Analysis of Core Physics Experiments of High Moderation Full MOX LWR

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1. Introduction

Full MOX LWR cores are favorable since they enable a large amount of plutonium to be loaded in a small number of reactors. Higher moderation LWR cores are also favorable to enhance the consumption of plutonium and reduce the residual plutonium in burned MOX fuel. Nuclear Power Engineering Corporation (NUPEC) studied such high moderation full MOX cores as a part of advanced LWR core concept studies from 1994 to 2003 supported by the Ministry of Economy, Trade and Industry. In order to obtain the major physics characteristics of this advanced MOX cores, high moderation full MOX LWR cores, NUPEC carried out the core physics experimental programs called MISTRAL and BASALA in collaboration with CEA in the EOLE critical facility of the Cadarache Center from 1996 to 2002. NUPEC also obtained a part of experimental data of the EPICURE program that CEA had conducted for 30 % Pu recycling in French PWRs under the collaboration with French industrial partners. Those experimental data was transferred to Japan Nuclear Energy Safety Organization (JNES) by March 2005 for further effective utilization.

The analysis of the experimental data was performed by NUPEC from 1996 to 2003 with SRAC, a deterministic code system for pin cell and core calculations, and MVP, a continuous energy Monte Carlo calculation code, based on a common nuclear data library, JENDL-3.2. A part of analysis was also done with JENDL-3.3, ENDF/B-VI and JEF-2.2.

2. Outline of Experimental Programs

The UO2 and MOX fuel rods other than 11 wt% MOX fuel rods used in the experiments have the same geometry of the standard PWR 17x17 assembly with Zry-4 claddings of an outer diameter of 9.5 mm except for the fuel effective length, about 800 mm; the 11 wt% MOX fuel rods have a little smaller diameter that had been used for High Conversion LWR studies before. Those rods are sealed by aluminum over-claddings for adjusting the core moderation ratio and protecting the rods in handling. The MOX pellets except for the 11% MOX fuel are composed of typical reactor grade plutonium with a fissile plutonium content of 60 to 70% and ^{240}Pu content larger than 20% in a depleted UO2 matrix. The total plutonium contents of the MOX pellets are 3.0, 4.3, 7.0 and 8.7wt%. These experimental programs consist of eleven reference critical cores and several derivative cores based on each reference core.

This paper describes an outline of those MOX core physics experiments and summarizes the analysis results that have been published in a large number of papers and others.

Nuclear Data for Design of Reduced Moderation Light Water Reactor

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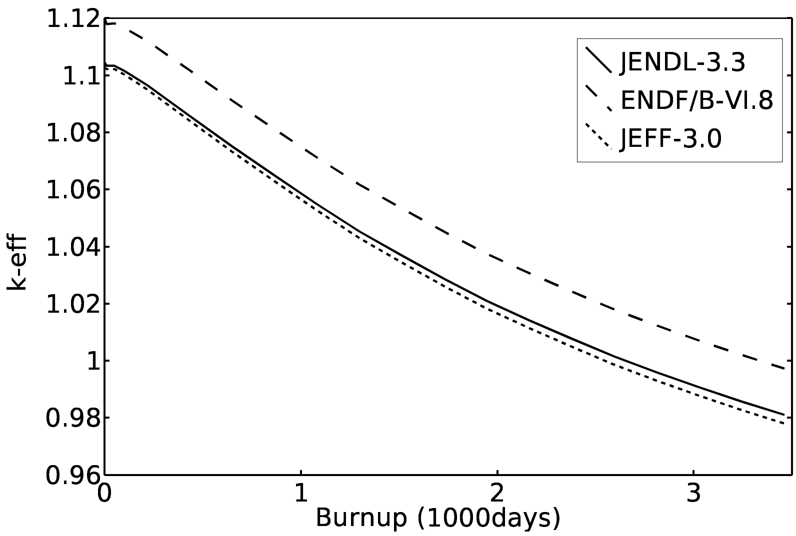
Because of the delay of the fast breeder reactor (FBR) program, the excess plutonium in commercial reactor spent fuels is estimated to already amount more than 1000 ton all over the world. For the efficient utilization of this excess plutonium based on the well experienced light water reactor (LWR) technology, Reduced Moderation Water Reactor (RMWR) concept has been studied in Japan Atomic Energy Agency (JAEA). In RMWR, the conversion ratio of 1.0 is achievable and the plutonium quality (ratio of fissile to total plutonium) can be kept high after burnup. The reactor can therefore sustainably supply energy for a long term through plutonium multiple recycling. The reactor can also act as an active storage of plutonium until the commercial introduction of FBRs.

The current RMWR design is a boiling water reactor (BWR) type, and the very high conversion ratio is realized by reducing the moderator to fuel ratio with the triangular tight pitched fuel lattice and the higher core averaged moderator void fraction than the conventional BWR. Neutronically, the reactor has intermediate neutron spectrum between conventional LWR and FBR. In the reactor core design study, the treatment of the resonance energy region becomes very important. At the lower part of RMWR core, however, the moderator void fraction is not high and the reactions in the thermal energy region is still notable. On the other hand at the upper part of the core, the neutron spectrum becomes very hard due to the high void fraction and the fast energy region is of great importance.

To study the effect of nuclear data uncertainty on the reactor physics characteristics of RMWR, by using a 1-dimensional simplified benchmark calculation model on the axially heterogeneous RMWR core, reactor physics characteristics were estimated with the different nuclear data libraries JENDL-3.3, ENDF/B-VI.8 and JEFF-3.0. As a result, ENDF/B-VI.8 was found to give nearly 1.5% larger effective multiplication factor than JENDL-3.3. This difference corresponds to nearly 500 days or more than 5 GWd/t of burnup period. The difference is caused mainly from the neutron production rate of U-238, Pu-239 and Pu-240 in the fast energy range due to the difference in the fast neutron spectrum calculated with ENDF/B-VI.8 and JENDL-3.3. Other than the multiplication factor, the other important physics characteristics of RMWR, conversion ratio and void reactivity were also compared.



Calculated multiplication factor of 1-d axially heterogeneous RMWR core model of 2 MOX regions with inner, lower and upper blankets

Nuclear data for Non-refueling core design

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The 4S, Super Safe, Small and Simple~~,~~ reactor is a kind of fast reactor core in which burn-up reactivity loss is compensated by decrement of neutron leakage probability with movement of reflector. For extending core life of 4S up to 30 years generating 30MW thermal power, a core of 2.5m height has been ~~is~~ designed and studied in which 20%/24%-Pu-enriched Pu-U-Zr metallic fuel pins are loaded.

For design of innovative control system and safety characteristic of the 4S core of long life, we have to verify and improve neutronics calculation methods. However, there are few experimental data measured focusing on reflector reactivity, small (zero or negative) Na void reactivity, etc. For the verification of the design methods, A series of critical experiments is conducted at the fast critical facility, FCA of JAEA-Tokai. A core of metallic fuels of Pu and Pu+U surrounded by massive reflector of stainless steal has been mocked up and measurement of several kinds of reactivity and reaction rate distribution has been conducted. The measured data have been analyzed by conventional deterministic diffusion / transport codes and continuous energy Monte Carlo codes. By the comparison of calculated one to the data, prediction accuracies of neutronics codes have been clarified.

Using the experimental data, the bias factor, the ratio of measured neutronics characteristics to the calculated ones, is obtained. The uncertainty of this bias factor is also calculated. To quantitatively estimate the uncertainty reduction through critical experiments, an uncertainty reduction ratio (UR) is introduced, using the cross section error. By using UR, better experimental core can be mocked up and required accuracy for experiments have been identified to reduce the uncertainty of the bias factor, i.e., to improve the accuracy of design calculation of the target 4S core.

For achievement of long core life of 30 years without refueling, prediction of burn-up reactivity depletion is important. With sensitivity analysis of the cross sections, important cross sections are clarified for burn-up calculation.

The present study is the results of "Development of Advanced Controlling System for Non-Refueling Reactor Core" in fiscal year 2003 and 2004 entrusted by Ministry of Education, Culture, Sports, Science and Technology (MEXT) to Central Research Institute of Electric Power Industry (CRIEPI).

Impact of Nuclear Data on Design Work for High Temperature Gas-cooled Reactors

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Concerning to nuclear design for the high temperature gas-cooled reactors (HTGRs), the calculation method has been improved with experimental data of the research reactors, such as the HTTR in Japan and the HTR-10 in China. One of the HTGRs type with very high temperature, called VHTR, has been proposed in the Generation IV International Forum.

In nuclear design of HTGRs, several cross sections were interested to characterize the criticality and burn-up situations.

- U-235, Pu-241: (n,f) and (n,g) reactions
- U-239, Pu-239, Pu-240 : (n,g) reaction
- C : elastic and (n,g) reactions
- MAs and LLFPs: generation and transformation

From the recent studies, it is indicated that JENDL-3.3 gives the k_{eff} agreement with the experiments within 1.5%Δk, JENDL-3.2 gives within 1.7%Δk, and ENDF/B-IV.8 and JEFF-3.0 give within 1.8%Δk for the some HTTR core conditions. The k_{eff} discrepancy between JENDL-3.3 and JENDL-3.2 is caused by difference of U-235 fission data and its ratio of (n,f)/(n,g) reaction in neutron energy range of 0.1-1.0eV. There is no discrepancy of k_{eff} value between ENDF/B- IV.8 and JEFF-3.0.

In thermal energy range, the capture cross section of carbon in the nuclear library JENDL is about 4% larger than those of ENDF/B and JEF. From the calculation results of the HTR-10 and HTTR, it was found that the reactivity discrepancy by the carbon capture data is about 0.6%$\Delta k/k$ for criticality analysis, although the section is very small as about 3mb at 2200m/s.

The influence of cross sections of carbon and impact for design work of HTGRs will be cleared by the neutronics calculation using the Monte Carlo code MVP and the diffusion code system SRAC.

Past Organization for Nuclear Data Evaluation in Japan

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Nuclear data activities in Japan were started by organizing Japanese Nuclear Data Committee (JNDC) in 1963. The Nuclear Data Laboratory (former organization of the Nuclear Data Center) was organized in JAERI in 1968. Since then, successful cooperation between JNDC and the Nuclear Data Center has been continued. As a result, these organizations have created various versions of Japanese Evaluated Nuclear Data Library (JENDL). The present talk will review such organizations mainly for nuclear data evaluation work.

Nuclear Data Evaluation Activities in JAEA and the Mid-Term Plan

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Japan Atomic Energy Agency (JAEA) was established at October 1, 2005 after the merger of Japan Atomic Energy Research Institute (JAERI) and Japan Nuclear Fuel Development Corporation (JNC). Missions of the JAEA are followings: (1) Establishment of nuclear fuel cycles, (2) Research and development of nuclear fusion energy, (3) Contribution to hydrogen economy by nuclear process heat, (4) Quantum beam technology, (5) Research on nuclear safety, (6) Non-proliferation and safeguards technology, (7) Decommissioning of nuclear facilities, treatment and disposal of low level waste, (8) Cooperation with academic and industrial communities/ international collaboration/ human resource development/ atomic energy information and (9) Basic nuclear engineering research, advanced basic research. The nuclear data evaluation activities are provided for the mission of (9).

When the JAEA was established, the mid-term plan was issued to the Minister of Education, Culture, Sports, Science and Technology and was approved. The mid-term plan is the work list JAEA promises to perform during the mid-term which is from October 1, 2006 to March 31, 2010. In the mid-term plan the nuclear data activities are written as "With fuel burn-up rate becoming higher, FP and MA nuclide will be playing increasingly important role. Thus, efforts will be made to mainly assess such nuclear data, and to complete JENDL-4, the General-Purpose, Pre-Assessed Nuclear Data Library featuring expansive error data, so to enhance the reliability of nuclear calculation." (From the JAEA English home page.). So the first priority of Nuclear Data Center is to complete the Japanese Evaluated Nuclear Data Library JENDL-4 until the end of the mid-term.

In the presentation, the activities of nuclear data center of JAEA will be discussed.

Human Resources

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The nuclear data is a fundamental data base for nuclear technology and science. It has played a crucial role in the course of nuclear energy development of fission reactors and fusion reactors, and will be so also in the future. Nowadays, the data requirement is extending over various fields such as astrophysics and space technology etc.

The characteristic and important point of "nuclear data" is that it should provide "complete data set" covering all the items required e.g., cross sections, physical quantities like fission yield etc, in consistent manner with accuracy as high as achievable. The accuracy required is very high, e.g., the accuracy required for fission cross section and number of prompt fission neutron of 235U is as high as 1 % or higher.

Such high performance of nuclear data has been achieved and maintained through well-organized collaboration among experiment/ measurement, evaluation and compilation. In addition, in Japan, many benchmark analyses were undertaken by reactor physicist/engineer and they contributed greatly to quality assurance of the data.

Recently, however, reduction of manpower and budget for nuclear data work is becoming a serious problem over the world. One reason will be "perfection" of nuclear data which means the present data files have reached to "satisfactory level" in completeness and accuracy so long as the data for traditional fields is concerned. Nevertheless, nuclear data requirement is extending to "exotic fields" like high energy region, higher actinides, basic fields and medical fields and so on.

To meet such wide requirement and keep the activity of nuclear data society, good human resource and the organization will be key issues. In addition to specialists who are well acquainted with "nuclear data" (evaluation, processing and treatment), collaboration with peoples in nuclear physics, particle physics, and other related fields will be essential for the evaluation of "exotic nuclear data" employing new models and theories developed in the physics fields. For the reason, an appropriate way is highly required to promote the collaboration with physics people in particular young people. Universities in the nuclear engineering field will be expected to provide specialists through education and encouragement to students in the nuclear physics and engineering which are the bases of nuclear data activities.

A Proposal for New Treatment of Radiation Behavior with Combination of Nuclear Data and Reaction Model.

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The nuclear data are extensively used in the Monte Carlo transport calculations to analyze the radiation behavior in various fields such as accelerator facilities, spacecrafts, and radiotherapy. Most of the Monte Carlo transport calculations are based on Boltzmann equation for one-body phase space distribution of the transport particles. By such transport calculations, particularly with the nuclear data, one could obtain only the mean value of the one-body observables in the phase space, e.g. heat, flux, and so on. We cannot calculate the fluctuations around the mean value, since the Boltzmann equation has no information for the two-body and higher order correlations which determine the fluctuation around the mean value.

Recently, however, the higher order quantities, i.e. the fluctuations around the mean values of the one-body observables are often required in a certain field. A typical example for such a correlated quantity is the deposit energy distribution in a cell, which is necessary for the estimation of the response function of the detector or a single event upset probability of a semiconductor memory cell. The solution of the Boltzmann equation cannot describe the distribution but only the mean value. Furthermore, Monte Carlo calculations by using the nuclear data cannot deal with these quantities, since the nuclear data includes only the inclusive one-body cross sections but no information of the correlations.

We have therefore developed a new treatment of radiation behavior in the transport calculations by combining the nuclear data with the reaction models so as to trace all higher correlations. We would like to discuss a possibility of this new treatment in the nuclear data field.

Comparison of Major Nuclear Data Libraries
- JENDL-3.3, ENDF/B-VI.8, ENDF/B-VIIβ1.2, and JEFF-3.1 –

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ENDF/B-VII will be released in July, 2006. It is worthwhile to compare the data of major nuclear data libraries: JENDL-3.3, ENDF/B-VI.8, ENDF/B-VIIβ1.2, and JEFF-3.1. Data comparisons are made for major and minor actinides, long-lived fission products and major structural materials. For example, the capture cross sections of ^{238}U, which most reactor experts should be interested in, are compared in the following table.

Thermal capture cross sections and resonance integrals at 300 K

Library	σ_c at 2200 m/s (b)	I_c (b)
JENDL-3.3	2.718	278.1
ENDF/B-VI.8	2.718	278.1
ENDF/B-VIIβ1.2	2.684	275.3
JEFF-3.1	2.684	275.3
Mughabghab '03[1)]	2.680±0.019	277±3

Not only cross sections but also prompt fission neutron spectra and nu-bar values for some nuclei are compared and discussed in the symposium.

Reference

1) S.F. Mughabghab: "Thermal Neutron Capture Cross Sections and Resonance Integrals and g-factors," International Atomic Energy Agency, INDC(NDS)-440 (2003).

Integral Comparison of Library Performance

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Abstract

The 2003-2004 activity of Reactor Integral Test WG under Subcommittee on Reactor Constants of Japanese Nuclear Data Committee will be presented. During this period, the WG carried out integral tests of JENDL-3.3, ENDF/B-VI and JEFF 3.0 for reactor applications. Some results of integral tests for other latest libraries, JEFF-3.1, ENDF/B-VII, etc. will be also presented.

Nuclear Data Library in Design Calculation

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The characteristics of the lattice codes used for the core design calculation of light-water reactors in Japan are reported from the viewpoint of nuclear data library.

At the time of the light-water reactor introduced into Japan, various technologies including the lattice code were also introduced from the nuclear fuel vender such as W.H. or GE. Those lattice codes are based on ENDF/B-4/5 library. They have been constantly improved, and are widely used for the commercial core design.

Decades have passed since the introduction of initial lattice codes, and domestic venders are currently developing next generation's lattice codes. Some domestic venders have started employing JENDL series (3.2/3.3) as one of the main nuclide data besides ENDF series.

Besides the nuclear data library, the current methodologies of energy groups and resonance calculation will also be reported.

Present Status of CENDL Project

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China Institute of Atomic Energy

Chinese Nuclear Data Committee assumes responsibility the management of CENDL project. CENDL is carried out by China Nuclear Data Center and China Nuclear Data Network. From 1996 to 2001, we have completed the evaluation of CENDL-3.0, total 209 nuclides are include CENDL-3.0. The fission product nuclide file of CENDL-3 has been officially released on October 6, 2001, the other file of CENDL-3.0 have been tested and improved for the problems found in the test within china. The next release version of CENDL is CENDL-3.1 library.

The evaluation for special purpose file was continued . In order to satisfy the need of ADS project of China, a code MEND for calculating the nuclear data in medium energy region has been developed, some nuclear data have been calculated and evaluated.

Utilization of J-PARC
-Research Plan with Neutron-Nucleus Reaction Measurement Facilities-

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We proposed installing "Neutron-Nucleus Reaction Measurement Facilities" in the Materials and Life Science Facility (MLF) in the High-Intensity Proton Accelerator Project (J-PARC: Japan Proton Accelerator Research Complex) in 2002 to conduct three research projects: (1) fast-neutron reaction and nuclear astrophysics, (2) neutron nuclear data on minor actinides and long-lived fission products, and (3) all-elements simultaneous, non-destructive and high-sensitivity nuclide-quantification. Fortunately, the proposal was approved in 2004. MLF will receive the first 3-GeV proton beam by the end of 2007, and will provide test neutron beams for users in 2008. In this contribution, the research plan with our facilities will be presented.

Measurement of Neutron Capture Cross Sections

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The social acceptability of nuclear power reactors is related to the waste management of long-lived fission products (FP) and Minor Actinides (MA) during the burn-up of nuclear fuel. The transmutation is one of ways to reduce the radio toxicity of nuclear waste. In the transmutation study of FP's and MA's, the accurate data of neutron capture cross sections are necessary to evaluate reaction rates by reactor neutrons. In this view point, the cross section measurements have been made by an activation method, neutron time-of-flight (TOF) method and so an.

As for neutron TOF measurement, a high-speed data acquisition system has been developed, which comprising two parallel channels with a flash-ADC shown in Fig.1. One channel is intended for measuring fast neutrons, of which energies range from 10 eV up to several keV. The sampling rate is 40 MHz. The other is operated at a 4 MHz sampling rate for measuring slow neutrons of which energies range down to a few 10 eV. Laboratory tests for the developed system were performed, and the good efficiencies for the incoming counting rate were obtained.

The ^{241}Amd and ^{243}Am nuclides are important in the nuclear waste management, since the presence of these nuclides in the nuclear waste induce long-term radio toxicity because of long-lived alpha emitters. However, there are discrepancies among the reported data for the thermal neutron capture cross section σ_0 of ^{241}Am, which reach more than 20%. In addition, there is a discrepancy among the values for ^{243}Am(n,γ) reaction cross section, which reaches about 10%. In these problems, the cross section measurements were made for the ^{241}Am(n,γ)^{242}Am and ^{243}Am(n,γ)$^{244m+g}$Am reactions.

In the session, the recent activities of cross section measurements will be presented as well as the details of the experiments and the tentative results.

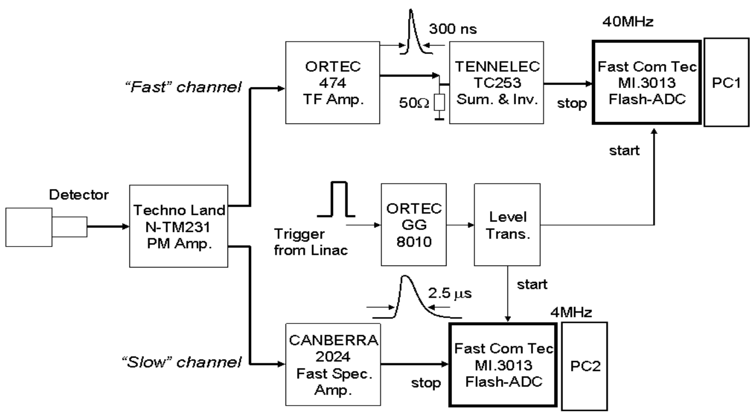

Fig. 1 Block diagram of the data acquisition system with two parallel F-ADC's

Measurement of Cross Section for ^{94}Zr(γ, n) Reaction Using Laser Inverse Compton Gamma rays

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In resent years, nuclear transmutation of minor actinides (MA) and long-lived fission products (LLFP) has drawn a lot of attention. Nuclear transmutation process of LLFP is based on neutron capture reaction. However, the (n, γ) cross sections for LLFP are not well measured in both quality and quantity. The (γ, n) cross section measurement makes it possible to supplement the (n, γ) cross sections.

In the electron storage ring facility TERAS, quasi-mono-energetic γ rays are produced in the energy range of 1 – 30MeV by means of inverse Compton scattering with a Nd: YVO_4 laser and its harmonic modules. The inverse Compton beam line is shown in figure 1. The inverse Compton scattered photons passing through a lead collimator was used to irradiate the target. A ^{94}Zr target material was placed at the center of a neutron detector. The number of γ ray was monitored with a NaI(Tl) scintillation detector located behind the neutron counter.

A schematic view of the neutron detector is shown in figure 2. The neutron detector is composed of twenty ^{3}He proportional counters (CANBERRA/ Dextray: Eurisys Mesures) embedded in a polyethylene moderator. The ^{3}He counters are mounted in 3 concentric rings to achieve high detection efficiency. The diameter of each ring is 76mm, 140mm and 200mm, and the number of the ^{3}He counters of each ring is 4, 8 and 8, respectively.

We measured the (γ, n) cross section for ^{94}Zr from 8.4MeV to 9.8MeV in gamma ray energy. Measured cross sections are presented.

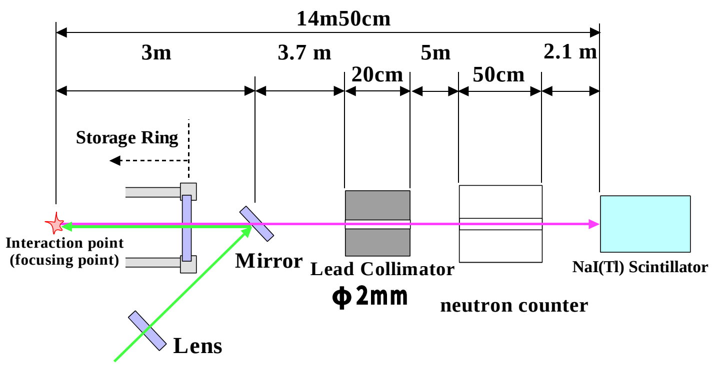

Fig.1. Schematic view of the inverse Compton beam line

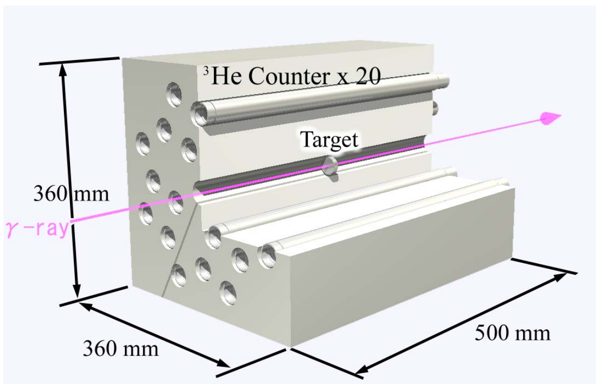

Fig.2. Schematic view of the neutron counter

The investigation of deuteron production double differential cross section induced by 392 MeV protons.

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There are many protons that have intermediate energy in space as a cosmic ray. We have to control the expose dose of human and also determine the cosmic ray affection to apparatus like single event upset of memory. And intermediate energy protons will be used in accelerator driven system (ADS). We should estimate the radiation damage of beam window, beam duct materials and so on.

To realize these technique there are high request of high accuracy evaluated nuclear data. However, around intermediate energy region, nuclear data are still-inadequate.

We have investigated the deuteron productions from 392 MeV proton induced reaction for target nuclei of ^{12}C, ^{27}Al and ^{93}Nb as data of the energy region. The detectors were stacked GSO(Ce) scintillators detectors [1] Fig.1. Deuteron production double differential cross sections were determined over a broad energy range and scattered angles from 20 to 105 degrees in laboratory system. Those spectra were compared with two theoretical models; quantum molecular dynamics (QMD) model [2] and intra nuclear cascade (INC) model. [3,4] For QMD model, we used the JQMD code that was developed by JAERI. [5] Although the calculated (p,p$'x$) double differential cross section spectra had certain degree of accuracy, [6] the spectra of (p,d) double differential cross section spectra didn't have good reproducibility. We developed the code of INC model and we've got good results to reproduce the experimental data.

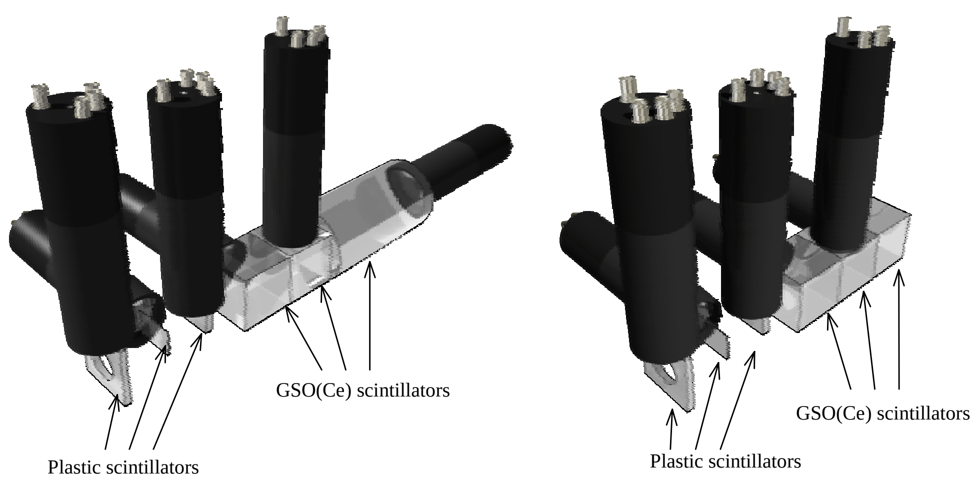

Fig. 1 Stacked GSO(Ce) spectrometers. The long one (left) and the short one (right).

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Measurement of double differential fragment production cross sections of silicon for 70 MeV protons.

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Abstract

Secondary charged particles for neutron-induced reactions are the principle cause of radiation dose and radiation effects of medium to high energy neutrons. Thus far, however, only light charged particles with high production rate have been taken into account for the estimation of the effects. Secondary charged particles heavier than lithium which are called fragment have recently been recognized to be of importance particularly in the fields of the space technology and accelerator applications because they have significant effects even with a single hit due to a large LET (linear energy transfer) in spite of the lesser production rate. A large fraction of radiation effects to semiconductor devices such as single-event upset and latch up on the ground level are thought to be triggered by neutron-induced fragments. However there are no systematic experimental data for neutron-induced fragment production in particular for the energy angular double-differential cross-section (DDX). Even for proton-induced reactions, experimental DDX data are too few to do quantitative assessment of the effects.

We have developed a Bragg curve spectrometer (BCS) suitable to DDX measurements for fragment emission reactions induced by neutrons as well as protons, and obtained the experimental data in ten's of MeV region. The BCS has advantages of a large detector solid angle, large dynamic ranges and good signal-to-background ratio compared with the past similar counters.

To measure systematic DDX data of fragment production for silicon, we used a silicon film (310μg/cm^2) vaporized on a tantalum foil (10μm) as a silicon sample, and measured background events using a pure tantalum foil (10μm). The measurements were performed at the AVF cyclotron facility in NIRS. Fragments from He to O emitted to 30 deg – 135 deg were measured systematically. An energy-time-of-flight (E-TOF) method was also employed at 30 deg. measurement to complement the BCS data.

Experimental DDX data of fragment production obtained by the BCS and E-TOF method are compared with the evaluated data and theoretical calculations such as LA-150, TALYS, INC model (Bertini and Isobar model), QMD model, respectively. The comparisons show systematic deviation of calculations from the experimental data.

Measurement of Double-differential Cross Section of fragments on C, Al, Cu, Ag Induced by 400 MeV Helium

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Fragment production data at intermediate energy are important for evaluations of radiation dose and induced radioactivity. For verification of these evaluations, systematic experimental data are indispensable. The data are also needed to confirm validity of reaction models which are implemented in the theoretical calculations used such evaluations. Our group has started cross section measurements at the energies which enables fragment production. There are two typical methods to measure the fragments, one is an activation method and the other is a direct measurement. An activation method can determine absolute value easier; however the method cannot measure products without activity. A direct measurement can obtain double differential cross section, however it takes long time. We adopt both methods to obtain fragment production data in wide incident energies and target nuclides. In this presentation, devices and results of direct measurement are described.

Experiments are carried out at HIMAC. Incident beam is 400 MeV helium. Samples are Al, Cu, Ag in 10 μm thickness (only C is 100μm thickness). These samples are set at the center of a vacuumed scattering chamber. The chamber also equipped sample holder for activation measurement in front of beam dump. Fragments are measured a detector telescope which consists of Si surface barrier detectors (SSDs). The thicknesses of SSDs are 6μm and 250μm as ΔE and E, respectively. The distance between these two detectors is about 500 mm. This setup allows mass and Z determination simultaneously. Short rise time preamplifiers and a digital storage oscilloscope were adopted to accumulate signals with high time resolution. By using the device, double differential cross sections of ^{6}Li, ^{7}Li, ^{8}Li, ^{7}Be, ^{9}Be, ^{10}Be, ^{9}B, ^{10}B, ^{11}B production are obtained at 30-deg emission angle.

Measurements of cross-sections of producing short-lived nuclei with 14 MeV neutrons. - ^{27}Al(n, α)^{24m}Na, ^{144}Sm(n, 2n)^{143m}Sm, ^{206}Pb(n, 2n)^{205m}Pb, ^{208}Pb(n, 2n)^{207m}Pb -

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There are a lot of data of activation cross-sections with 14 MeV neutrons from a viewpoint of the DT fusion reactor design. In general, most data are long-lived nuclides whose half-lives are longer than several minutes. There are few data for very short-lived nuclides whose half-lives are equal to or shorter than a few seconds, however these data are important for the nuclear data base and improvement of accuracies for the evaluated value. Hence, we aimed to measure the cross-sections producing short-lived nuclei with 14 MeV neutrons by using an in-beam method. This method can measure the induced activities during irradiating the neutron.

The d-T neutrons were generated by bombarding a tritiated titanium (Ti-T) target with a 350 keV d^+-beam at the 0° beam line of the FNS at the JAEA. The experimental arrangement is shown in Fig.1. The distance between the sample position and the surface of the HPGe detector was 50 mm. A typical neutron fluence rate at the sample position was 6.5×10^5 n·cm^{-2}·s^{-1}. The induced activities were measured with 36% HPGe detector. Samples were 1 mm thick rectangular or disk-shape (10 mm × 20 mm, ϕ = 15 mm), and typical weights were 0.045 to 0.95 g.

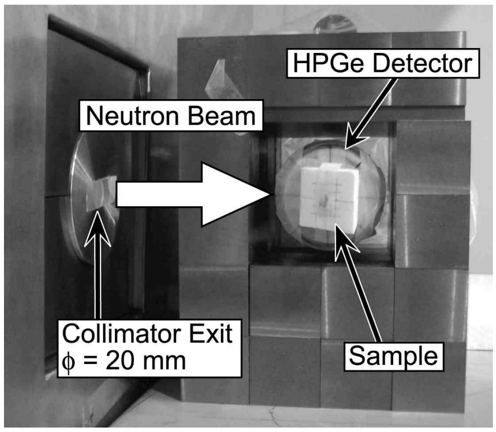

Fig.1 The picture of the experimental arrangement. The distance between the collimator exit and the samples is 150 mm. Tungsten blocks are used to prevent HPGe detector from the background γ-rays.

In order to reduce neutron damage [1], a neutron collimator at the 0 ° beam line (ϕ = 20 mm, E_n = 14.2 MeV) was used. The damage by neutrons, which were scattered by samples, its holder and an atmosphere, were taken into account using the Monte-Carlo simulation code "MCNP-4C". As a result, it was found that the neutron fluence rate at the surface of the HPGe detector was 4.0×10^{-4} times against the sample position.

The cross-section data of ^{27}Al(n, α)^{24m}Na ($T_{1/2}$=20.20 ms), ^{144}Sm(n, 2n)^{143m}Sm (30 ms), ^{206}Pb(n, 2n)^{205m}Pb (5.54 ms), ^{208}Pb(n, 2n)^{207m}Pb (806 ms) reactions were obtained by the in-beam method. That of ^{144}Sm(n, 2n)^{143m}Sm was measured for the first time. Accuracies were 4.4 to 23%, These accuracies were mainly caused by statistics.

The result of the ^{27}Al (n, α) ^{24m}Na reaction is shown in Fig.2. An effect of ^{27}Al(n, α)^{24m}Na reaction that the scattered neutron interacts with the Al of HPGe detector housing was corrected. The amount of the correction was about 12%. The evaluated data for the ^{27}Al (n, α) ^{24m}Na listed in FENDL/A-2.0 were underestimated 0.63 times as small as the present result, approximately. Other previous experimental data were also larger than the evaluated one, re-evaluation for this reaction is recommended.

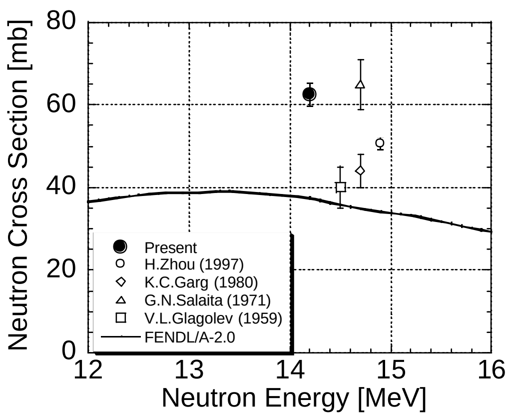

Fig.2 The cross-section of ^{27}Al (n, α) ^{24m}Na reaction ($T_{1/2}$=20.20 ms). The solid line indicate the evaluated data of FENDL/A-2.0. A solid circle shows the present data.

We measured the cross-sections producing short-lived nuclei whose half-lives are between 5.54 and 806 ms by the in-beam method.

Reference [1] M. Sudarshan et al., Meas. Sci. Technol. 2 (1991) 1192-1194.

Measurement of Angle-correlated Differential (n,2n) Reaction Cross Section with Pencil-beam DT Neutron Source

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The (n,2n) reaction is a neutron multiplication reaction. And its cross-sections is needed to design a nuclear fusion reactor. Especially, the ^{9}Be(n,2n) cross-section is very important because beryllium is one of the candidate for neutron multiplier in a fusion reactor. In the previous experiments, the foil activation method was generally used to measure (n,2n) reaction cross-sections. However this method cannot be applied to the element that cannot make measurable isotopes by activation. And this method cannot give energy and angular distribution. Therefore there are many elements of which experimental value of (n,2n) reaction cross-section have not been measured.

In the present experiment, the coincidence detection technique was used. Two spherical NE213 detectors (4cm in diameter) were located at 18.8cm from a beryllium sample (2cm in diameter, 2cm long), and two neutrons simultaneously emitted by the (n,2n) reaction were detected by each detector. A pencil-beam DT neutron source of the Fusion Nuetronics Source (FNS) in Japan Atomic Energy Agency (JAEA) was used to remove background signals caused by directly incoming source neutrons and room returned neutrons. In addition, time difference spectrum between anode signals of the two detectors was measured to pick up the coincidence events from (n,2n) reaction.

Three angle parameters (θ_0, θ and ϕ) with respect to the detector position to measure azimuthal and longitudinal neutron emission distribution were defined as shown in Fig.1. Measurement points were decided by combination of these angle parameters for two detectors. Experiments for 9Be(n,2n) reaction have been done by about fifteen points.

Measured pulse height spectra have to be transformed into energy spectra. Energy spectra were obtained by unfolding pulse height spectra using SCINFUL and FORIST code. The necessary detector response matrix was estimated with the SCINFUL code. As a result, there was a strong forward oriented longitudinal distribution as shown in Fig.2. While, a slightly forward oriented azimuthal distribution was confirmed although it is not considered in the evaluated nuclear data. Total ^{9}Be(n,2n) cross-section was obtained by integrating energy and angular distributions by fitting with Legendre polynomials. The Total cross-section is agreed with JENDL-3.3.

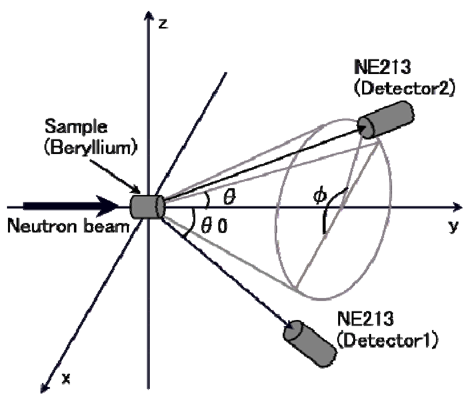

Fig.1 Experimental arrangement around detectors.

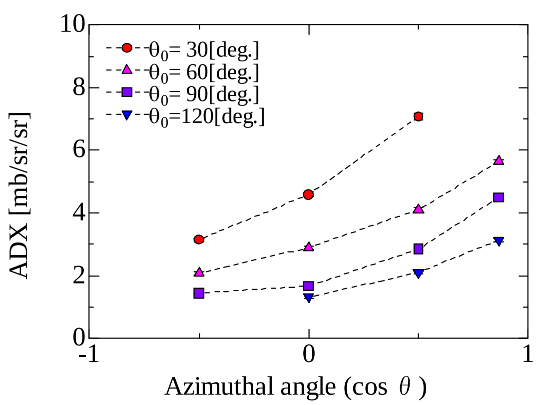

Fig.2 Angular distribution for axial direction.

Study on keV-neutron capture cross sections and capture gamma-ray spectra of ^{117,119}Sn

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The neutron capture cross sections of long-lived fission products (LLFPs) are important physical quantities for the study on the transmutation of radioactive nuclear wastes. The nuclide ^{126}Sn is one of the LLFPs. However, there is no experimental data for ^{126}Sn, because the preparation of high-purity sample is difficult and, moreover, gamma-ray radiation from a sample causes a serious background.

On the other hand, keV-neutron capture cross sections and capture gamma-ray spectra of stable Sn isotopes contain important information which is useful for the evaluation of capture cross sections of ^{126}Sn. Thus, we have started a systematic measurement and calculations of keV-neutron capture cross sections and capture gamma-ray spectra of stable Sn isotopes. In the present contribution, the results for $^{117, 119}$Sn are shown.

The capture cross sections and capture gamma-ray spectra of $^{117, 119}$Sn were measured in the incident neutron energy region from 10 to 100 keV and at 550 keV, using the 3-MV Pelletron accelerator of the Research Laboratory for Nuclear Reactors at the Tokyo Institute of Technology. Pulsed keV neutrons were produced from the ^{7}Li(p,n)^{7}Be reaction with a 1.5-ns bunched proton beam from the accelerator. The $^{117, 119}$Sn samples were highly enriched metal plates, and the net weight of each sample was about 1 g. Capture gamma rays were detected with a large anti-Compton NaI(Tl) spectrometer by means of a time-of-flight method. A pulse-height weighting technique was applied to the observed capture gamma-ray pulse-height spectra to obtain capture yields. Using the standard capture cross sections of ^{197}Au, the capture cross sections of $^{117, 119}$Sn were derived with the error of about 5%. Capture gamma-spectra were derived by unfolding the observed capture gamma-ray pulse-height spectra. The present cross section results were compared with other experimental data and the evaluated values in JENDL-3.3 and ENDF/B-VI.

The calculation of capture cross sections and capture gamma-ray spectra of $^{117, 119}$Sn were performed with the EMPIRE-II code in the incident neutron energy region from 10 to 1000 keV. The calculated results were compared with the present experimental results.

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Measurement of Fission Cross-Sections with Lead Slowing-down Spectrometer using Digital Signal Processing

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For transmutation of minor actinide (MA) in nuclear waste and generation of electricity by Accelerator Driven System (ADS), a variety kind of nuclear data are needed. The nuclear data are essential for calculation of reactor characteristics such as critical safety, kinetics, decay heat and so on. Especially, neutron induced fission cross-sections are crucial because the transmutation of the nuclear waste is based on a fission reaction.

However, the present nuclear data are not enough in quality and quantity. For example, evaluated neutron induced fission cross-sections of ^{237}Np show marked discrepancies.

Therefore, the present study aims to measure neutron induced fission cross-sections of actinide nuclei and to contribute the improvement of nuclear data.

The neutron induced fission cross-sections of ^{237}Np and ^{241}Am have been measured relative to that of ^{235}U with a back-to-back type double fission chamber. These nuclei account for the greater part of the nuclear waste. The measurement was performed using the Kyoto University lead slowing-down spectrometer driven by an electron linear accelerator (KULS).

In the present work, the energy region was extended to about 1 MeV owing to the reduction of the electromagnetic noise with digital signal processing (DSP). DSP is the method of acquiring various information by analyzing the digital waveform data.

The results are compared with the evaluated data of JENDL-3.3, ENDF/B-VI.8 and JEFF-3.1.

Effect of ^{140}Ba Fission Yield on Fission Rate Distribution Measurements in UO2-MOX Mixed Core of REBUS Program

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1. Introduction

In core physics experiments a fission rate distribution is one of the essential data that is used to validate the core analysis methods. The measurements of this parameter have been adopting spectroscopy of specific gamma-rays from fission products, such as 1,596.5 keV gamma-rays from ^{140}Ba ($T_{1/2}$=12.752d) - ^{140}La (T1/2=1.6781d) after short period irradiation of experimental cores. When this method is applied to UO_2 - MOX fuel mixed cores, it is necessary to take into account the difference of the fission yield of ^{140}Ba in the UO_2 and the MOX fuel. For instance, the JNDC Nuclear Data Library of Fission Products[1)] shows that the cumulative fission yield of ^{140}Ba is 6.295 % for ^{235}U-thermal fission and 5.545 % for ^{239}Pu-thermal fission.

Japan Nuclear Energy Safety Organization (JNES) has been participating in the REBUS international program organized by Belgonucleaire and SCK/CEN. The aim of the participation is to obtain measured reactivity change with burn-up of MOX fuel and UO2 fuel and the fission rate and the flux distribution of the cores containing burned MOX and UO2 fuel and analyze these data in order to validate nuclear core analysis methodologies for burned MOX and UO2 cores. The program partly contains UO2 - MOX mixed cores and a fission rate distribution has been measured with the gamma-ray spectroscopy of 1,596.5 keV gamma-rays from ^{140}La.

We have studied an effect of the ^{140}Ba fission yield on the measured fission rate distribution through the analysis of a UO2 - MOX fuel mixed core of the REBUS program.

2. Summary of Study

(1) The ratio of fission rate of the MOX and the UO_2 fuel rods depends on the cumulative fission yields of ^{140}Ba that is used in the process of the experimental data, (2) The difference in the ^{140}Ba fission yield for the ^{239}Pu thermal fission among the nuclear libraries, JENDL-3.2, ENDF/B-VI and JEF-2.2, is up to 5 % and not negligible. (3) The fission yield data of ^{140}Ba used in the process of the experimental data should be precisely reviewed to evaluate the calculation errors for the ratio of the fission rate of the MOX and the UO_2 fuel rods in the UO2 - MOX mixed cores, (4) Effort to decrease uncertainty of the fission yield data of ^{140}Ba for ^{239}Pu (Thermal fission) is requested for the precise evaluation of the calculation errors of the fission rate distributions in UO2 - MOX mixed cores.

Measurement of neutron production spectra at the forward direction from thick graphite, aluminum, iron and lead targets bombarded by 250 MeV protons

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Various Monte Carlo transport calculation codes have been widely employed for the shielding designs of proton accelerator facilities. In such designs by use of simulation codes, it is important to estimate the production spectra of secondary particles, especially neutrons, produced by beam losses in thick materials of beam line modules and the beam dump as source terms. The accuracy of calculated results has been verified by the benchmark experimental data. The double differential neutron production spectra at 0-degree by bombarding 210 MeV protons on a thick iron target were measured at RIKEN [1]. No other experimental data are available to confirm the accuracy of neutron production spectra at 0-degree from thick target. We have measured neutron production spectra from thick graphite, aluminum, iron and lead targets at the forward direction bombarded by 250 and 350 MeV protons at the TOF course of the RCNP (Research Center of Nuclear Physics) ring cyclotron of Osaka University. The 350 MeV measurement results were already presented [2]. In this work, the 250 MeV measurements are reported.

The experimental results were compared with the calculated ones by the Monte Carlo particle transport code, the PHITS and MCNPX codes. In these calculations, the JENDL/HE2004 and the LA150 evaluated neutron data libraries for energies up to 150 MeV and the Bertini model based on intranuclear cascade model were employed.

Figure 1 shows the measured and the calculated neutron production energy spectra from the graphite target. The difference of calculation results between using the JENDL/HE2004 and the LA150 is very small. A discrepancy of the results between PHITS and MCNPX is observed above 200 MeV. This is come from the difference of the type of Bertini model. All calculation results underestimate the experimental ones in the neutron energy range between 30 MeV and 200 MeV. The underestimations of the calculations are also found in 210 MeV proton incident experiment at RIKEN [1] and 350 MeV experiment at RCNP [2]. Those may result from the underestimation of neutron-production cross sections at small angles and the strong self-shielding in target nucleus.

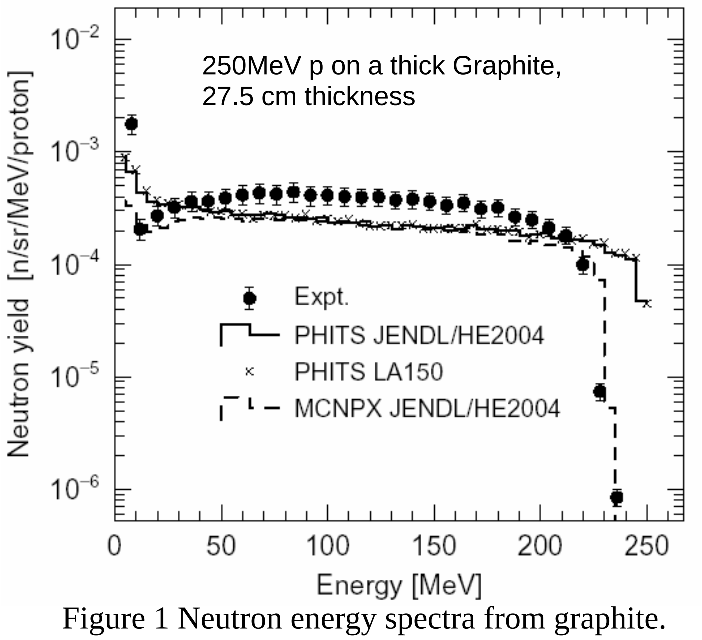

Figure 1 Neutron energy spectra from graphite.

[1] S. Yonai, T. Kurosawa, H. Iwase, H. Yashima, Y. Uwamino, T. Nakamura, Nucl. Instr. and Meth. A 515 (2003) 733.

[2] Y. Iwamoto, S. Taniguchi, N. Nakao, T. Itoga, T. Nakamura et al., the AccApp05 Proceedings

Measurement of 40 MeV Deuteron Induced Reaction on Fe and Ta for Neutron Emission Spectrum and Activation Cross Section

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To improve the data accuracy of the neutron emission spectra and the activation cross section for the deuteron interaction with ^{nat}Ta and ^{nat}Fe which will be used as the structural materials in IFMIF(International Fusion Materials Irradiation Facility), we have measured the 1) differential thick target neutron yields from tantalum and iron targets bombarded by 40 MeV deuterons and the 2) excitation functions of activation cross sections for deuteron interaction with tantalum and iron up to 40 MeV, at the AVF cyclotron (K=110) facility of Tohoku University. The activation data were obtained with the stacked target technique and by detecting γ-rays with a high pure Ge detector.

The neutron spectra were measured at seven laboratory angles between 0- and 110-deg with the time-of-flight (TOF) method using a beam swinger system and a well collimated neutron flight channel, and obtained over the almost entire energy range of secondary neutrons using a two-gain method. In the neutron spectra, large peaks around 15 MeV appears in the case of 0-15 deg., and the spectra show very strong angular dependence.

The activation cross-sections were obtained for the ^{nat}Fe(d,x)^{51}Cr, ^{nat}Fe(d,x)^{52}Mn, ^{nat}Fe(d,x)^{56}Co, ^{nat}Fe(d,x)^{57}Co and ^{nat}Fe(d,x)^{58}Co reactions, and are compared with other experimental data, the evaluated data by IAEA and calculations by a recent code TALYS. The present data is generally consistent with other experimental data. The TALYS results are similar to the experimental data in higher energy region, but much lower in magnitude in the lower energy region. The present data for the ^{nat}Fe(d,x)^{56}Co, ^{57}Co and ^{58}Co reactions are consistent with other experimental data and evaluated data except for the TALYS results. To estimate radioactivity induced by deuterons, therefore, improvements will be required for cross-section calculation models or parameters in TALYS.

Analysis of Induced-radioactivity using DCHAIN-SP for Light Nuclei at a Mercury Target Irradiated by 2.8 or 24 GeV Protons

T. Kai, Y. Kasugai, F. Maekawa, S. Meigo, H. Takada, Y. Ikeda,

Japan Atomic Energy Agency

Reliability estimation was carried out on a radioactivity calculation code system consisting of PHITS, MCNP/4C and DCHAIN-SP 2001 by analyzing an activation experiment, which was performed by using AGS (Alternative Gradient Synchrotron) accelerator at Brookhaven National Laboratory. In the experiment, a cylindrical mercury target having hemispherical head was bombarded with 2.83- and 24-GeV protons, and 13 kinds of samples were irradiated on top and side surfaces of the mercury target. At the top surface of the target, the samples were irradiated with the incident protons and spallation neutrons. Those on the side surface of the target were mainly irradiated by spallation neutrons. After the irradiation, the radioactivity of each sample was measured by using HPGe detectors at the cooling time between 2 h and 267 d. The number of protons injected to the mercury target and the samples on the top surface were determined by an integrating current transformer (ICT) and an activation method using a copper foil, respectively. As for the activation method, the Cu(p,x)^{24}Na reaction was adopted as a reference. Experimental results for boron-10, boron-11, carbon, aluminum, iron and copper samples are discussed in this presentation.

In the analysis, the calculation model included the mercury target, a target container of stainless steel, all samples and concrete walls of the irradiation room. Proton beam profile was assumed to be a Gaussian distribution which was measured parameters of a full width at half maximum (FWHM) and the center of proton beam. For each sample, proton spectrum was calculated by PHITS, and neutron spectrum was obtained by PHITS (>20 MeV) and MCNP/4C (<20 MeV). Using the proton energy spectra and the neutron energy spectra above 20 MeV, nuclear production yields were calculated by PHITS. DCHAIN-SP calculated the radioactivity of the samples by using the nuclear production yields and the neutron energy spectra below 20 MeV. For the samples on the top surface of the mercury target, the proton energy spectra were normalized to the number of protons obtained by the activation method using the copper foil. The proton energy spectra of the samples on the side surface and the neutron energy spectra were normalized to the number of incident protons measured by ICT.

Comparisons between the calculated results and the measured radioactivity will be presented.

Analysis of Induced-radioactivity using DCHAIN-SP for Heavy Nuclei at a Mercury Target Irradiated by 2.8 and 24 GeV Protons.

Y. Kasugai, T. Kai, F. Maekawa, S. Meigo, Y. Ikeda.

Japan Atomic Energy Agency

Radioactivity estimation in spallation neutron field including incident high-energy protons is essential for designing spallation neutron target and accelerator driven nuclear transmutation system. In particular, the estimation for heavy material such as mercury, lead and bismuth is important since those elements are used as target materials. However, the radioactivity estimation for such heavy material has not been easy and reliable, because the products cover wide range of nuclei, consideration of huge kinds of reaction paths is required, and the most of reaction cross sections for high energy incident particles are unknown or unreliable.

A radioactivity calculation code consisting PHITS, MCNP4C and DCHAIN-SP 2001 has been used for the radioactivity estimation for design of the J-PARC facilities. In this work, we validate the code system for mercury, lead and bismuth samples by the experimental activation data obtained using AGS (Alternative Gradient Synchrotrons) accelerator at Brookhaven National Laboratory.

In the experiment, the samples of mercury-oxide, lead and bismuth were irradiated around the mercury target, which was bombarded with 2.83 and 24 GeV protons. The samples were placed at the top and side of the target. The top samples were irradiated by incident protons in addition to secondary neutrons, and the side samples were mainly irradiated by the spallation neutrons from the target. The number of protons injected to each top sample were determined by the foil activation method using the reference reaction of Cu(p, x)^{24}Na. The total incident protons were measured by an integrating current transformer (ICT) and separated electron chamber (SEC). The neutron flux at the side samples were validated using ^{93}Nb(n, 2n)^{92m}Nb reaction. After the irradiation, the radioactivities of samples were measured with HPGe detectors at the cooling time between 2 h and 267 d. The detail of the experimental procedure and the experimental data were shown in the reference [1]. The calculation procedure using the calculation code system is described in the abstract of this symposium [2].

As the results of the comparison between the calculated values and the experimental data, we found that the calculations underestimated the activities by a factor of 2 on the average. We also found that the C/E-values have the dependence on mass numbers of products, which implies that the calculated mass yield curve for spallation reactions shows different tendency from the real one. We will present the comparison between the calculated mass yield curve and the experimental-basis mass yield curve deduced from the activation data.

References

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[2] T. Kat, et al., "Analysis of Induced-radioactivity using DCHAIN-SP for Light Nuclei at a Mercury Target Irradiated by 2.8 and 24 GeV Protons", to be presented in this conference.

Resonance Analysis Combined with Optical Model

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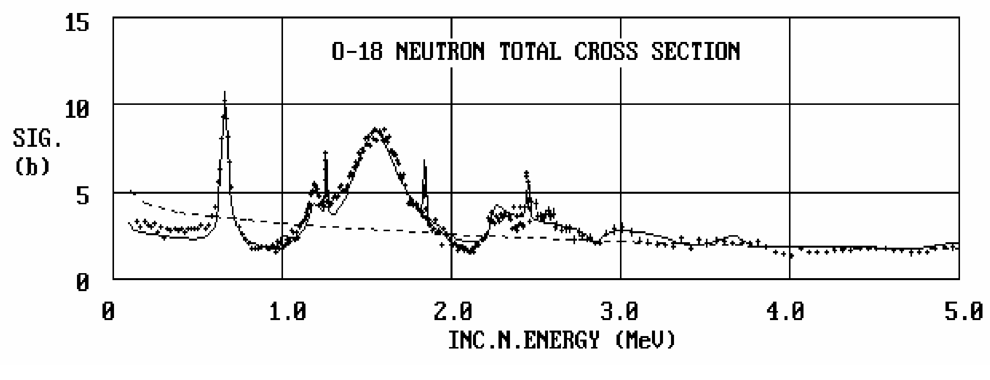
Cross sections of light nucleus show composite resonance structures in the incident energy range over 10 MeV. Ordinary method of resonance analysis requires frequently some wide resonances in the outside of the objective resonance region, which can not be explained physically. These wide resonances could be predicted by the optical model, as some dispersed single particle states, using adequate potential parameters. So, combination of resonance analysis and the optical model will be useful to analyze composite resonance region cross sections and effective to obtain continuation of nuclear data between resonance region and higher energy region that can be analyzed with the optical model.

In the present model, the collision matrix U which describes the neutron elastic channel is given by the sum of optical model U_{opt} and R-matrix U_R, for the same spin-parity state, such as

$$U^{J\Pi}=kU_{opt}^{J\Pi} + (1-k)U_R^{J\Pi},$$

where k is the optical model weight factor and resonance term weight was determined to hold U unitary.

The following figure shows an example of the present calculation (solid line); neutron total cross section of ^{18}O comparing with the experimental data[1),2)]. The dashed line shows the optical model calculation. The optical model U_{opt} was calculated with ELIESE-3 code[3)] using Wilmore-Hodgeson potential parameters[4)] and the resonance U_R was calculated with the approximated R-matrix code[5)] using 14 resonance levels and weight factor k=0.35.



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Analysis of Fission with Selective Channel Scission Model

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The selective channel scission model [1-3] has been proposed to calculate the fission product yield. This model deals with the fission process with each channel. The fission product yield is obtained from the penetrability of the channel-dependent fission barrier. However, the channel-dependent fission barrier has not been calculated theoretically. Therefore the adjustable parameter which is concerned with the elongation and the deformation of the nucleus and the Coulomb potential between two fission fragments has been introduced to obtain the channel-dependent fission barrier. And the correlation between the parameter and the fission modes on the multimodal random-neck rupture model has been shown.

In this work, the mass distributions are calculated on simple assumptions about the channel-dependent fission barriers and potentials. 1) The channel-dependent fission barrier is approximated by the difference between Q-value and the Coulomb potential between two fragments of the channel. This Coulomb potential is estimated at the distance which is the sum of the radii of two fragments at the grand state and the distance of the nuclear interaction, which is 1-2 fm. The deformations of nuclei are reported in some mass evaluations, for example KTUY [4] and FRDM [5]. This assumption is related to fusion as the reversal process, although these processes are not equivalent completely because of the irreversible process such as dissipation which causes neutron emission from the fission fragments. 2) The curvatures of the potential near the saddle point, which are not able to be calculated directly in this analysis, are assumed as constant for all channels for simplicity.

The mass distributions for some fission reactions will be reported in this conference. Also, the relation between the multimodal random-neck rupture model and the shape elongation calculated in this work will be discussed.

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Investigation of Nuclear Reaction Data for Analyses of Single-event Effects in Semiconductor Devices

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In recent years, nucleon-induced single-event upsets (SEUs) have always been a serious concern for microelectronic devices employed in various radiation environments. For instance, terrestrial cosmic-ray neutrons hitting the earth have the wide energy range from MeV to GeV, and are regarded as one of major sources of the SEUs in the devices used on the ground or in airplanes. And, protons are very important for the SEUs in the devices used in the space environment.

In a microscopic scale, the nucleon-induced SEU process starts from the interaction of an incident cosmic-ray particle with materials in a device. Then, light charged particles and heavy recoils are generated via the nuclear reaction with atomic nucleus. They give rise to local charge burst in a sub micron-size volume while passing through the device, which result in upsets of the memory cell information quantum. Therefore, one needs nuclear reaction data that provides us with the probability of the initial interaction of neutrons with device materials, when estimating the SEU rate by numerical simulations.

We have developed a cross section database of ^{28}Si for incident neutron energies ranging from 2MeV to 3GeV, on the basis of evaluated nuclear data files and nuclear model calculations. The database has been applied to simulation of the initial processes in the SEU phenomena, *i.e.*, the nucleon-induced nuclear reaction and the sequential charge deposit by secondary ions, using a simplified device model. [1)] In this study, we have investigated three effects (simultaneous multi-ions emission, energy spectral shape of secondary ions, and elastic scattering) which are expected to be important for initial processes in nucleon-induced SEU phenomena. In addition, we discuss a method of estimating nucleon-induced SEU cross-section empirically from experimental heavy-ion SEU data on the basis of the simulation of the initial processes.

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Improvement of Prediction Power of FP Summation Calculations by Use of the TAGS Experimental Data

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In May 2005, the European evaluated nuclear data library JEFF3.1 was released, which includes the FP decay data. It is noteworthy that the discrepancy in the gamma-ray component of the decay heat is not improved from the previous version, JEF2.2, and becomes even worse (Fig.1). The beta-ray component is also discrepant (overestimation).

On doing fission product (FP) decay heat calculations, we have to pay attention to the problem that the beta-transitions to the highly-excited levels are apt to be lost, which is known as the pandemonium problem. The calculated results based on JENDL are quite consistent with the sample-irradiation measurements that were performed world wide. On the other hand, JEFF3.1 could not reproduce the integral measurements so well. It is because JENDL is made up of experimental data with supplementation of the nuclear model calculation on the basis of the gross theory of beta decay. On the contrary, JEFF3.1 is composed only of experimental data except for no data nuclides.

In the early 1990's, Total Absorption Gamma-ray Spectrometer (TAGS) measurement was carried out at INEL (Idaho National Engineering Laboratory) for 45 isotopes. One of the most important properties of TAGS measurement is that it is expected to be pandemonium problem free. In this respect, the TAGS measurement is considered that it may provide a solid basis of the summation calculations.

On introduction of TAGS data into the summation calculations, the average beta- and gamma-ray energy par decay (E_β, E_γ) are replaced by the TAGS-origin values for these 45 nuclides. As a result, JEFF3.1 becomes quite consistent with sample-irradiation measurements. The excellent agreement on JENDL, however, is no longer maintained. It is probably because that the theoretical correction of JENDL is valid only on the average property of the nuclides, whereas TAGS measurement provides the information of each individual nuclide. Therefore, we must get the more TAGS data.

Now some isotopes that should be measured on the future TAGS measurement are suggested. They are ^{98}Nb, ^{100}Nb, ^{105}Mo and ^{102}Tc. Their isotopes have great difference between JENDL and JEF2.2 after introduction of the TAGS data. It is shown in Fig.2. At the same time, it is necessary to measure a couple of 45 isotopes again to verify the reliability of TAGS measurement. The result based on JEFF3.1 will be presented at the meeting.

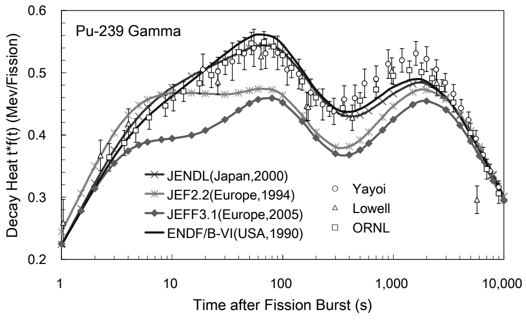

Fig.1 Effect of introducing TAGS energies into summation calculations Pu-239 fission burst, Gamma-ray component

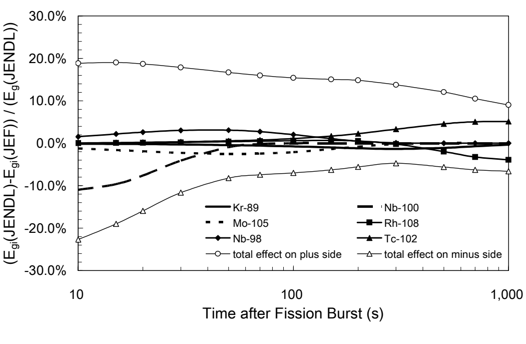

Fig.2 Nuclide-wise contributions to the difference between JENDL and JEF2.2 after introduction of the TAGS data measurement

Density distributions and form factors in neutron-rich nuclei

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Results of charge form factors calculations for several unstable neutron-rich isotopes of light, medium and heavy nuclei (He, Li, Ni, Kr, Sn) are presented and compared to those of stable isotopes in the same isotopic chain. For the lighter isotopes (He and Li) the proton and neutron densities are obtained within a microscopic large-scale shell-model, while for heavier ones Ni, Kr and Sn the densities are calculated in deformed self-consistent mean-field Skyrme HF+BCS method. We also compare proton densities to matter densities together with their rms radii and diffuseness parameter values. Whenever possible comparison of form factors, densities and rms radii with available experimental data is also performed. Calculations of form factors are carried out both in plane wave Born approximation (PWBA) and in distorted wave Born approximation (DWBA). These form factors are suggested as predictions for the future experiments on the electron-radioactive beam colliders where the effect of the neutron halo or skin on the proton distributions in exotic nuclei is planned to be studied and thereby the various theoretical models of exotic nuclei will be tested.

Burn-up Calculation of Fusion-Fission Hybrid Reactor Using Thorium Cycle

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A fusion-fission hybrid system consists of fusion reactor with blanket containing nuclear fuel. Even for a relatively lower plasma condition, neutrons can be well multiplied by fission in the nuclear fuel and tritium is thus bread so as to attain its self-sufficiency. Enough energy multiplication is then expected and moreover nuclear waste incineration is possible.

In the present study, a burn-up calculation system has been developed to estimate performance of a fusion blanket under subcritical condition. In this system, track length data of neutron for each cell are stored in the MCNP-4B calculation. The data are fed directly to a routine for evaluation of one group cross-sections for burn-up calculation. The one group cross-section is made by the product of the track length data and the pointwise cross-section. With the cross-sections, burn-up calculation was performed by ORIGEN2. Burn-up cycle was repeated for necessary times.

A 3-dimensional ITER model was used as a base fusion reactor. The nuclear fuel (reprocessed plutonium as the fissile materials mixed with thorium as the fertile materials), transmutation materials (minor actinides and long-lived fission products) and tritium breeder were loaded into the blanket. The blanket consists of three layers, i.e., 1st one is on the plasma side, 2nd one is in the middle and 3rd one is the outer layer.

For an example of calculation results, the nuclear performance and transmutation efficiency are shown in Table 1. In this case, a plasma condition shown in Table 2 was employed. In the blanket, 8% reprocessed plutonium was loaded in the 2nd layer and 24% of the blanket was used for transmutation.

Table 1 Calculation Result

year	Power MW	TBR	Keff
0	6901	1.2478	0.7061
1	6466	1.2082	0.6946
2	6312	1.1731	0.6916
3	6042	1.1599	0.6791
4	5779	1.1415	0.6762
5	5746	1.1251	0.6651

Table 2 Plasma Parameter

Major radius (m)	6.2
Minor radius (m)	2.1
Plasma volume (m^3)	884
Plasma temperature (KeV)	19
Confinement time (s)	1.1
Electron density ($/m^3$)	4.80E+19
Fusion power (MW)	646
Neutron yeild (n/s)	2.20E+20
Neutron wall load (MW/m^2)	0.4
Plasma factor	0.7

Database Retrieval Systems for Nuclear and Astronomical Data

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Data retrieval and plot systems of nuclear data and astronomical data are constructed on a common platform. Web-based systems will soon be opened to the users of both fields of nuclear physics and astronomy.

The compilation of nuclear data has played an important role in contributing not only to the scientific research but also to the technological progresses. At the same time, this invokes demands for the utilization of nuclear data. There are some systems in the world that can search and plot the data from enormous database. However, no retrieval system can treat both experimental and evaluated nuclear data simultaneously. Based on the needs for comparisons of evaluated data with experimental data in a more convenient way, we have developed a web-based retrieval system [4].

On the other hand, we have launched a project of constructing the database of astronomical data that treat the observed properties of stars in the Galactic halo born in the early universe [1]. This project is motivated by the recent growing number of known extremely iron-poor stars and by our recent work on the origin of such stars [2] after the discovery of the most iron-poor object [3], which is more encouraged by the recent break of the record [4]. The purpose of the project is to identify the first generation objects as well as the comprehensive understanding of the history of our universe through the accumulation of observational data. Due to the difficulty of compiling the data from individual papers, the database of this kind has not yet been opened to the astronomical society.

Both systems are composed of the CGI form and the SQL database system. Users are easily accessible to the required data by setting the queries in the form and hitting a submit button. Selected data are plotted on the browser through the creation of files of the relevant data. Furthermore, the interchange between the database and the data at hand is done easily via the web-form. In this poster, the outline and some demonstrations of the retrieval systems for nuclear and astronomical database are presented with the captured images from the web.

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Development for the measurement system of the ^{189}Os(n,n'γ) cross section and Re/Os chronometer

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So far, the cross section of the keV neutron inelastic scattering reaction from the ground state of ^{189}Os to the 36 keV excited state has been measured by detecting neutrons inelastically scattered by ^{189}Os [1]. However, it is not easy to accurately measure the cross section with the method, because both the inelastically and elastically scattered neutrons from ^{189}Os are detected simultaneously by a neutron detector. Hence, we are now developing a new measurement system to detect the gamma ray from the (n,n'γ) reaction. In the poster session I will present a preliminary result of the test experiment of the new measurement system.

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Analysis of Continuum Spectra for Proton induced Reactions on ^{27}Al, ^{58}Ni, ^{90}Zr ^{197}Au and ^{209}Bi at 42 and 68 MeV—Direct Reaction Model Analysis.

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Theoretical analyses of the double differential cross sections for proton induced deuteron pickup reactions are described in this paper. The differential cross sections have been measured for various nuclei with mass number from 27 to 209 at incident energies of 42 and 68 MeV (for ^{197}Au only at 68 MeV, and for ^{209}Bi only at 42 MeV), using an approach based on the DWBA and an asymmetry Lorentzian function having energy-dependent spreading width. This method has been described briefly by Hirowatari et al. [1] and Syafarudin et al. [2] for the proton induced reactions and then applied to both for proton and neutron induced reactions by Sultana et al. [3,4]. The values of the calculated double differential cross sections have been compared with the experimental ones.

Figure 1 shows the double differential cross sections at 25^0 laboratory angle, with solid lines represent the theoretical cross sections, and circles the experimental ones. The experiments were carried out at the TIARA facility of JAERI. Potentials due to Koning et al. [5], which have symmetric term for both neutron and proton optical model potentials, have been used here in the DWUCK calculations. From the figure it is found that the shapes of continuum spectra are well reproduced by the theoretical calculation. It is interesting and also obviously desired to examine the applicability of the present analysis method in wide mass and incident energy regions.

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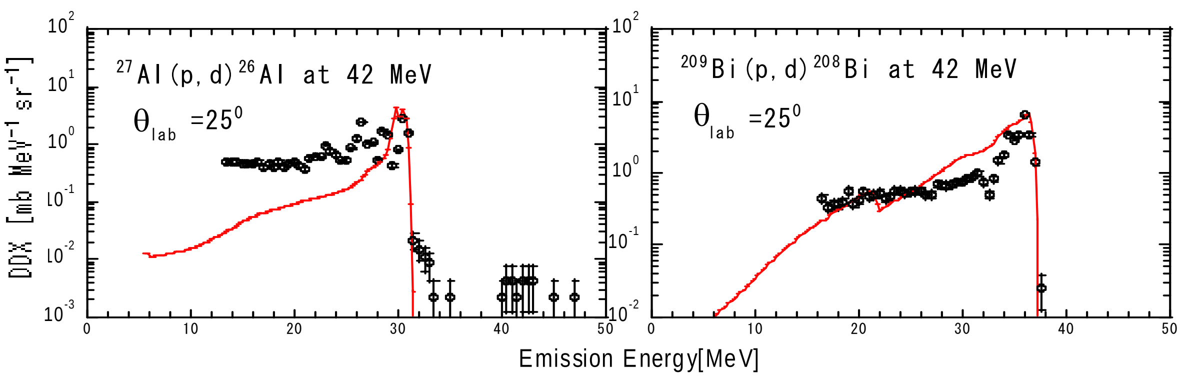

Fig 1. (p,d) DDX data of ^{27}Al (left) and ^{209}Bi (right) for 42 MeV proton at 25^0 laboratory angle.