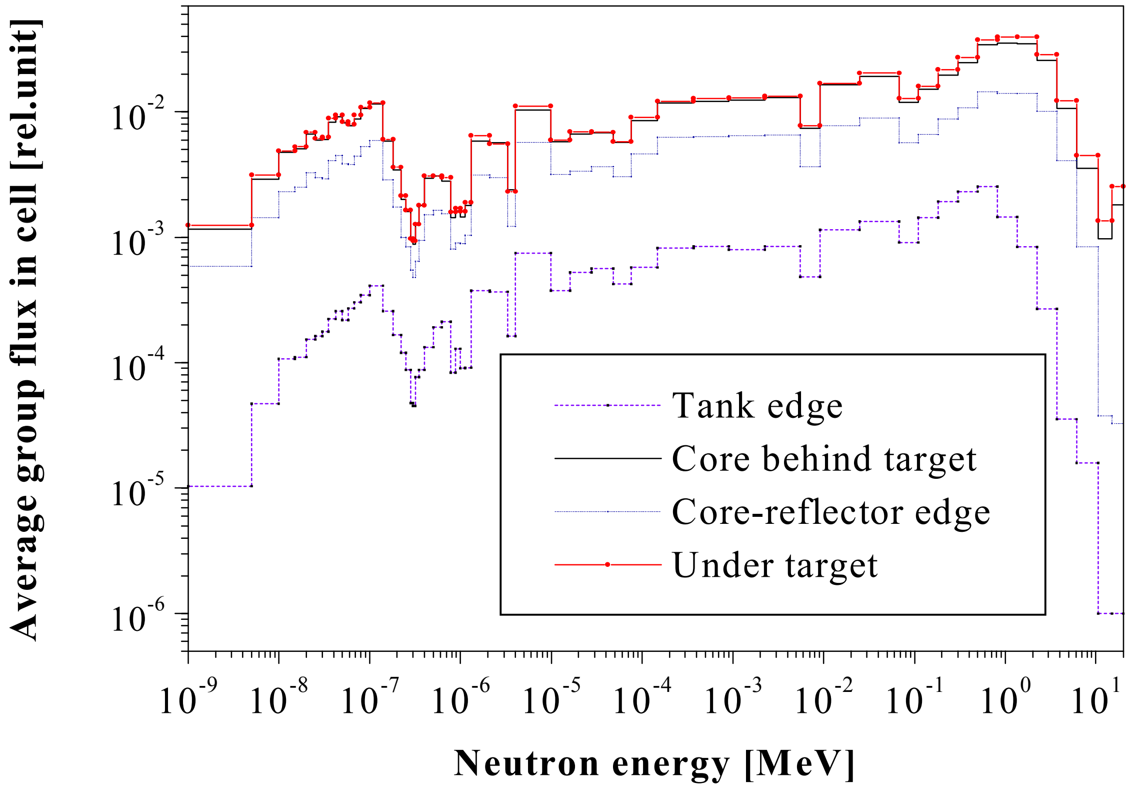

Figure 2. MCNP code: Neutron spectra in various zones of the ADSRR

Calculated relative distributions of 'macro' four-group neutron flux and fission rate in few lattice cells of the system, in case of 75 MeV proton incident beam and axial Pb reflector of 12.3 cm, are given in Table 2. The origin of (x, y, z) co-ordinate system is set at bottom of the central vertical column of fuel elements.

The total neutron yield from the selected Pb target, for the beam of 75 MeV protons with current of 5 μA, is about 5.85 10^{12} neutrons per second, i.e., the average neutron flux leaking the target surfaces is ~2.0 10^{11} cm^{-2} s^{-1}. The MCNP calculations show (Table 2) that neutrons in the system are generated dominantly by thermal fission (E_n < 0.465 eV) in ^{235}U. In two-group representation, fraction of 'all fast' fission (E_n > 0.465 eV) is only 6% - 8% of total number of fission, depending on position of the fuel slugs in the core (see Table 2). Average neutron flux in the lattice cells with HEU fuel slugs is dominantly fast (E_n > 0.465 eV) and about 4 times higher than the thermal one (see Table 2 and Figure 2). This is consequence of small amount of light water in the core and low energy decrement per neutron collision with lead atoms. According to the MCNP results, peak of fast energy neutron flux (E_n > 0.465 eV), for incident 75 MeV proton beam, is estimated, in the lattice cells near to the target, at value of 7.4 10^{10} cm^{-2} s^{-1}, with corresponding thermal energy neutron flux value of 2.3 10^{10} cm^{-2} s^{-1}.

Total fast energy neutron flux (E_n > 0.465 eV) is about 6 - 7 times higher than thermal one (E_n < 0.465 eV) at reflector-tank edge of the un-shielded system (Table 2 and Figure 2).

Table 2. MCNP code: Relative distributions of four-group neutron flux and fission rate in proposed ADSRR in case of proton incident beam at Pb target

Cell id. [ix, iy, iz] and Cell centre (x,y,z) [cm] and Cell position	Group no.	Neutron flux	Fission rate
[1, 0, 3], (5.0, 0.0, 39.55) near to Pb target	1	0.1139	0.8425
	2	0.0906	0.0566
	3	0.1658	0.0063
	4	0.1106	0.0026
[5, 0, 3], (25.0, 0.0, 39.55) core-reflector edge	1	0.0566	0.4177
	2	0.0499	0.0295
	3	0.0751	0.0030
	4	0.0418	0.0010
[9, 0, 3], (45.0, 0.0, 39.55) reflector-tank boundary	1	0.0035	
	2	0.0061	
	3	0.0127	
	4	0.0027	

* Group energy boundaries defined in the text above

The values of the effective factor of neutron multiplication (Table 1), calculated by the MCNP code (0.971 and 0.987) are close to recommended values (0.95 – 0.98) to keep safely the sub-critical system in an inherent safe conditions with significant multiplication (from 20 to 50). In the second case of the proposed ADSRR configuration, with thickness of axial reflectors equal to two fuel slugs height, the *keff* could be reduced to acceptable safe value (less than or equal to 0.98), by removing one top layer (of two designed axial layers) of Pb reflector.

It is expected that *keff* factor in the real ADSRR will not deviate much from these calculated values, because of the reactivity effects of all components (including neutron absorption in construction materials and neutron reflection of surrounding shielding of the system) were not included in this initial study. This major system factor can be adjusted to desired value by proper using (e.g. disposition) of fuel elements and reflector thickness. In addition, calculated values of the prompt neutron removal lifetime (85 μs - 90 μs) shown in Table 1, and assumed fraction of delayed neutrons in the system with uranium fuel (about 0.67%), shows that the proposed ADSRR configuration can be controlled in a normal way of a thermal system, as it is done in a typical critical light water research or power reactor.

From the calculated spatial distributions of the fission rates in the ADSRR, the total (maximum) fission power of the proposed sub-critical reactor system is estimated to about 5 kW, for the maximum intensity of neutron source generated by 5 μA proton beam with energy of 75 MeV. Time and frequency of operation of the proposed ADSRR will depend primary on the constrain set in design that used fuel should not undergo any significant burn-up (i.e., the fuel should stay “fresh”) and on availability of the TESLA accelerator beam.

5. CONCLUSION

The basic aim of the proposed ADSRR Project is to study the physics and development of the technologies necessary to design a small sub-critical low neutron flux research reactor driven by an accelerator beam. The reactor will be used for basic and applied research in neutron physics, metrology, dosimetry and radiation protection, in radiobiology and for development of modern nuclear technologies. Initial results of the neutronic study show that it is possible to design a low neutron flux ADSRR with dominant fast neutron spectrum using fresh HEU fuel existing in the Vinča Institute and light water as primary moderator in lead matrix. That system is driven by a neutron source generated in a lead target by interaction of proton beam, extracted from the TESLA Accelerator Installation.

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REFERENCES

1. M. Pešić, N. Nešković, I. Plećaš, “ADS Project in the Vinča Institute”, *Proceedings of the International Conference on Sub-Critical Accelerator Driven Systems*, SSC RF ITEP, Moscow, Russia, (October, 11-15, 1999) pp. 27-33 (1999)
2. M. P. Pešić, “Research on ADS in Vinča Institute”, *Proceedings of the International Conference on 'Back-End of the Fuel Cycle: From Research to Solution' - GLOBAL 2001*, Paris, September 9-13, 2001, CD ROM, pp. 1-8 (2001)
3. M. Pešić, N. Sobolevsky, “ADS with HEU in the Vinča Institute”, *Proceedings of the 10th International Conference on Emerging of Nuclear Energy Systems - ICENES 2000*, Petten, The Netherlands (September 24-28, 2000), paper no. 067, pp. 420-428 (2000)
4. M. Pešić, “RB Reactor: Lattices of 80%-Enriched Uranium Elements in Heavy Water”, HEU-COMP-THERM-017, in *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, NEA/NSC/DOC(95)03/II, 2000 Edition, OECD/NEA, Nuclear Science Committee, Paris, France, **II**, pp. 1-191, (September 30, 2000)
5. M. Pešić, “Calculation of Neutron Production in Selected Materials by Beam of Charged Particles of Intermediate Energies”, Nuklearna Tehnologija, **XVI**, No. 1, pp. 34-38 (2001)
6. A. V. Dementyev, N. M. Sobolevsky, “SHIELD - Universal Monte Carlo Hadron Transport Code: Scope and Applications”, Radiation Measurements, **30**, p. 553 (1999)
7. R. E. Prael, H. Lichtenstein, “User Guide to LCS: The LAHET Code System,” LA-UR-89-3014, LANL, Los Alamos, N.M., revised Sept. 15, 1989 (1989)