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JET-EFDA, Culham Science Centre, OX14 3DB, Abingdon, UK

[1]EURATOM–IPPLM Association, 23, Hery St, 01-497, Warsaw, Poland
[2]EURATOM–ENEA Fusion Association, Consorzio RFX, I-35127, Padova, Italy
[3]EURATOM-CCFE Fusion Association, Culham Science Centre, OX14 3DB, Abingdon, OXON, UK
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ABSTRACT.
The neutron diagnostics of large tokamaks like JET are essential to estimate the fusion power and to determine some various parameters. The neutron activation method supported by neutron transport calculations is very useful especially for the evaluation of the total neutron yield from a single plasma discharge.

The paper presents some results of the activation experiments and calculations carried out for the JET plasmas, from selection of the activation materials to their irradiations in the neutron field of JET discharges. Neutron transport calculations leading to activation coefficients for new materials have been performed. The results of the calculations have been used to design new composite foils to obtain information also about the neutron spectrum and not only the total yield. Measurements using these new activation materials have been performed during last JET experimental campaigns. The obtained experimental results are compared with the theoretical calculations.

1. INTRODUCTION

Large tokamak fusion experiments, involving deuterium or deuterium tritium plasmas, can generate more than 10^{12} neutrons per second. At these emission levels, it becomes possible to obtain information about the plasma with various techniques measuring fluxes and energy distributions of neutrons. Neutron emission rates and energy spectra of neutrons, for plasmas with Maxwellian energy distribution, can provide information about ion temperature, fuel density; moreover the generated fusion power can be evaluated. In the case of discharges with significant additional heating, the ion distribution functions and ratios of thermal and non-thermal reaction rates can be evaluated.

One of the measuring techniques used on large tokamaks, to determine neutron fluence and/or energy spectrum at specific locations around the machines, is the neutron activation technique. This technique is very suited to accurately and precisely monitor the fusion power by measuring the neutron yield. It†gives stable and linear response with the fusion power level [6].

The method is based on recovering the information about neutrons by registering the products of induced reactions. Samples of selected materials are used as detectors. During the irradiation of the samples reactions creating radioactive nuclei are induced. The chosen materials have relatively high cross-sections for reactions with neutrons in specified energy ranges. The†reactions are selected in such a way that their products decay with emission of gamma-rays, therefore they can be detected by means of gamma-spectrometry. Exceptionally, also reactions leading to beta-radiation have been used. Additionally, the materials of the samples can be chosen such that neutrons with energies of 14.1 and 2.45MeV and scattered neutrons are measured separately.

The uncertainty in the measurements performed with activation techniques depends mainly on the accuracy of the cross-section data and on the geometry factor. It is generally accepted that the total yield of the 2.45MeV neutrons can be determined with an accuracy better than 10% in JET.

The measured neutron fluence at the irradiation point is related to the total neutron production by applying a calibration factor that can be determined using neutron transport calculations (e.g.

MCNP code), taking into account the geometry of the detector, the geometry and composition of the samples. Benchmark experiments, using suitable calibration neutron sources, ought to be performed to check the reliability of the calculations.

The advantages offered by the activation techniques are the large dynamic range of the measurements; moreover-a virtue important for the application in fusion, the method is free from mechanical and electro-magnetic interference problems. Unfortunately, the sensitivity of the technique is relatively low and the lowest neutron fluence at the irradiation point should be of the order of $10^6 - 10^7$ cm^{-2}.

This paper describes the activation of selected materials in the neutron field generated by deuterium plasmas at the Joint European Torus (JET). The irradiations have been carried out by means of JET activation system [1] additionally supplied with a modern High Purity Germanium (HPGe) gamma-spectrometer. It†has allowed to measure the activity of each irradiated sample and therefore to calculate each reaction rate.

In this paper we compare the obtained reaction rates with neutron yields determined by another JET diagnostics. They appear to be in quite good agreement.

2. EXPERIMENTAL SETUP

The JET activation system (KN2) consists of eight irradiation ends (IE) arranged around the tokamak in upper and lower positions. Seven of these ends are located behind the tokamak shield and only one of them (located in Octant 3 in upper position – so called 3U irradiation end) is placed inside the vacuum vessel [1]. Because of the high temperature close to the tokamak first wall (c 300^{o}C), the 3U IE is being cooled-down by water.

The activation samples are transported in a polyethylene capsule to the irradiation ends and then back to the detectors through a so called carrousel by means of a pneumatic system†[1]. In given irradiation position the samples are exposed to the fusion neutrons. For the measurements described in this paper only the 3U IE has been used to irradiate the multi-element set. Simultaneously, in the 6U†IE a 10†g silicon sample being the indicator of 14MeV neutron yield has been systematically irradiated. In order to do that one specifically prepared capsule containing silicone has been transported in closed circuit between 6U IE and a NaI scintillation-probe. The short half-life time of ^{28}Al (2.2min.), being produced by fast neutrons in silicon, makes it possible to use the same sample for subsequent JET pulses. Also always two thorium samples have been irradiated in the 2U and 2L IEs respectively. The delayed neutrons emitted from the irradiated thorium samples have been counted by means of two stations of ^{3}He counters. It allows to estimate the total neutron yield and to compare this value with the results of the fission chambers (KN1) measurements being used at JET as permanent neutron yield monitor.

After irradiation and delivering by pneumatic system the γ-rays emitted by the samples have been measured by means of a HPGe gamma-spectrometer. It allows to identify the occurring reactions and to precisely estimate the activity of their products and therefore to calculate the rate of each

reaction with the relations:

$$\alpha = \varphi(E) \cdot \sigma(E) \cdot dE = \frac{A_0(t, \gamma)}{N_a \cdot (1-e)^{-\lambda \cdot t_r)}} \quad (1)$$

$$\textit{where} \qquad A_0 = N_r \cdot \lambda\, N_r = \int_0^{\infty} \int_0^{t_r} \varphi(E) \cdot \sigma(\mathrm{E}) \cdot N_a \cdot e^{-\lambda \cdot t} \cdot dt \cdot dE = \alpha \cdot N_a \cdot \frac{1}{\lambda} \cdot (1-e)^{-\lambda \cdot t_r});$$

The symbols in the previous relations indicate: A_0 – the activity of the sample at the end of irradiation, N_r – the number of radioactive nuclei, N_a – the number of given target nuclei in the sample, t_r – the irradiation time, λ – the decay constant.

Deriving the neutron yield from the measured reaction rates is possible after carrying out the required neutron transport calculations for the specific measurement conditions. Such calculations are developed by our teem by means of the MCNP code [2] based on the model of JET originated from M. Loughlin [4].

3. SELECTION OF THE ACTIVATION MATERIALS

In addition to indium, which is particularly well suited as an activation detector for plasma devices, also other materials are needed. For a possible upgrade of the activation method for the neutron measurements at JET suitable activation materials are under investigation [7, 8]. The reactions needed to perform useful measurements on a fusion device have to meet several requirements. First of all the isotopes naturally occurring in the activation materials must have a reaction cross-section high enough in the relevant neutron energy range. At the same time, the products of these reactions should decay with appropriate half-lives and emit suitable gamma-photons.

The neutrons at JET during deuterium operation come from d-d fusion and in small percentage from tritium burn-up. Originally they are 2.45 and 14.1MeV in energy respectively, which is broadened due to the plasma temperature and the applied heating systems. The typical plasma pulse at JET takes usually about 30 s, whereas the main neutron emission occurs only during the additional heating phase, mainly with Neutral Beam Injection (NBI), which lasts for a few seconds only. Therefore the chosen reactions have decay products with half-lives of more than a few seconds. Because of properties of the applied gamma-spectrometer the reaction products must emit gamma-lines of energy between100keV and 2000keV. It was assumed that the most useful cross-sections are of the order of hundreds of millibarns.

An additional very important issue is the accuracy of the available cross-sections. We strive to use only verified and current nuclear data libraries, like ENDF/B-VII.0, JEFF 3.0/A, IRDF-2002. Some of the reactions considered for measuring 2.45 and 14.1MeV neutrons are presented in Table 1 and 2 respectively. Threshold-less reactions like radiative capture are used for slowed-down (scattered) neutrons measurements. We also registered this type of the reactions.

The materials selected in our laboratories have been subjected to number of testing measurements [7, 8].

4. COMPOSITIONS OF COMPOUND SAMPLES

In order to simplify and accelerate the operation with the activation samples to measure their activities it has been proposed to apply multi-element samples. Using one sample consisting of a number of elements simplifies gamma-acquisition and gamma-efficiency calculations. This approach requires to measure a number of short-lived products of the neutron induced reactions. They should be measured simultaneously because of their quick decay.

The compositions of the samples have been selected to emit well-separated gamma-lines. The applied HPGe spectrometer allows to discriminate lines even less than 5keV apart. The materials have been selected to have comparable half-life times values of the reaction products.

The proportions of the various components in the samples have been matched in such a way that the expected number of counts due to the chosen reaction products should be similar. The geometric efficiency coefficients for the gamma-rays have been estimated for the used Canberra® HPGe spectrometer by means of the LABSOCS/ISOCS® software.

A number of different sample compositions have been tested. They are shown in Table 3.

Exemplary gamma lines which were obtained for the sample Mix9 (see Table 3) irradiated during 73746 JET pulse are presented in Fig.1.

One can see (Fig.1) that the chosen combination of the materials in this sample provides a similar number of counts in the specified measuring conditions, e.g. 100s of cooling time and 180s of acquisition in case of short-lived nuclides.

5. TECHNOLOGY FOR THE MANUFACTURING OF THE MULTIELEMENT SAMPLES

For the production of multi-element activation samples we have considered alloy and powder metallurgy methods. We have chosen the second manufacturing procedure for the simplified production because some of the selected elements need specific melting conditions. For example vapour pressure of cadmium (14,8Pa at 597K) is higher than vapour pressure of indium ($1{,}42\times10^{-17}$Pa at 429K) what is unfavorable for alloy metallurgy.

High purity (99.9%) powdered elements have been used in the manufacturing process. To obtain homogeneous samples the powder particles should be of similar size, in our case between 40μm and 100μm. The first step of the production process consists of mixing the individual elements in the correct stechiometric proportions. To reduce friction between the powder particles, special components, such as graphite, stearin, alcohol, ether or acetone less than 1% of mass, have been added. Then powders have passed through a die to produce a cohesive structure of the required dimensions (Fig.2a). By means of a purpose-made die the powders have been pressed into a cylindrical pellet of 18mm in diameter (Fig.2(b),(c)). The mass of the pellets was typically of about 0.005kg.

6. EXPERIMENTAL RESULTS

The neutron activations have been carried out only in the 3U irradiation end by means of the JET

activation system (KN2).

The selected samples have been located into a polyethylene capsule of the KN2 system. Before the JET pulse the capsules with the activation samples have been transported to the irradiation ends by means of the pneumatic system [1]. After irradiation and delivering by the pneumatic system the samples were measured by means of a HPGe gamma-spectrometer.

A number of reactions which are available for the measurement of the 2.5MeV neutrons look potentially very optimistically. Particularly promising is yttrium which has not been used as an activation detector before.

Unfortunately, we had to abandon some of the selected reactions. For some radio-nuclides it resulted from difficulty in identifying their original reaction. This†happens only when the activated elements are multi-isotopes. In some cases the radio-isotopes can also be produced by inelastic scattering as well as by radiative capture or by (n,2n) reaction. It is impossible to distinguish the proportions between these reactions only by measuring the resulting activity of their products. Such elements can not be used directly to estimate neutron fluence but they can be used to verify the correctness of the MCNP calculations. Some radionuclides which could not be unambiguously assigned to one desired reaction are presented in TabLE 4.

In the case of hafnium, apart from above-mentioned difficulties, there appear to be also a problem in distinguishing the origin of one of the registered gamma-lines. This is due to the fact that the gamma-photons with energy around 215keV are emitted by six different isomers of four hafnium isotopes, i.e. ^{177m}Hf, ^{178m}Hf, ^{178n}Hf, ^{179m}Hf, ^{179n}Hf and ^{180m}Hf. Additionally, uncertain cross-sections of the reactions with hafnium have caused problems with its application. Therefore, hafnium has been rejected as an†activation material.

The products of the reactions listed in table 1 as N^{o} 5, 8, 11 and 14 have too short half-life times. In case of the reaction N^{o} 13 the short half-life also causes difficulties in the decay measurements. Furthermore, the gamma-lines coming from the desired reaction product (^{197m}Au) have been covered by, the so called, Compton-background originated from the products of decay following the radiative capture induced by scattered neutrons, i.e. ^{198}Au.

The reactions induced in indium (N^{o}10 Table 1), yttrium (N^{o}7 Table 1) and nickel (N^{o}3 Tab. 1) appear to be very valuable. They have been registered with uncertainties less than 10%.

Among the reactions singled out for 14MeV neutrons particularly useful turned out to be all the reactions of aluminium, iron, cobalt, titanium and niobium. The induced activities are always low, nevertheless they have been usually estimated with uncertainties not larger than 10%–15%.
During planned deuterium-tritium campaigns and even during trace-tritium experiments at JET the number of 14MeV neutrons increases by a few orders of magnitude. Then the uncertainties of the measured activities will be much lower.

The attempts to register the products of the reactions induced only by 14MeV neutrons in zirconium, molybdenum, gold and lead have not been successful. In case of lead the cause is deemed to be the very low natural abundance of the target isotope and therefore the very low induced

activity, which has prevented sufficient counting of the emitted gamma-photons. In case of gold it has proven to be difficult to measure the gamma-lines because they are covered by Compton-background from the product of radiative capture. In†case of zirconium and molybdenum the reason for not registering the required reactions is again a too low reaction cross-section. We†think that this should be completely different (quite the contrary) in the case of tritium experiments.

Knowing the activity of the irradiated samples, their properties and the irradiation conditions, it is feasible to calculate each reaction rate (see equation 1).

The calculations have been made assuming that the overwhelming majority of the neutron emission from JET pulse takes place only during the NBI heating period, which usually lasts not more than 10 seconds.

Some of the obtained reaction rates are presented in Fig.3 as a function of total neutron rate.

The presented results have been obtained from the activations performed during seventeen selected JET Pulse No's: (73157, 73179, 73344, 73529, 73567, 73743, 73746, 73811, 73927, 76307, 77043, 77152, 77270, 77397, 77625, 77947 and 78055), which were executed during the C20–C26 Experimental Campaigns. All of the pulses were high neutron yield stable deuterium plasmas with high heating energy coupled/deposited mainly by the Neutral Beam Injection system. In the selected pulses the contributions of other heating systems is negligible.

One can see that the reaction rate varies roughly linearly with the neutron emission (Fig.3). Some scatter of the results can be explained by uncertainties in determining the details of the measuring procedure (the cooling and counting times were measured by a hand watch, not automatically) and also by uncertainties in the detector calibration and counting statistics.

The reaction rate is related directly to the neutron rate by the so called activation coefficient (K) [4]:

$$Y = \frac{a}{K} \tag{2}$$

where: Y is the total neutron rate [s^{-1}].

The activation coefficient (K) represents the number of reactions per target nucleus per source neutron. Usually a neutron transport code MCNP [2] is used to calculate the activation coefficients for the different activation materials. They can be also determined from slopes of the straight lines showed in Fig. 3 which represent the reaction rates versus the total neutron rates.

The activation coefficients for the ^{115}In(n,n') and ^{232}Th(n,f) nuclear reactions induced in indium and thorium samples were calculated a few years ago as a result of the last complete simulation of the JET geometry for deuterium plasmas [4].

The measured reaction rates can therefore be used to obtain the activation coefficients for the materials which were not be considered during the previous simulations [4] and therefore they can confirm and verify the future MCNP calculations.

Apart from aforementioned threshold reactions we have registered a number of radiative capture

(threshold-less) reactions which also can be used to determine the neutron fluence and/or neutron energy spectra, especially in the low-energy region.

Radiative capture reactions have been applied on JET using the cadmium difference method†[5] – the activation technique for distinguishing scattered and thermalized neutrons. It is based on simultaneous irradiation in a given neutron field of two identical samples in which the radiative capture reaction occurs. One sample is inside a cadmium shield and the other one is without any shield. These measurements have revealed circa 20% difference in†the activity of the both gold samples when they were irradiated simultaneously in the same JET discharge (see Fig. 4).

It proves the presence of thermalized neutrons in the 3U IE but at a comparatively low level.

7. NEUTRON TRANSPORT CALCULATIONS

The Monte Carlo neutron transport code – MCNP5 [2] has been applied to calculate the activation coefficients for the new materials used during recent JET campaigns. It†allows to perform the simulations of the influence of the real surroundings (different materials in†defined geometries) on the neutron flux and spectrum.

The input file for MCNP represents the simplified geometry and materials of JET structure. It was elaborated by M. Loughlin et al. [4]. It contains also the details of the irradiation point (3U IE) surroundings. The considered model did not include the cooling water washing the 3U IE. To analyze the influence of the water on the neutron energy spectrum a 10 mm thick water jacket (see Fig.5) has been implemented in the code. Such an arrangement is an overestimate of the real water layer which was not known precisely during the writing of this paper.

The neutron energy spectrum calculated by means of the MCNP code, taking into account the above mentioned complement, is compared in Fig.6 with the spectrum calculated without this complement. One can see that the alterations of the input file result in distinct change of the calculated spectrum – more slowed-down neutrons are revealed. This refined model seems to provide a more reliable reconstruction of the real neutron spectrum in the 3U irradiation end, which has been confirmed by the results of the cadmium-difference measurements.

The MCNP5DATA cross section libraries [3] have been used for the neutron transport calculations, whereas for the calculations of the activation coefficients the data of some carefully selected libraries have been used, i.e.

- for the ^{115}In (n,n') reaction the MF = 10MT = 4 data from the IRDF-2002 cross-section library;
- for the ^{58}Ni (n,p) reaction the MF = 3 MT = 103 data from the IRDF-2002 cross-section library;
- for the ^{89}Y (n,n') reaction the MF = 3 MT = 51 data from the JENDL 3.3 cross-section library.

The adapted MCNP input file has then been used to calculate the activation coefficients for materials applied in our experiments. The obtained activation coefficients allow to link the measured activities to the total neutron yield and further they could be used for the energy spectrum deconvolution procedure.

8. COMPARISION OF THE ACTIVATION SYSTEM RESULTS AND FISSION CHAMBERS MEASUREMENTS.

In order to finally estimate the activation coefficients for the following nuclear reactions; ^{115}In(n,n'), ^{89}Y(n,n') and ^{58}Ni(n,p) the use has been made of the fission chambers system (KN1), the activation system (KN2) and the MCNP calculations with the input file [4] including appropriate cross-sections.

At first, we have applied the original input file [4] to calculate the activation coefficients. The coefficients obtained from the measurements are compared in Fig.7 (a-c) with the ones from the above mentioned MCNP calculations. The experimentally determined activation coefficients correspond to the slopes of straight lines presented in these figures.

The activation coefficient measured by us for the ^{115}In(n,n') reaction is $4.03 \cdot 10^{-31}$ (see†Fig. 7a), whereas according to [4] it is $4.0 \cdot \sum 10^{-31}$. The very good agreement between these two values is due to the fact that the fission chambers (KN1) were originally cross-calibrated by activation measurements of ^{115}In(n,n') reaction with the coefficient presented in [4].

A small discrepancy between the activation coefficients for the ^{115}In(n,n') reaction calculated by M. Loughlin et al. [4] and our calculation is to be noted. It can result from different cross-section libraries used in our MCNP calculations.

Using the activation coefficients obtained from the MCNP calculations, the total neutron yields have been calculated for the data collected by our activation measurements. Fig. 8 presents the comparison of the neutron rates calculated in such a way for three reactions with the measurement of the fission chambers (KN1).

A linear fit indicates an agreement to within 6 % for the nickel and indium, and 8 % for yttrium samples.

The activation coefficients have been recalculated once more, but this time applying the improved input file taking into account the details of the 3U IE as presented in Fig. 5 (which resulted in the blue spectrum presented in Fig. 6). The refined new coefficients allow to recalculate the total neutron yield. Comparison of the neutron rates from KN1 and the ones calculated from our activation measurements with the two sets of coefficients are presented in Fig.9.

One can see (Fig.9) that the comparison of the neutron rates as recorded by the KN1 fission chambers with those calculated from the activation measurements for coefficients coming from the new (dotted lines) and old input files. There is roughly a 26% discrepancy in these coefficients and, therefore, in the calculated neutron yield. This discrepancy arises from the assumed thickness of the water layer and will change when the real value will be implemented.

CONCLUSIONS

The obtained results can be summarized as follows:

1. The manufactured and prepared multi-element samples are very useful for the activation method because each of them consists of a few elements which makes the gamma-acquisition easier and faster and the gamma-efficiency calculations simpler.

2. Application of the cadmium difference method allows to confirm the presence of thermalized neutrons in 3U IE.
3. The comparison of the reaction rate measurements performed for short, medium and long time decay isotopes (^{89}Y, ^{115}In, ^{58}Ni) with the KN2 activation system on JET and the fission chambers (KN1) (Fig. 3) provides activation coefficient for 3U IE. The activation coefficient for ^{115}In(n,n') evaluated from Fig.3a agrees well with the coefficient calculated in [4] (see Fig.7a) because the fission chambers (KN1) were originally cross-calibrated by MCNP calculations for indium samples placed in 3U IE.
4. The MCNP calculations of the activation coefficients including current nuclear data libraries of cross-sections (IRDF-2002, JENDL 3.3) for the above mentioned reactions, using adapted input file of MCNP code [4] have been performed.
5. The comparison of the total neutron yield from KN1 and KN2 indicates an agreement to within 6% for indium and nickel and 8% for yttrium (see Fig.8). The bigger discrepancy for yttrium seems to be connected with the low accuracy of the specific cross-section.
6. The activation coefficients for the analyzed reactions have been also calculated including details showed in Fig.5. Comparison of total neutron rate from KN1 and KN2 for these activation coefficients is presented in Fig.9. The calculations showed that the activation coefficients change significantly when the cooling water around 3U IE is taken into account.

In light of the above summary points we can conclude that the activation method can be used to verify the neutron transport calculations and can be useful to cross calibrate other neutron diagnostics provided that all important details, especially those close to the irradiation-end, are included in the MCNP input file. Therefore, for the verification of the input file the calibration in situ by an in situ neutron source is absolutely necessary. Another useful test for the verification is a comparison of the calculated and measured neutron spectra at the irradiation end.

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–	Reaction	Threshold [MeV]	Product half-live
1	Ti-47 (n,p) Sc-47	1.8	3.3 d
2	Fe-54 (n,p) Mn-54	1.8	312 d
3	Ni-58 (n,p) Co-58	1.6	71 d
4	Se-77 (n,n') Se-77m	0.2	17 s
5	Br-79 (n,n') Br-79m	0.2	5 s
6	Sr-87 (n,n') Sr-87m	0.4	2.8 h
7	Y-89 (n,n') Y-89m	1.2	15.7 s
8	Zr-90 (n,n') Zr-90m	2.3	0.8 s
9	Cd-111 (n,n') Cd-111m	0.5	49 m
10	In-115 (n,n') In-115m	0.6	4.5 h
11	Er-167 (n,n') Er-167m	0.3	2.2 s
12	Hf-177 (n,n') Hf-177m	1.3	51 m
13	Au-197 (n,n') Au-197m	0.5	7.7 s
14	Pb-207 (n,n') Pb-207m	1.6	0.8 s

Table 1:

_	Reaction	Threshold [MeV]	Product half-live
1	Al-27 (n,p) Mg-27	4.3	9.5 m
2	Al-27 (n,_) Na-24	6.8	15 h
3	Ti-46 (n,p) Sc-46	3.8	84 d
4	Ti-48 (n,p) Sc-48	7.4	44 h
5	Fe-56 (n,p) Mn-56	7.0	2.6 h
6	Co-59 (n,_) Mn-56	8.5	2.6 h
7	Co-59 (n,2n) Co-58	10.8	71 d
8	Zn-64 (n,2n) Zn-63	12.6	38.5 m
9	Zr-90 (n,p) Y-90	7.6	3.2 h
10	Zr-90 (n,2n) Zr-89	12.2	4.2 m
11	Nb-93 (n,2n) Nb-92	9.1	10.1 d
12	Mo-92 (n,2n) Mo-91	12.8	15.5 m
13	Au-197 (n,2n) Au-196	8.8	9.6 h
14	Pb-204(n,n')Pb-204m	2.7	67 m

Table 2:

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Label	Composition	Mass [g]
Mix1	Al (17%), Hf (18%), Cd (32%), Y (21%), Se (5%), $AlBr_3$ (7%)	10.0
Mix2	Y (16%), Se (31%), $AlBr_3$ (53%)	3.9
Mix3	Y (24%), Se (43%), Cd (33%)	3.8
Mix4	Se (52%), Y (48%)	5.0
Mix5	Se (40%), Y (40%), Al (20%)	5.0
Mix6	Y (49%), Se (16%), Fe (15%), Si (20%)	4.9
Mix7	Y (40%), Fe (20%), Si (20%), Cu (20%)	5.0
Mix8	Y (47%), Se (16%), Al (16%), Fe (20%), Au (1%)	5.0
Mix9	Y (48%), Se (16%), Al (16%), Fe (20%)	4.9
Mix10	Y (50%), Se (16%), Al (17%), Fe (17%)	5.9
Mix11	Y (48%), Se (16%), Al (20%), Fe (16%)	4.9
Mix12	Y (40%), Al (30%), Fe (30%)	5.0
Mix13	Fe (40%), Al (55%), Cu (5%)	5.0
Mix14	Y (57%), Cu (43%)	3.5
Mix15	Y (36%), Si (64%)	2.8

Table 3: Multi-element samples manufactured in our laboratory

Probable reaction	Radionuclide
In-113 (n,γ) In-115 (n,2n)	In-114
Mo-98 (n,γ) Mo-100 (n,2n)	Mo-99
Cd-110 (n,γ) Cd-111 (n,n') Cd-112 (n,2n)	Cd-111m
Se-77 (n,n') Se-76 (n,γ)	Se-77m
Se-78 (n,γ) Se-80 (n,2n)	Se-79m
Se-80 (n,γ) Se-82 (n,2n)	Se-81m
Hf-176 (n,γ) Hf-177 (n,n') Hf-178 (n,2n)	Hf-177m
Hf-177 (n,γ) Hf-178 (n,n') Hf-179 (n,2n)	Hf-178m
Hf-178 (n,γ) Hf-179 (n,n') Hf-180 (n,2n)	Hf-179m

Table 4:

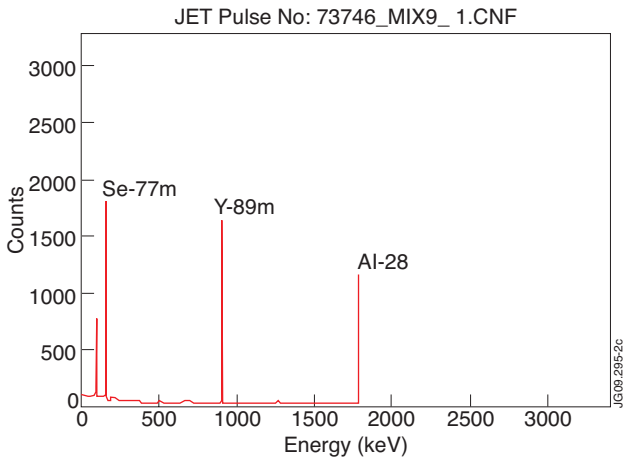

Figure 1: Gamma-spectrum from sample Mix9 irradiated during Pulse No: 73746.

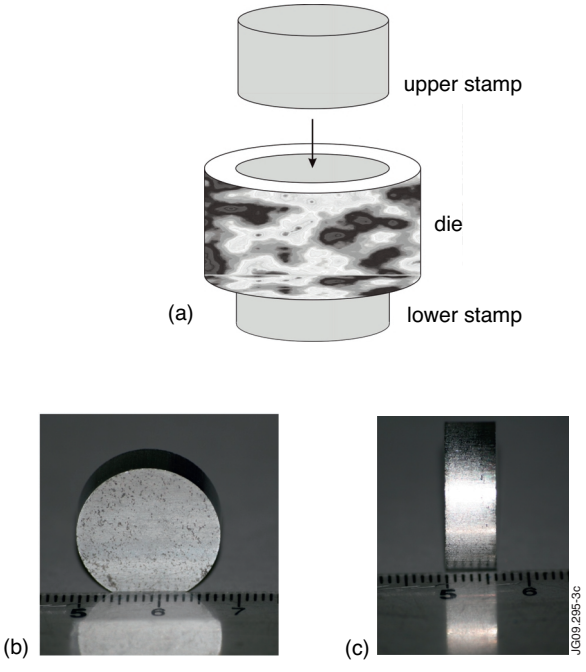

Figure 2: A pictorial view of the die (a) and size of the sample (b), (c).

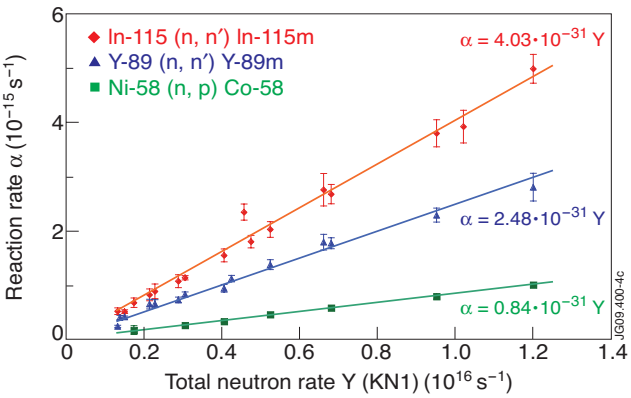

Figure 3: Measured reaction rates for indium, yttrium and nickel samples versus total neutron rate as measured by fission chambers (KN1) for DD discharges. The coefficients of straight lines fitted to the measuring points presented above are corresponding to measured activation coefficients.

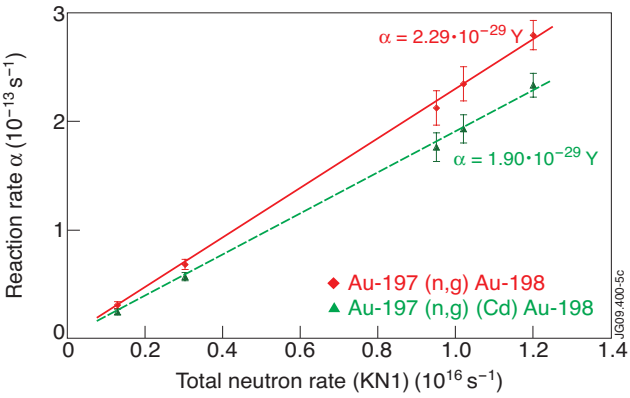

Figure 4: Comparision between Au and Au in Cd measured reaction rate vs total neutron rate measured by fission chambers (KN1).

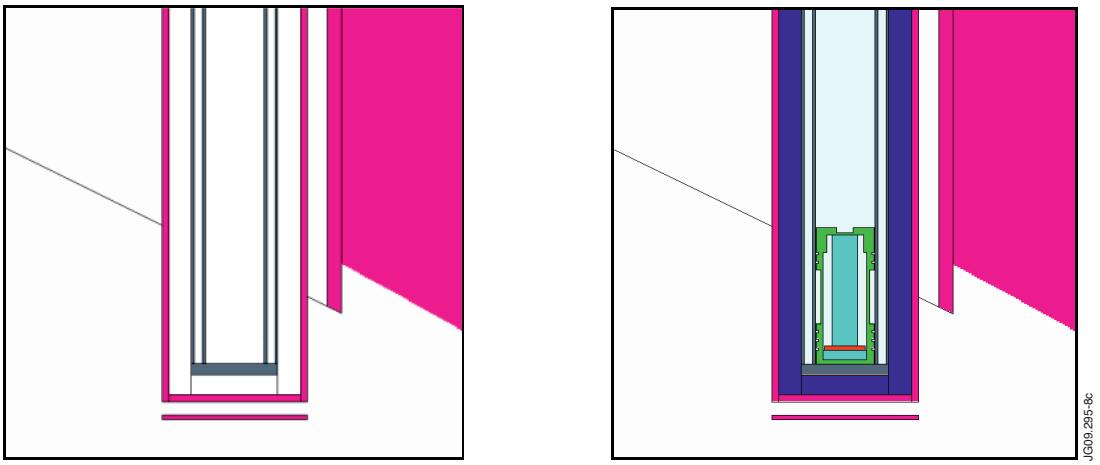

Figure 5: Comparison of the original and supplemented geometry of the 3U irradiation end applied to the MCNP calculations.

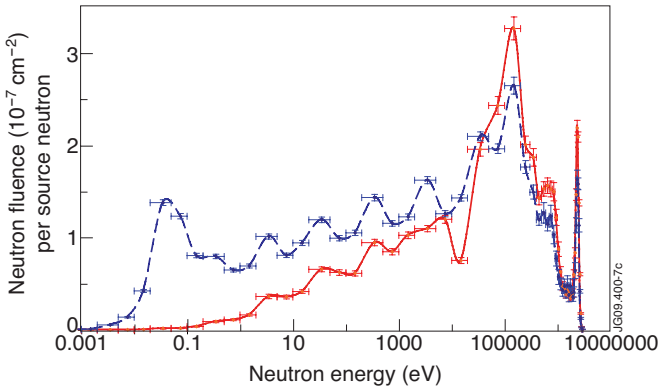

Figure 6: MCNP-calculated neutron spectrum at 3U IE with cooling water (dotted line) and without water (straight line).

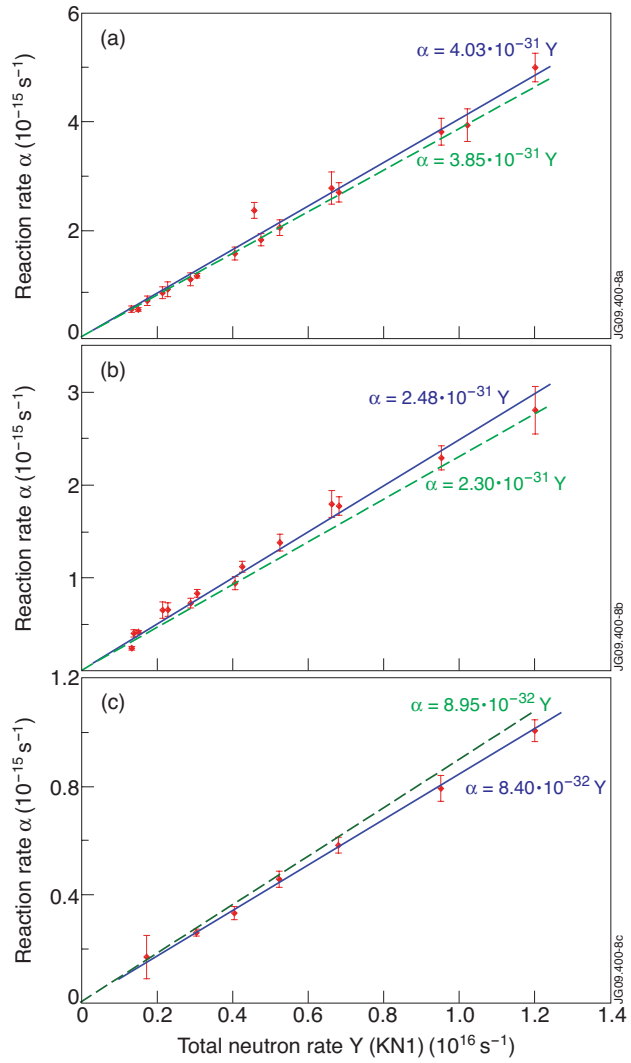

Figure 7: Comparison of the activation coefficients obtained from the MCNP calculations (dotted lines) and those measured by means of activation system - KN2 (straight lines) for a) ^{115}In(n,n), b) ^{89}Y(n,n'), c) ^{58}Ni(n,p) reactions. Marked points come from activation measurements.

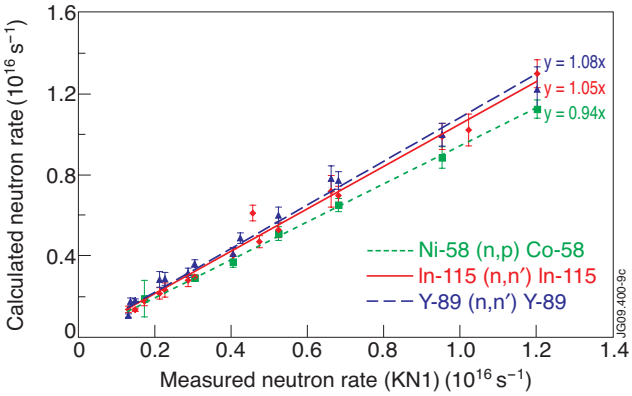

Figure 8: Comparison of the neutron rates obtained from fission chambers and ones calculated by means of the MCNP code.

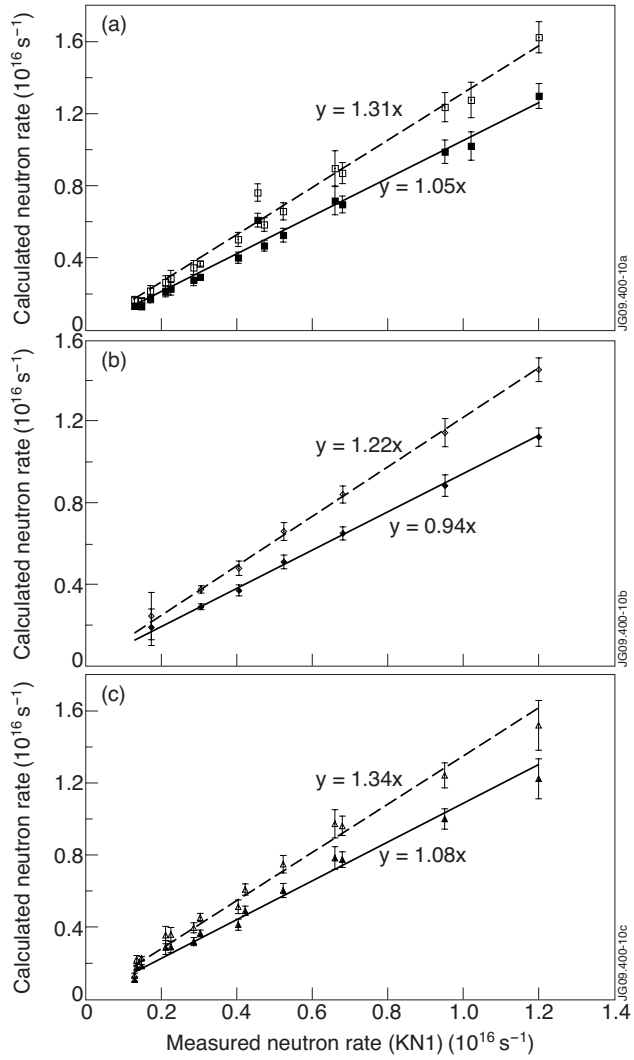

Figure 9: Comparison of the neutron rates obtained from KN1 and those from the MCNP calculations with the supplemented (void points and dotted lines) and original version (filled points and straight lines) of the 3U irradiation end geometry implemented to the calculations (see Fig.5) for a) $^{115}In(n,n')$, b) $^{89}Y(n,n')$, c) $^{58}Ni(n,p)$ reactions.