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TUNGSTEN AND URANIUM TARGETS FOR THE MOSCOW MESON FACTORY

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ABSTRACT

This paper describes the concept of uranium targets of Moscow meson factory, reasons for choosing the geometry and material (U_3Si) of target as well as the shape of fuel elements. Results of neutronic computations, data on corrosion rates of design materials and measures for providing their compatibility are presented. The advantages of the chosen fuel composition as compared with others, design details and possible alternatives are discussed.

1. Introduction

The pulse neutron source based on the high-current proton linac (proton energy 600 MeV, average current 0.5-1.0 mA, pulse width 100-200 μs and repetition rate up to 100 Hz) and proton storage ring (pulse width 0.32 μs, average current up to 0.4 mA, repetition rate 100 Hz) /1/ that is under construction at INR RAS will allow one to develop a wide range of research on condensed matter and nuclear physics with thermal and epithermal neutrons.

Keywords: Target,Uranium,Tungsten,Corrosion

The neutron source consists of the following principal parts (see Fig.1):
- water cooled vessel containing target, moderator, reflector and top steel shielding with coolant pipes inside;
- gas vessel with vacuum bend reducing scattered neutron backgrounds at the lower moderator and joining the target and ion guide vacuum system;
- remote-controlled vacuum seal which provides disassembly of the whole neutron source;
- proton beam monitoring system.

The scheme allows us to use vessels with different targets and moderators (including cryogenic ones) and to replace all equipment in the central part, as well as to modify the source. A set of Ti-coated tungsten plates or tight lattice of natural uranium rods in stainless steel cans placed in Al-alloy vessel and cooled with light water are planned to be used as neutron targets /2-4/.

Additionally, we will be able, in future, to use enriched uranium targets with multiplication up to 10 without significant changes of the the existing target design / 5, 6/. Using the multiplying target will allow us:
- to increase by several times the intensity of the pulse neutron source at repetition rates from 10 to 50 Hz, which is the most suitable range for time-of-flight experiments;
- to decrease the beam current usage down to 5-50% of the total accelerator intensity, thus allowing more experimental programs to be executed concurrently;
- to use a relatively low proton current in the storage ring (< 0.5 mA); this will make easier the achievement of required operational parameters and will increase its stability;
to provide a high performance neutron source at the initial stage of accelerator operation, before the design parameters are achieved.

However, multiplying targets have high background caused by delayed neutrons. In order to decrease this background, it is necessary to use fissile isotopes with low fraction of delayed neutrons.

To use highly enriched uranium fuel in the multiplying target instead of natural uranium, it is necessary to decouple it from moderators in order to prevent the neutron pulse broadening and overheating of outer fuel elements by thermal neutrons. We are going to make the identical design for both natural and enriched uranium targets to provide for full-scale testing of all elements of the multiplying target during the natural uranium target operation.

It should be noted that the current design does not enable us to alter transverse core dimensions without replacing the whole central part of the source including the gas tank, the top shielding, and the target vessel. Moreover, altering the core height is limited by the design of neutron channels in shielding within the range of 6-12 cm (Fig.1) /2-4/. Therefore, to unify dimensions of the target vessel and reduce expences, the transverse dimensions of the core could not exceed the earlier adopted diameter of 32.8 cm. The required gain factor could be achieved by changing the height of the target core and upper and lower moderators. If the core height exceeds the vertical distance between neutron guides, which are 20 cm in diameter, additional collimators are needed to prevent direct leakage of the fast neutrons from the core to the

experimental samples. The part of the neutron source shown in Fig.2 includes the core, the lower and upper moderators, and the Be-reflector surrounding the upper moderator /5,6/.

The target design is close to the classical wing arrangement. Thus, if a powerful proton beam is used, it allows us:
- to minimize in a most easy and rational way the inevitably negative influence of several layers of material between the target and moderators, the inlet and outlet collectors, fuel element lattice, etc.;
- to provide reliable cooling of the proton beam window;
- to ensure nuclear safety of multiplying target and to use highly enriched uranium only in minimal quantity.

Other possible layouts (e.g. split target and flux-trap type moderators /7/) require more complex design. In particular, it would be necessary to provide cooling of three proton windows, to have several collectors, etc. In the case of a very high power target, the advantages predicted by simple physical models /8,9/ may completely vanish, as we have experienced with the reproduction factor when going from a simple fast neutron critical assembly without coolant and construction materials to a real breeder.

2. Tungsten and uranium elements

The tungsten target contains 25 plates made of W of high purity (99.95 wt.%) having dimensions of 5x7x160, 5x57x220, 10x57x160 and 10x57x220 mm. The design provides a 0.7 mm gap between the plates for coolant flow. The tungsten plates have a titanium corrosion resistant coating 10 μm thick. Without protection the rate of tungsten corrosion under operating temperature conditions may as great as 50 microns per year /10/.

The uranium target consists of cylindrical fuel elements with intermetallic substoichiometric composition in the form of U_3Si in stainless steel cans of 8 mm diameter and 0.3 mm wall thickness. The fuel element design was chosen for the following reasons. In case of a multiplying target, due to the high price of highly enriched uranium, it is important to provide that the lifetime of the target be at least 3-5 years, corresponding to fuel burnup 1-2 at.%. Moreover, during operation, the target will experience regular and irregilar thermal shocks. These negative phenomena are almost eliminated if the fuel is arranged in the target in the form of cylindrical rods /11,12/.

The triangular packing of the fuel rods, 8-9 mm diameter with a gap between them about 0.4 mm, makes possible single-phase water cooling of the target at a water pressure $(1.5\text{-}2.0)\times10^5$ Pa and flow rate of 3.5-4.5 m/s. In this case the volume fraction of the coolant does not exceed 20%, moreover it provides acceptable physical characteristics of the neutron source and allows removal of the heat power up to 2.0-2.5 MW at proton pulse repetition rates up to 50 pps /13-15/.

The choice of a high density uranium alloy for fuel core was made taking into account the properties, manufacturing technology and existing experience with their operation under

conditions similar to those of neutron target. The characteristics of two types of alloys are shown in Table 1: (1) alloys based on stable phases of the type of U_3Si /16/ and (2) alloys based on metastable solid solutions of the type of U-Mo and U-Nb-Zr /17,18/, both showing satisfactory resistance to corrosion in water up to 200°C. The maximum operating temperature of U-Mo and U-Nb-Zr alloys depends on the thermal stability of their corrosion resistant metastable γ-phases. After several hours at 350°C the beginning of γ-phase disintegration with the appearance on grain boundaries of an α-phase subject to corrosion was detected. This results in the loss of the desired fuel core mechanical properties and an increase of the corrosion rate by 10-100 times, up to unacceptable values 0.2-1.0 mg/cm^2hr. In several cases catastrophic destruction of U-Mo samples after 10-30 days of exposure in water was observed. Some aspects of corrosion failure were discussed in /19/.

Table 1. Characteristics of alloys

Alloy (fractions expressed as weight percents)	U-3.8% Si[1]	U-2.9% Si[1]	U-6% Mo	U-9% Mo	U-5%Nb-5%Zr
Density, g/cm^3	15.1	16.1	18.3	18.1	16.1
Uranium content, g/cm^3	14.5	15.8	17.1	16.5	15.1
Max allowable temperature, °C	550[2]	550[2]	350	350	400
Corrosion rate in water, mg/cm^2hr:					
at 100°C	0.01	0.02	0 01	0 01	0 01
at 200°C	0.11	0.07	0.03	0 01	0 04
at 200-350°C (after annealing at 350°C)	0 5	0 5	1-2	0.5-1	>5
Thermal conductivity at 200°C, W/m·K	19	21	23	21	22

[1] **Contains 0.2-0.4 % of elements stabilizing properties of U_3Si.**

[2] **Recommended temperature, tested up to 680°C.**

Under conditions of frequent changes of the operation regime, shut-downs, uneven heating (thermal shocks ~10·15°C in the case of multiplying targets) it is impossible to provide reliable thermal contact between the can and fuel rod based on U-9% Mo. Experience with U-Mo alloys in breeder blankets with burnup ≤1% shows that the thermal contact is not always achieved even in the case of rare changes of operating conditions and shut-downs. In Fig.3, for targets based on ^{238}U, ^{235}U and ^{233}U, the expected temperature in the centre of the most thermally stressed fuel rods made of U-9% Mo and ^{233}U-2.9% Si alloys is shown as a function of multiplication coefficient at different values of the thermal gap betwen fuel and can, and different pulse repetition rates. For U-9% Mo alloy, the technological gap between the fuel rod and can is varied from 0.025 to 0.05 mm. Due to amorphization (discussed below) U-2.9% Si alloys provides full thermal contact between the fuel and can (δ=0). Plotted on the same graph

is the curve of the stability of the γ-phase of metastable alloys. As it follows from Fig.3, for the proton beam of the Moscow meson facility, U-9% Mo alloy can be used for the targets with low multiplication coefficients ~3-5 at pulse repetition rates ~25-50 pps, respectively. At larger multiplication coefficients, the temperature of both central and peripheral fuel elements exceeds the temperature of γ-phase stability.

For fuel based on uranium silicides, the temperature of the outer layers of fuel elements is about 130°-140°C even at multiplication coefficients 8 - 10. In Figure 3 the dotted line represents the temperature of the most thermally stressed fuel rod in the core based on ^{233}U-2.9% Si. For ^{235}U-2.9% Si the temperature is obviously lower beause of the larger core volume. Therefore, the operating temperatures and corresponding rates of corrosion for U-9% Mo and U-2.9% Si differ substantially. They are indicated by dashes in Table 1.

During reactor experiments with nonhermetical silicide fuel elements it was established that under temperatures less than 200°C the control system did not detected uranium in coolant.

Thus under real operating conditions in multiplying targets we cannot be sure of γ-phase stability and corrosion safety of the target. Moreover, we have not enough experience with U-Mo and U-Nb-Zr fuel elements operation in water coolant, but some experiments indicate that these alloys under irradiation become rigid and deform the fuel element can. In case of non-uniform operating conditions, such as we have for the fuel core in the target, we must expect strong deformation and bending of the fuel elements.

Swelling of uranium alloys under irradiation at temperatures less than 350-400°C is caused by accumulation of fission products in the crystaline lattice with the rate of 2-3 vol.% per 1 at.% of burnup. For fuel elements made of U_3Si the diameter increase of fuel element will be 0.1-0.3 mm at a burnup of about 2 at.% (Fig.4). For comparison, on the same figure, the swelling of U-9 wt.% Mo alloy is presented. For both cases the swelling is shown not accounting for the restriction factor of the can. Radiation amorphization of uranium silicide at temperatures below 210°C and burnup more than 10^{-4}at.% is a useful phenomenon /20/, since it provides:
- increased corrosion resistance of fuel;
- full thermal contact between core and can after several hours of operation; according to experimental investigation, this contact was not broken under non-stable temperature conditions;
- decreased negative influence on fuel element lifetime due to non-uniform core swelling and deformation under thermal shocks.

As we have established, uranium silicide in the amorphous state exhibits radiation creep comparable to that of the crystaline state under temperatures a little lower than the melting point. According to our calculations there will be an outer layer of fuel in the amorphous state under the full range of operating conditions of the target. This outer amorphous layer, as we expect, will prevent the core from cracking, unlike the case of U-Mo and U-Zr-Nb alloys.

In the most powerful operation conditions, that is the case of a multiplying target made of 98% $^{233}U_3Si$, the local maximum heating per pulse will be ~15°C. In order to decrease the additional influence of thermal shocks the fuel core is split into three parts (Fig.2) /11/. The cores made of substoichiometric silicide U - 2.9 wt.% Si are canned in stainless steel without a barrier sublayer, since at temperature less than 500°C and exposure for one year, the interaction layer does not exceed 10 microns.

Current technology enables us to manufacture not only fuel elements from natural uranium, but also from highly enriched uranium. Prototypes of fuel elements were tested under reactor conditions with water cooling for 3.5 years to a burnup of 3 at.%. The maximum local power density of the silicide core ranged up to 660 W/cm^3. This power density for a multiplying target at a design beam current of 0.5 mA will be achieved at a frequency of 25-30 Hz. Increasing the repetition rate up to 50 Hz will raise the power by two times and fuel core temperatures will be close to their operating limit. Thus the parameters of target operation demand careful investigation.

Experience in developing, testing and using the uranium silicide as a high-density fuel for light-water reactors is an additional reason for choosing it.

Besides the silicide, other high density amorphyzing compounds of U_6M - type, where M is Ni, Co, Fe, Mn and their combinations were investigated. But their corrosion resistance in water under temperatures about 200°C was insufficient. Taking this into account as well as the level of investigation of uranium silicide, that material was finally chosen as a fuel for target elements.

3. Core, moderators, reflector

The arrangement of fuel elements in the target lattice and their separation is provided by stainless steel wire 0.4 mm in diameter. Thermal - hydraulic calculations of the compact fuel element lattice was carried out on the basis of experimental data summarized in /13,14/.

In order to compensate for thermal expansion and the swelling of fuel elements in the transverse direction and to prevent vibration due to coolant flow, springy elements embracing the core from three sides are used. The most suitable materials for springy elements is Zr-Mo alloy since it retains elasticity under reactor conditions and has good compatibility with other target materials.

The inlet and outlet channels of the core coolant system must provide uniform coolant flow with the minimum volume of water in order to avoid lengthening of fission neutron lifetime. Therefore collectors are made in the form of vertical narrow channels made of material well-absorbing thermal and epithermal neutrons (item 4, Fig.2). The most suitable materials from the point of view of their nuclear properties, manufacturing technology and compatibility with other materials of core are Hf, Zr- Hf (up to 10 wt.% of Hf) and Ti-Gd (up to 36 wt.% of Gd) alloys. Profiled plates of these alloys are placed on the top of the core partially covering the

collectors.

The presence of different metals and alloys in the core required extensive studies of their compatibility and water coolant chemistry. Corrosion rates of metals under different conditions including neutron flux are shown in Fig.5. For additional corrosion protection the inner side of the aluminium vessel is coated with Ti. Investigations of Al and W samples coated with titanium under reactor conditions have shown that Ti-coating in water at about 80°C was compatible with other materials of the core. No modifications of Ti coating were observed after a period of time comparable with that expected for target operation /10/.

Before entering the core and after leaving it the coolant contacts the Be-reflector surrounding the upper moderator.

The outer vessel surface in the vicinity of the core (operating in vacuum or in the air), the moderators (except the radiating areas), and the reflector are coated with Gd, which acts, in particular, as a decoupler shortening the neutron pulse. In addition, this layer together with other construction materials of flow channels and the core ensures nuclear safety in case of accidental flooding of the gas vessel.

The upper moderator has a radiating area of 12×12 cm^2 and is 5.5 cm thick. A replaceable H-type insert from Ti-Gd alloy splits the moderator into two parts, e.g. 2.0 and 3.5 cm thick, which enables the optimization of moderator properties for certain experiments. The side surface of this insert serves as a decoupler separating the moderator and reflector. For replacement of the insert there is a removable Be-plug in the upper part of the reflector. Of course, the replacement of the insert is only possible on the stopped facility during maintenance periods, since it requires removing the shield and coolant piping. The lower moderator has dimensions of 20×20×3 cm and has been optimized for producing epithermal and intermediate energy neutrons.

4. Neutronics: calculations and optimization

Using rod-shaped fuel elements instead of plates decreases the number of spallation neutrons per incident proton, as the fuel volume fraction is less. The computed conversion coefficient K_n (number of fission and evaporated neutrons per incident proton) and multiplication factor K_m as a function of the core height for U-2.9% Si and U-9% Mo alloys are shown in Fig.6. The K_n coefficients for both a tungsten target and a monolithic natural uranium block of equal size are also shown for comparison.

Using natural uranium rod fuel elements made of U_3Si or U-Mo alloys, the number of generated neutrons increases in 1.4 - 1.5 times in comparison with a tungsten target. The same theoretical value for a monolithic natural uranium block with the density of 19.1 g/cm^3 is 2.0-2.1. The known maximum value of 1.7 was achieved in KENS-1 target consisting of four monolithic low-alloy uranium blocks with a small amount of construction materials /21/.

In all other cases, going to a more powerful proton beam and to arrangements of target fuel elements in the form of plates, this value does not exceed 1.5 /22-24/. Therefore, due to its low efficiency, the natural uranium target is considered only as an intermediate step towards the target with multiplication factor of 2-10. The upper value of the multiplication factor is limited by the delayed neutron background between two pulses, by nuclear safety requirements for subcritical assemblies without control and safety system /25/ and, finally, by the capacity of the cooling system (2.5 MW). Its lower limit is determined by the desire to construct a facility with neutron yield per incident proton more than that for a uranium block target.

It is very important for a pulsed neutron source to achieve the maximum density of fast neutron leakage to moderators. In that case the neutron flux density on the moderator surface also reaches the maximum /26, 27/. The total neutron output from the target and minimum cross section for thermal neutron absorption are important for stationary and quasi-stationary thermal neutron source with heavy water moderators /4,28/. The neutron output in the transverse direction as a function of core height is also shown in Fig.6. One can see that the neutron flux density reaches its maximum at the target height of 5-6 cm. Moreover, this approach enables us to perform optimization calculations in a more rational way, since we can separate optimizations of the target and moderators, using for that purpose experimental and computed data /29-31/ and codes /32-34/.

For a multiplying target based on ^{233}U, a multiplication factor of 6-8 is also achieved for a height of the target core of 6-7 cm. From that point of view, the target based on ^{233}U is more efficient than one of ^{235}U, where this multiplication factor can be achieved at a core height of 11-12 cm /Fig.6/. In the latter case the target is less efficient as the side flux of neutrons leaking away does not contribute to the total thermal neutron flux in moderators.

However, when a multiplying target is used, increasing its height leads not only to increasing K_m and K_n, it also degrades the form factor (due to higher leakage through side surface of the target and longer distance between the target center and moderators) and broadens the fast neutron pulse (the width of the slow neutron pulse is usually determined by the moderator). The competition of these factors determines the optimal multiplication factor /27,35/. Its dependence on the neutron energy E used for TOF experiments and on the average lifetime of prompt fission neutrons in the target core is shown in Fig.7. For calculations it was assumed that the proton beam from a proton storage ring has a pulse width 0.32 μs /1/, pulse duration of moderated neutrons was estimated by equation $\tau_m = 2/\sqrt{E}$ (μs). The function FOM = j/θ^2 was used as an objective, where j - density of fast neutrons leakage on the face of the target, and θ - pulse duration of neutrons with the energy E_n in the moderator. It is evident that the optimal value of multiplication is higher for experiments with neutrons of lower energy, and for targets with shorter prompt neutron lifetime.

For a ^{238}U target we have one more option: depending on the relative priority of experiments, we can vary the vertical position of the proton beam and, as a consequence, vary the neutron

fluxes in upper and lower moderators.

Calculations at high energies (20-600 MeV) were made with the SHIELD/CG code, which has been developed from the SHIELD code /36/. Some results were duplicated with the MARS-12 code /37/. Both codes were tested by experimental data /38-40/. Results are shown in Fig.8. One can see that the accuracy of calculation of the spallation process in the energy range up to 600 MeV is better than 10%. In the range of 0.0023 eV to 10.5 MeV the modified version of the MCU code /32,33/, MCNP /34/, and a specially developed Monte Carlo code were used. Test calculations of the fast spectrum critical assemblies /29,30/ and of a moderated pulsed fast reactor /31/ have shown the accuracy of 5% and 10% in criticality and prompt neutron lifetime respectively.

The calculated characteristics of pulsed sources based on W, ^{238}U, 90% ^{235}U and 98.1% U^{233} with a pulse frequency of the beam from the proton storage ring of 25 Hz and for K_m ~6 are given in Table 2.

Table 2. Characteristics of targets for 125 μA, 600 MeV proton beam.

Fuel	W	U^{238}	U^{235}	U^{233}
Average power, kW	78	110	720	600
Background power, kW	0	1	4.6	1.9
Thermal neutron pulse width, μs[1]	35/24	35/24	35/24	35/24
Multiplication	1	1.14	5.7	5.7
Average thermal neutron flux density, n/cm^2s:				
3.5 cm	$0.9 \cdot 10^{12}$	$1.3 \cdot 10^{12}$	$4.1 \cdot 10^{12}$	$5.0 \cdot 10^{12}$
2.0 cm	$0.5 \cdot 10^{12}$	$0.8 \cdot 10^{12}$	$2.4 \cdot 10^{12}$	$2.9 \cdot 10^{12}$
Peak thermal neutron flux density, n/cm^2s:				
3.5 cm	$1.0 \cdot 10^{15}$	$1.5 \cdot 10^{15}$	$4.7 \cdot 10^{15}$	$5.7 \cdot 10^{16}$
2.0 cm	$0.8 \cdot 10^{14}$	$1.3 \cdot 10^{15}$	$4.0 \cdot 10^{15}$	$4.8 \cdot 10^{16}$
Repetition rate, pps	25	25	25	25

[1] **For 3.5 and 2.0 cm sections of water moderator.**

5. Some alternative variants of target cooling

In the course of target development the possibility of using liquid metal coolants is considered. Alkaline metals were not considered because of their flammability.

It was assumed that the Pb-Bi and Ga-coolants enable us /15,41/:
-to increase the neutron output and the linear density of the primary neutron source;
-to extend the range of high-density uranium fuels (UMo, UN, UC), that are compatible with

coolant;
-in powerful targets to avoid coolant boiling and improve the hydraulic stability of parallel channel cooling.
-to avoid the effect of irradiation decomposition of water coolant.

Calculations showed that despite the increase in the fraction of heavy metals in the cases of Pb-Bi or Ga-coolants there is no significant gain in the neutron production density along the beam direction and the total number of primary neutrons per proton. This is the result of additional ionization losses of the proton energy and the decreasing number of non-elastic reactions on uranium per proton. Since uranium nuclei have much higher neutron output as compared to Pb, the decrease of flux and energy of hadrons will result in the degradation of the neutron output, which will not be compensated by heavy coolant.

The total neutron output from the target of 70 mm height with a composition corresponding to the hexagonal lattice of fuel elements 5-13 mm in diameter, with can thickness of 0.3 mm and with 0.4 mm gap, and for natural uranium U-9% Mo alloy is shown in Fig.9. For comparison, this figure shows the neutron output from the target with nitride fuel (UN, ρ = 13.5 g/cm^3) and Gallium cooling. UN is a fuel of increased density and is the most suitable choice in view of its compatibility with liquid-metal coolant (Pb, Pb-Bi, Ga). Calculations were made with the SHIELD code. One can see, that in the interesting range of element concentration in the core, the increasing is about 0.5-0.7 n/p. This difference was even smaller when estimations were made by the method of nucleon flux decomposition into non-elastic interaction integrals /42/ using the data of /43/.

But even this small addition for the multiplying target with metal coolant is eliminated because of the harder neutron spectrum, requiring larger target height for the same multiplication factor and, correspondingly, a larger fraction of neutrons leaking away. The factors mentioned above reduce the density of neutron leakage towards the moderators. Besides the absense of significant gain in neutron output, using liquid metals as coolant raises many technology problems and, as a consequence, leads to a more complicated and expensive design of the target and supporting systems, particularly for Pb-Bi alloys.

Contrary to our expectations, using liquid-metal coolants does not extend the choice of high-density fissile materials compatible with these coolants, since experience with these materials has been obtained, mainly, in sodium-cooled reactors, while high-density fuels have not ever been tested with heavy coolants even in trial operation /44,45/.

As we mentioned above, the problem of ensuring single-phase coolant flow and hydraulic stability becomes significant for powerful multiplying targets. The physical characteristics of such a target which consists of a dense fuel element lattice with highly enriched uranium and water coolant allow us to maintain stable single-phase coolant flow due to high sensitivity to hydrogen content. The experience gained in developing the booster based on the plutonium oxide /46/ and other calculations indicate an abrupt decrease of multiplication factor and target power if any decrease of water concentration occurs due to boiling in either one channel or the whole target (ΔK_{eff} = - (18-20)% for the total coolant loss). This is one of the factors that

ensure nuclear safety, some aspects of which were discussed in /47/. Considering nuclear safety in all its aspects is not an intention of this article.

However, these questions include:
- conditions of fuel element loading;
- the biggest possible fuel cassette size;
- using liquid control additions ($Gd(NO_3)_3$, NaB_5O_8, H_3BO_3) to prevent the local change of spectrum and reactivity rise in case of partial replacement of some fuel elements with water or accidental increase of the coolant fraction.

Table 3. Comparison of characteristic for multiplying water and liquid metal targets

Fuel	**90% ^{235}UN**	**90% $^{235}U_3Si$**
Coolant	**Ga**	**H_2O**
Reflector	**Ni*)**	**Be**)**
Multiplication	**7.1**	**5.7**
Yield, n/p	**11**	**12.5**
Core height, cm	**12.2**	**9.0**
Average power, kW	**850**	**720**
Repetition rate, pps	**25**	**25**
Average thermal neutron flux density, n/cm^{2}s (3.5 cm - moderator)	**$5.8 \cdot 10^{12}$**	**$4.1 \cdot 10^{12}$**
Peak thermal neutron flux density, n/cm^{2}s	**$5.6 \cdot 10^{15}$**	**$4.7 \cdot 10^{15}$**

*** 40 cm Ni - reflector arround all target**
**** Be - reflector around upper moderator**

Stability of hydravlic and increase of target safety may be obtained in case of liquid metal coolant. Computed target parameters with nitride fuel-UN and Ga-coolant are shown in Table 3. Dimensions of target cross sections, fuel elements lattice parameters and ampule diameter are the same, as for silicide fuel with water coolant. In contrast to silicide target, where Be-reflector is sited only in the topof target, in liquid metal target the Ni-reflector (40 cm in thickness) surrounds all target. This is necessary to increase reflector capability, compensate lower density of uranium nitride and to reduce target dimensions.

6. Conclusion

A multiplying target with a gain factor of 10 cannot serve as a proton beam trap unlike the cases of tungsten and natural uranium targets because, at full proton beam power, the average

target heat power will be ~7 MW, that is three times greater than it was assumed.

Fuel elements on the basis of U_3Si with natural uranium, tungsten plates, vessel, shields, berillium moderators and some other elements of the core have already been manufactured.

Choice of core materials, investigation of compatibility and corrosion behaviour have been fulfilled.

The protective covering and assembly of the core are now in process.

For non-traditional usage of neutron source the possibility of rare pulsed booster creation for nuclear laser charging and irradiation of extended objects was investigated. It was shown that in central channel along the distance of more than 1.0 m it is possible to create almost uniform fast neutrons pulsed flux with dencity more than 10^{21} n/cm^{2}s having spectrum similar to that of fission and on outer surfaces more than 10^{20} n/cm^{2}s /48/.

7.Aknoledgements

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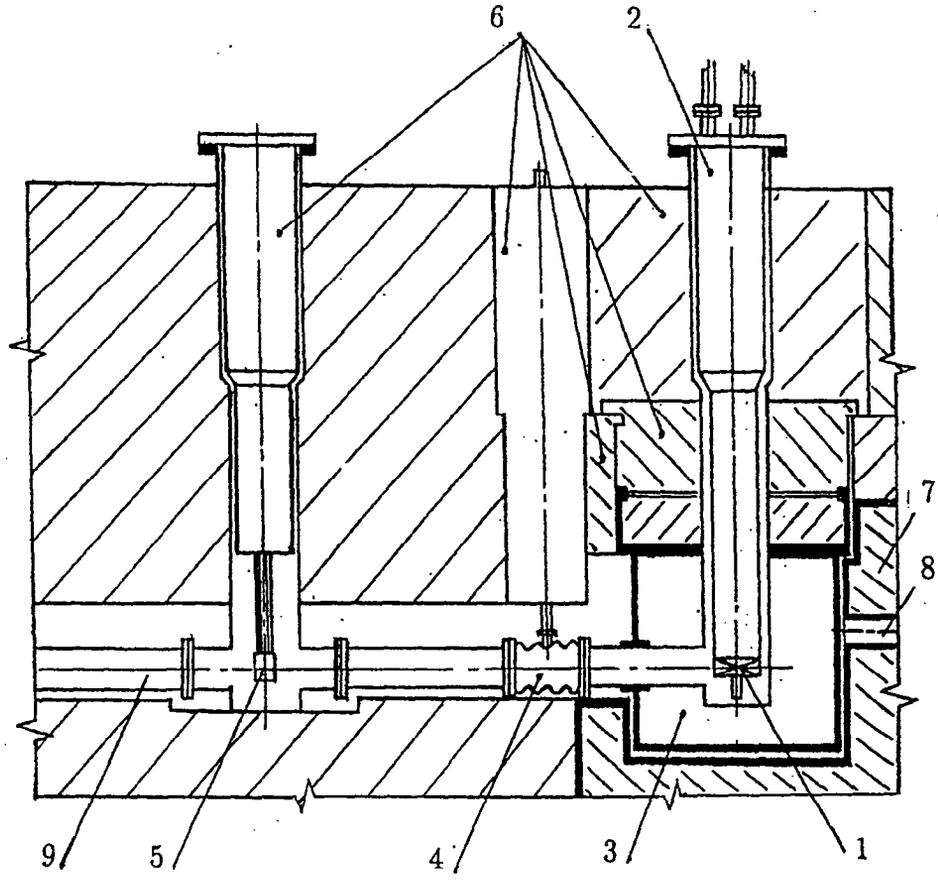

Fig.1. Pulsed neutron source layout: 1 - core; 2 - target.vessel with steel shielding plugs; 3 - gas tank; 4 - remote controlled vacuum seal; 5 - proton beam monitoring system; 6 - shielding plugs; 7 - thermal shielding; 8 - neutron guides in biological shielding; 9 - proton guide.

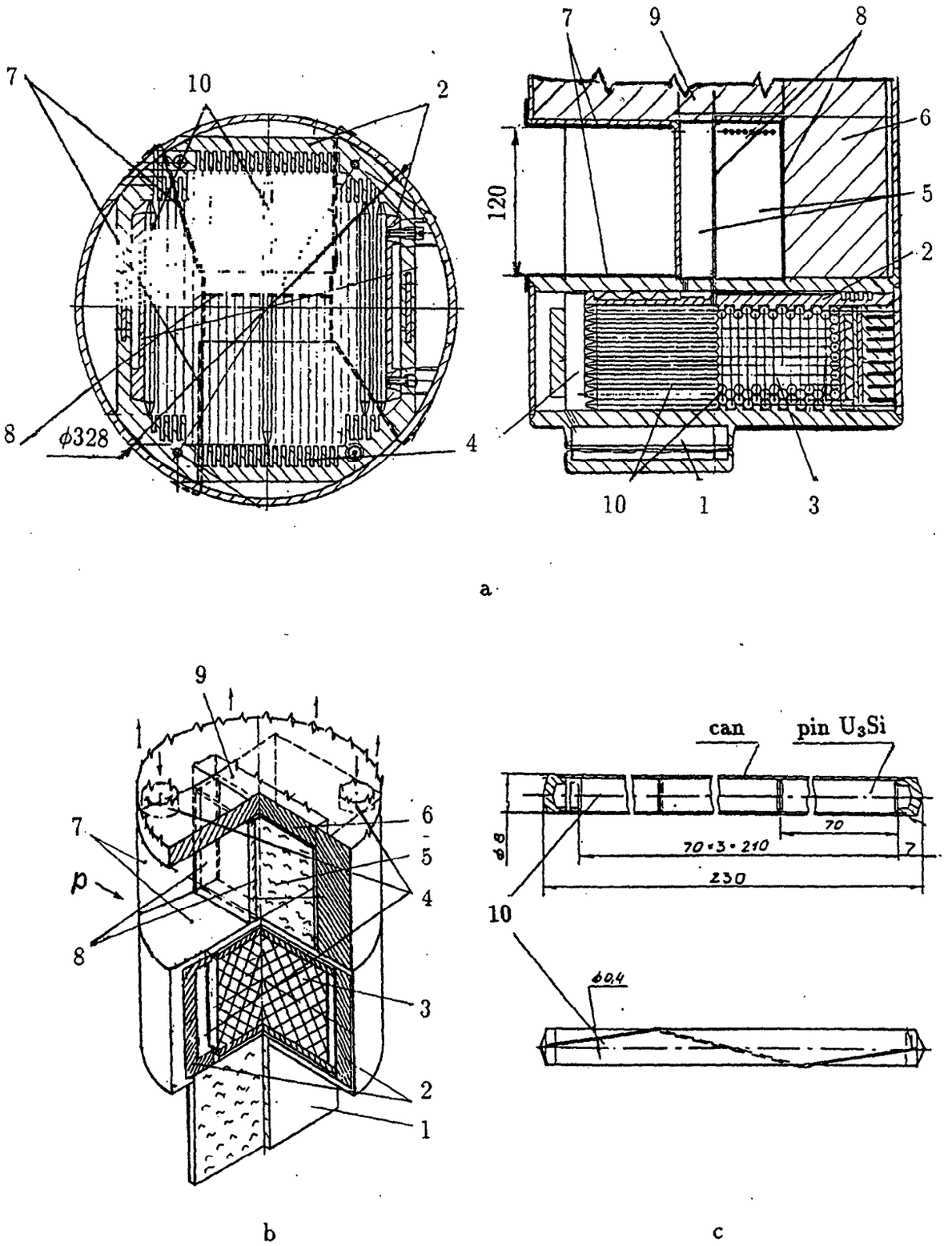

Fig.2. Central part of the pulsed neutron source: 1 - lower moderator; 2 - Ti-Gd decoupler; 3 - core; 4 - coolant inlet and outlet; 5 - upper moderator; 6 - Be-reflector; 7 - Gd - decoupler; 8 - Ti-Gd - insertion; 9 - Be-plug; 10 - fuel element.

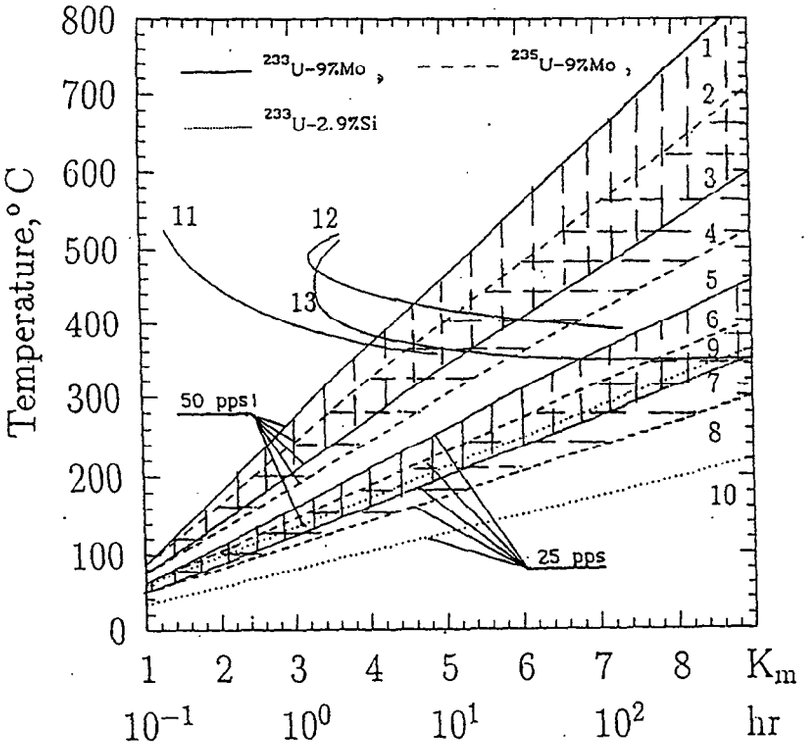

Fig.3. Expected temperatures in the center of the most thermal-stressed fuel element with the core from U-9 wt.% Mo and U - 2.9 wt.% Si alloys vs. the multiplication coefficient - K_m, pulse frequency and gap between core and can (for 1,2,5,6 δ = 0.05 mm; 3,4,7,8 δ = 0.025 mm; 9,10 δ = 0.), the curves represent the time-temperature gamma-phase stability boundaries for the same metastable alloys: 11 - U-Nb-Zr, 12 - U-9% Mo, 13 - U-6% Mo.

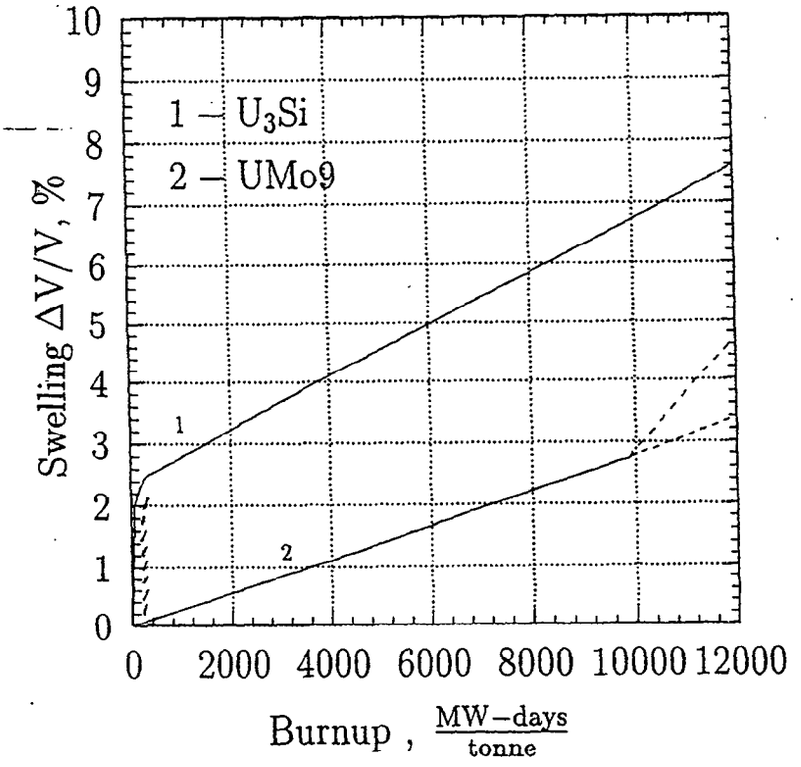

Fig.4. Swelling of core materials. The area of core cracking is shown by the dashed line (1 at % ~ 10000 MW·day/tonne); the area of amorphization is hatched.

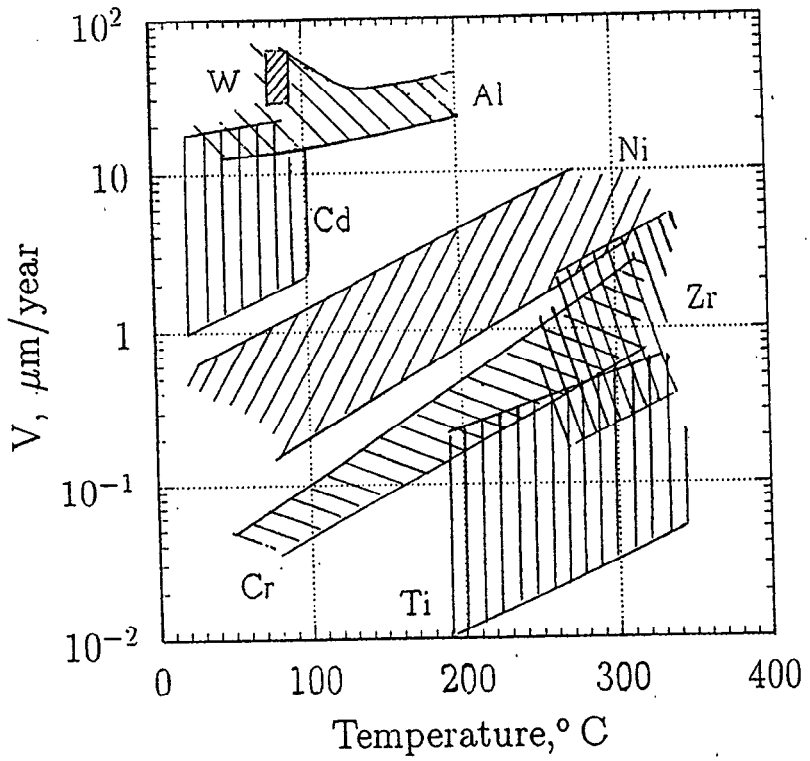

Fig.5. Corrosion rates of materials in distilled water at various temperatures.

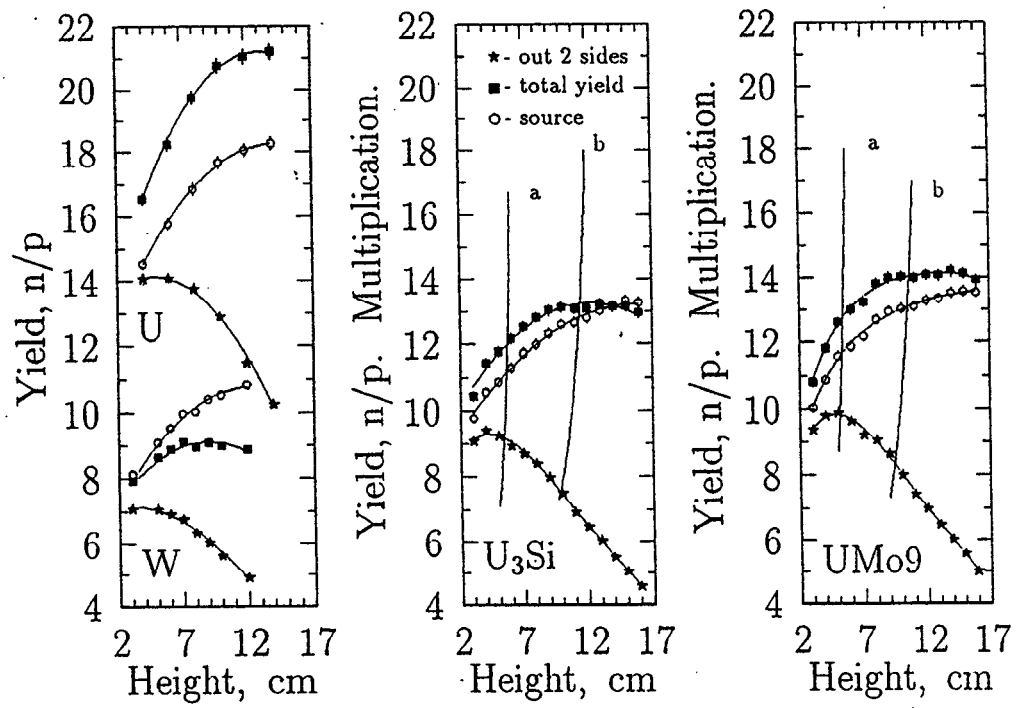

Fig.6. Computed spallation neutron yield per 600 MeV incident proton for W-target and U-2.9% Si and U-9% Mo natural uranium alloys target, and the multiplication factor (a - U^{233}, b - 90% U^{235}) for targets of various height (scale is the same). Neutron leakage from upper and lower surfaces is shown separately.

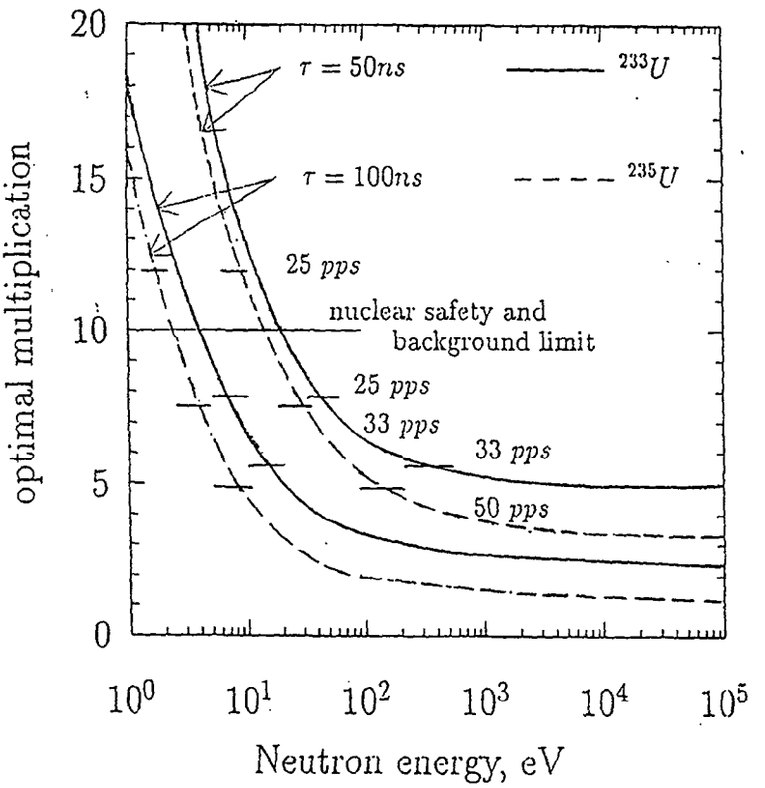

Fig.7. Dependence of optimal multiplication factor on the energy of the neutrons used in TOF experiments with the storage ring for various values of the prompt neutron lifetime. The optimal multiplication factor for the repetition rates of 25 pps, 33 pps, 50 pps are shown by horizon lines.

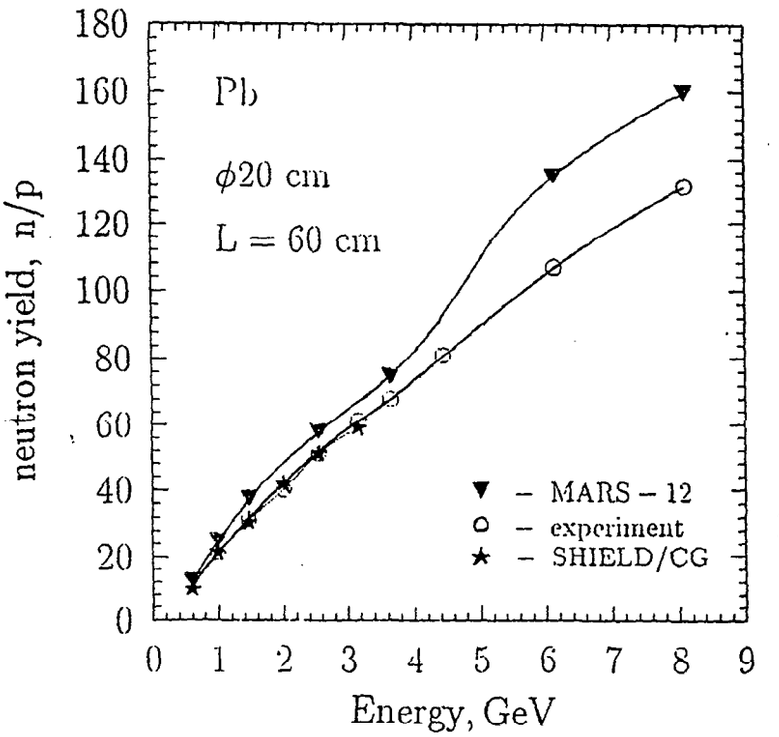

Fig.8. Comparison of experimental and computed data by SHIELD and MARS-12 codes of neutron output from lead cylinder surface.

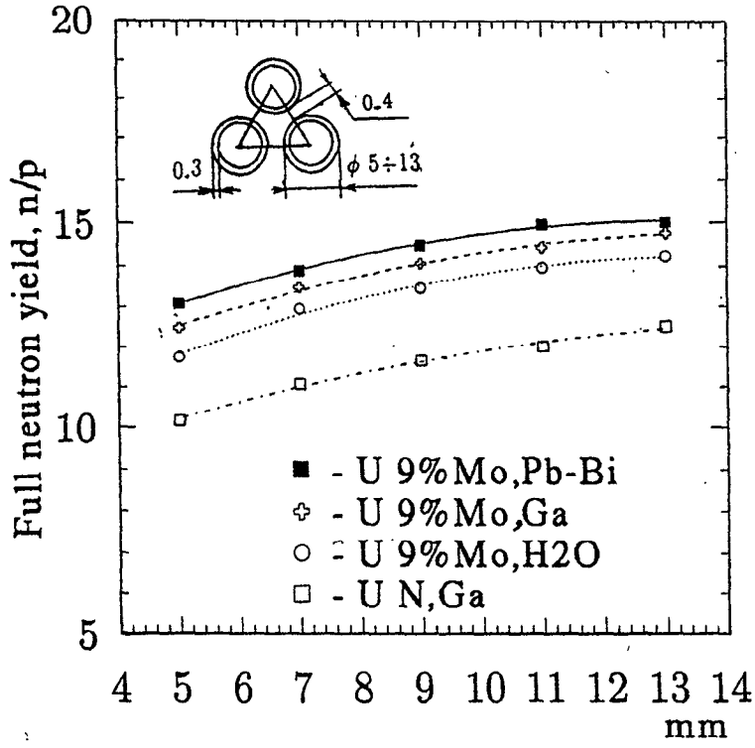

Fig.9. Influence of coolant type and rod fuel element diameter upon spallation neutron yield per 600 MeV incident proton for a target with dimensions of 7.0×19.88×21.1 cm. Core composition corresponds to a hexagonal lattice of fuel elements with U-9% Mo cores from natural uranium with 0.3 mm thick cans and coolant gap - 0.4 mm. For comparison the neutron yield for a target with the UN - core and Ga - coolant is shown separately.