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INDC INTERNATIONAL NUCLEAR DATA COMMITTEE

PREPARATION OF PROCESSED NUCLEAR DATA LIBRARIES FOR THERMAL, FAST AND FUSION RESEARCH AND POWER REACTOR APPLICATIONS

Texts of Papers presented at the IAEA Consultants' Meeting
organized by the International Atomic Energy Agency
and held at the IAEA Headquarters, Vienna, Austria,
8 to 10 December 1993

Compiled
by

S. Ganesan
IAEA Nuclear Data Section
Vienna, Austria

December 1994

IAEA NUCLEAR DATA SECTION, WAGRAMERSTRASSE 5, A-1400 VIENNA

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Abstract

The report contains 12 papers on nuclear data processing activities in Algeria, India, Indonesia, Italy, Japan, Republic of Korea, the Netherlands, Russia, Slovenia, United Kingdom, U.S.A., including ENDF formatted nuclear data libraries and computer code systems such as NJOY, AMPX, NSLINK, MCNP, multigroup data schemes such as WIMS, ABBN, and others. The role of the IAEA Nuclear Data Section in the establishment of nuclear data centers in developing countries is reviewed.

December 1994

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Introduction

An IAEA Consultants' Meeting on "Preparation of Processed Nuclear Data Libraries for Thermal, Fast and Fusion Research and Power Reactor Applications" was held during 8-10 December 1993 at the IAEA Headquarters, Vienna. The main purpose of the meeting was to review the current status of work in the subject of interfacing nuclear data banks to application calculations of thermal, fast and fusion research and power reactors, and, to assist the Agency in identifying appropriate IAEA activities related to nuclear data processing. This report contains the texts of the invited presentations delivered at this meeting. Since the meeting there have been many requests to make the texts of the presentations available in printed form. The texts are reproduced here, directly from the authors' manuscripts with little or no editing, in the order in which the presentations were made at the meeting. The summary report containing the agenda, conclusions and recommendations of the meeting has been separately published as document INDC(NDS)-299.

Srinivasan GANESAN
December 1994

THE ROLE OF IAEA IN FULFILLING NUCLEAR DATA NEEDS

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Abstract

The role of the Nuclear Data Section of IAEA has been unique and significant in India's nuclear data related work. Experience at Indira Gandhi Centre for Atomic Research Centre (IGCAR) has shown that the services of NDS-IAEA have been vital in building a capability at our centre, towards preparation of multigroup constants for application in fast reactor physics calculations, starting from the basic evaluated nuclear data libraries. However, bottlenecks exist in realising completeness in the benefits due to non-availability of certain data or codes to our country. Problems also exist due to lack of manpower and computer resources. In this presentation, our experiences with nuclear data processing, the problems faced etc. are outlined. Some suggestions to NDS-IAEA are also given towards removing some of the problems.

THE ROLE OF IAEA IN FULFILLING NUCLEAR DATA NEEDS

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I. Introduction

India has a programme on nuclear data related work, which is limited to meeting the needs of the design and operation of thermal and fast reactor applications. Basic nuclear data measurements and creation of evaluated nuclear data files are not part of our programme. We consequently rely upon the data made available from International Nuclear Data Centres like IAEA. The Nuclear Data Section of IAEA, in particular, has been playing a unique and significant role in meeting our nuclear data requirements. In fact, planning of our nuclear data programme, including allocation of manpower and other resources, takes into account what is available through IAEA. Our experience at the Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, India, reported below, clearly shows the important role played by IAEA in the development of a code system for the preparation of multigroup constants. IAEA has been effectively distributing evaluated data files, their updates, the available processing codes and so on along with the well formatted documents. In recent years IAEA has been making available preprocessed or even multigrouped data. The Code Verification Project and the WIMS Library Update Project are good examples of IAEA's initiative

towards ensuring correctness and consistency in the codes and the data used by the nuclear data user community. This forum gives us an opportunity to express our appreciation of the commendable services NDS-IAEA has been rendering to its member countries. As the nuclear data teams in the developing countries like ours continue to include the NDS-IAEA contribution in their development plans, its services in the future would remain essential.

However, some bottlenecks remain to be cleared such as non-availability of specific data or some processing codes. As far as our Centre is concerned, our programme on nuclear data for reactor physics purposes, includes updating our information with newer data and better processing, if feasible. It is felt the latest version of NJOY could help us appreciably in our efforts to create a new multigroup library for core physics applications and a neutron - gamma coupled library for shielding applications from the ENDF/B-VI data file supplied by IAEA. At present, we do not have the recent version of this code. IAEA processing codes (Dr. Cullen's LRSG system of codes) do not deal with coupled neutron-gamma library. They also are inadequate in treating the unresolved resonance region at elevated temperatures. They do not have the capability to extract $\bar{\mu}_L(E)$ from angular distribution data of ENDF/B-VI. Though the IAEA preprocessing codes by themselves are excellent tools for obtaining linearised point data, we at IGCAR, have had to supplement them with our own codes REX1/1/, REX2/2/ and REX3/3/ for obtaining group cross sections and self-shielding factors. NJOY on the other hand, has

all the advantages of IAEA codes in giving reconstructed point data, in addition to giving multigroup constants conforming to various definitions and formats used in several popular neutronic codes including ANISN and WIMS. NJOY's capability to process thermal scattering law data, to calculate displacement cross sections and kerma factors, to process gamma interaction cross sections etc. are also attractive for our immediate applications. In the absence of NJOY, though we have been able to prepare neutron multigroup cross sections from ENDF/B-VI using Cullen's codes, complemented by our own, we are unable to use ENDF/B-VI for creating neutron - gamma coupled library, displacement cross sections etc. The details of our nuclear data requirements, our experiences, problems faced etc. will be discussed in brief below.

II.Requirements of nuclear data at IGCAR

We require the following nuclear data for our research activities:

(1). Neutron multigroup cross sections for fast and thermal reactor core physics neutronics calculations.

(2). Neutron - Gamma coupled multigroup cross sections for fast reactor shield calculations.

(3). Neutron activation multigroup cross sections for activation studies.

(4). Neutron multigroup displacement cross sections for calculating the damage in the structural materials in a fast reactor environment.

(5). Multigroup cross sections for the important fission

products and higher actinides.

III. Experiences and problems of ENDF/B nuclear data processing

It has been our interest to create our own multigroup (non-adjusted) cross section sets in format suitable for neutronics codes for performing LMFBR core calculations from the latest ENDF/B type files to study the impact of recent revisions in the basic data. The Nuclear Data Section of IGCAR has developed the capability to generate the neutron multigroup cross sections from the differential neutron cross section data in ENDF/B format for the fast reactor core physics neutronics calculations. For this, we have used various IAEA preprocessing codes (LINEAR, RECENT, and SIGMA1) , our own multigrouping codes (REX1, REX2 and REX3) and some interfacing programs (LCAT/4/ and TOXIC/5/). A flowchart for the preparation of multigroup cross section set at IGCAR from a basic evaluated nuclear data file is given in Fig.1. Using the above code system, we have created earlier a 25 group neutron multigroup cross section set (IGCENDF4) /6/in format of French adjusted Cadarache Version 2 (1969) set/7/, available at IGCAR. The performance of IGCENDF4 set in predicting the reactor integral parameters was found to be similar to the Cadarache Version 2 set. Its creation gave us confidence and motivation for generating multigroup cross section sets from other more recent files.

We have recently completed creating a 25 group neutron cross section set (IGCJENDL)/8/ for nuclides of interest to LMFBR from the Japanese Evaluated Nuclear Data Library - Version 2 (JENDL-2) (1984) in the format of Cadarache Version 2 set. Since we do not

have a programme of critical experiments, the only way to check the predictional capability of the generated multigroup cross section set is through the analyses of the critical assemblies whose informations are published by the Cross Section Evaluation Working Group. The integral validation of IGCJENDL set /9/ was done by analysing nine fast critical assemblies having a wide range of core size, neutron energy spectrum, fuel composition etc. We had calculated effective multiplication factors, central reaction rate ratios and the reactivity coefficients for these assemblies and good agreement with the reported values was obtained.

So far the adjusted Cadarache Version 2 set has been used for the small LMFBR core physics calculations, at IGCAR. But, this set has been found inadequate for the core physics studies of the planned 500 MWe Prototype Fast Breeder Reactor (PFBR), mainly due to its overprediction (about 3%) of k-eff's of two theoretical LMFBR benchmarks of 1200 and 500 MWe capacity respectively, compared to the reported predictions by the adjusted French cross section set, CARNAVAL-IV (1977) or the Russian set BNAB-78, for the same assemblies. Analyses of the above theoretical benchmarks with IGCJENDL/10/ showed that this set could predict the integral parameters, k-eff in particular, close to that predicted by the CARNAVAL-IV or the BNAB-78 set, and hence could be used for large LMFBR analysis. Interestingly, we had found, after a detailed study, that the replacement of Pu-241 data in Cadarache Version 2 set by Pu-241 data taken from IGCJENDL set removes the 3% overprediction of k-eff in the above

benchmarks. We also propose this set, called ' Modified Cadarache set', for the design analysis of PFBR.

In most of the fast reactor core physics calculations, we make use of the 1969 version of the adjusted Cadarache Version 2 set (modified as above when necessary). But, this set contains the data for only 38 nuclides. So, it was planned to expand this set by adding multigroup data from ENDF/B-VI library for nuclides which are absent in the above set and are often required for reactor calculations. Since ENDF/B-VI does not explicitly give elastic scattering angle cosine data in the laboratory system, required to calculate transport cross sections, a program AMUL-4 /11/ was written to calculate this quantity from the Secondary Neutron Angular Distribution (SNAD) data. By employing the AMUL-4 code in our code system (see Fig. 1), the ENDF/B-VI data for several important nuclides which are absent in the Cadarache Version 2 set were processed and added. The fission products from ENDF/B-VI could also be processed this way.

For fast reactor shield calculations, a neutron-gamma coupled multigroup (100 n, 21 gamma) cross section library called DLC-37, procured from RSIC, is extensively being used. This library was generated from ENDF/B-IV and is meant for fusion reactor applications as reported. So, we have taken up the task of creating a coupled library from the latest evaluated data files. Even though the ENDF/B-VI library is available here, a code system to process this library for the above purpose is not available, like the new NJOY system (NJOY91). For the same

reason , we are, at present, unable to process the improved data in ENDF/B-VI related to displacement cross sections.

IV. Suggestions to Nuclear Data Section, IAEA

In developing countries, like India, the needs of nuclear data for the nuclear energy programme, are met with a considerable dependence on International Centres, and NDS-IAEA in particular. These countries would find it difficult to expand their teams and resources to keep pace with the advances made by the highly developed contributing countries. It is felt that NDS-IAEA's endeavour in distributing the latest cross section information in the most sophisticated formalisms and formats, would be made much more useful if IAEA itself takes the responsibility of distributing the necessary processing codes, that are fully compatible with the data it distributes. This may probably be achieved by forming a team under NDS-IAEA, with personnel selected from both advanced and developing countries, for preparing such codes and their updates whenever needed. Such a team, after the teething problems are over, could serve all the IAEA member countries for their nuclear data processing needs, of course, limited to the data supplied by IAEA. The areas requiring immediate attention are:

(1) Supplementing IAEA preprocessing codes systems with correct treatment for the unresolved resonance region at higher temperatures.

(2) Preparation of neutron - gamma coupled library using ENDF6 formatted libraries.

(3) Processing thermal scattering law data from ENDF6

(4) Calculation of radiation damage related parameters

(5) Preparation of interfaces to give the outputs in conformity with definitions and formats of most widely used neutronic application codes.

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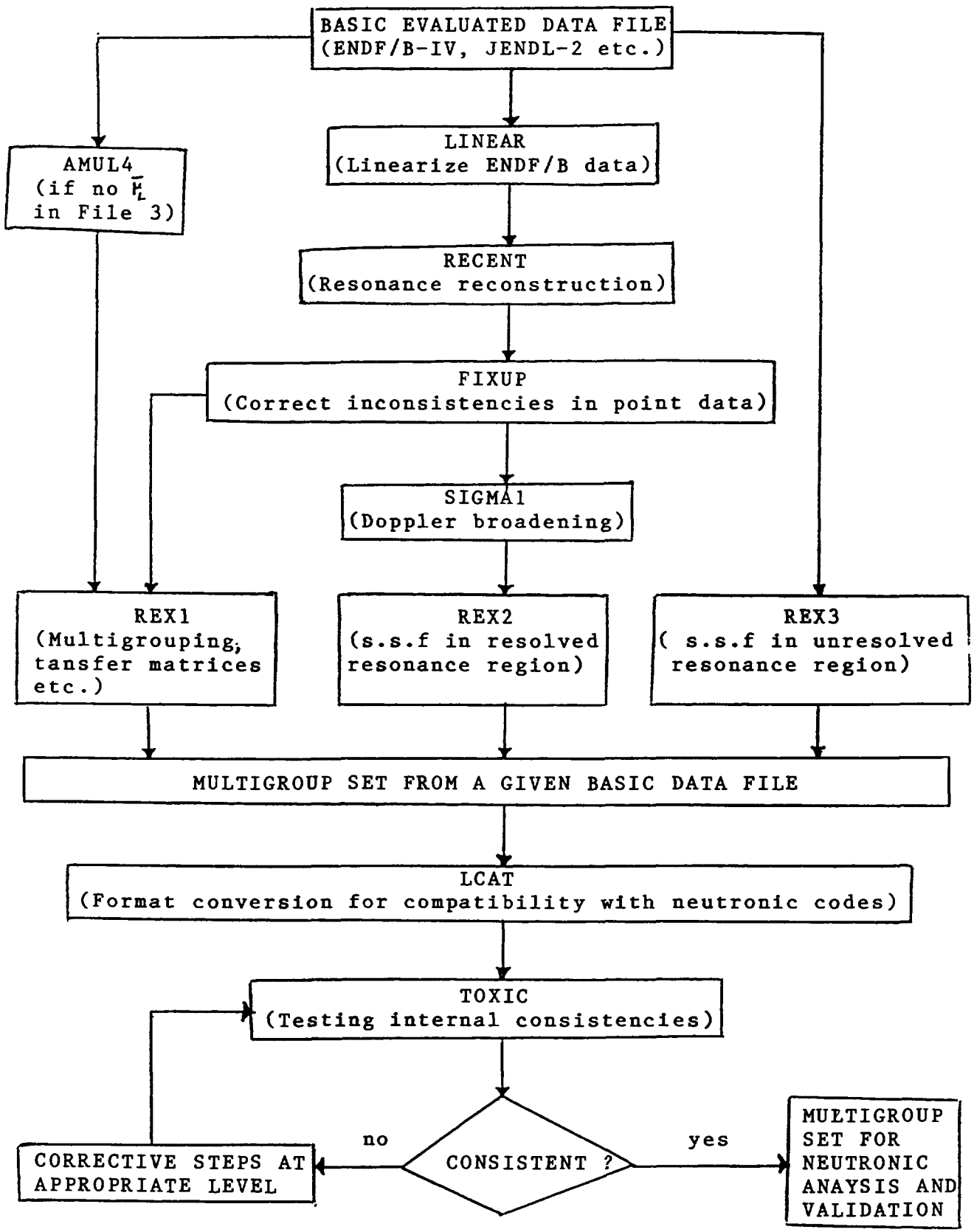

Fig.1 Flowchart for the preparation of multigroup neutron cross section set at IGCAR

ESTABLISHMENT OF COMPUTER CODE SYSTEM FOR NUCLEAR REACTOR DESIGN - ANALYSIS

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ABSTRACT

Establishment of computer code system for nuclear reactor design analysis is given in this paper. This establishment is an effort to provide the capability in running various codes from nuclear data to reactor design and promote the capability for nuclear reactor design analysis particularly from neutronics and safety points. This establishment is also an effort to enhance the coordination of nuclear codes application and development existing in various research centre in Indonesia. A very prospective results have been obtained with the help of IAEA technical assisstance.

1. INTRODUCTION

At present different kinds of reactor analysis capability have been developed at different BATAN centres for the analysis of the research reactors at Serpong, Bandung and Yogyakarta. This mainly involves using certain whole core calculation codes such as TRIGA - C and IAFUEL with inbuilt cross sections. Shielding and kinetics analysis are also undertaken. Further, individual efforts at running other codes such as NJOY, WIMS, CITATION etc are being made at different centres.

However, a comprehensive and unified nuclear design-analysis computer code system with all the modules properly interfaced and compatible with each other is lacking. In view of the commitment towards nuclear power

program in Indonesia and taking into account the plans to set up a nuclear power plant in the coming decade it is important to start work towards development of such a comprehensive nuclear analysis code system which can start from basic nuclear cross section processing and go right down to whole core and shielding calculations.

Such a tested and validated computer code system will provide quality assurance in the calculation of safety and other parameters for chosen reactor power plants.

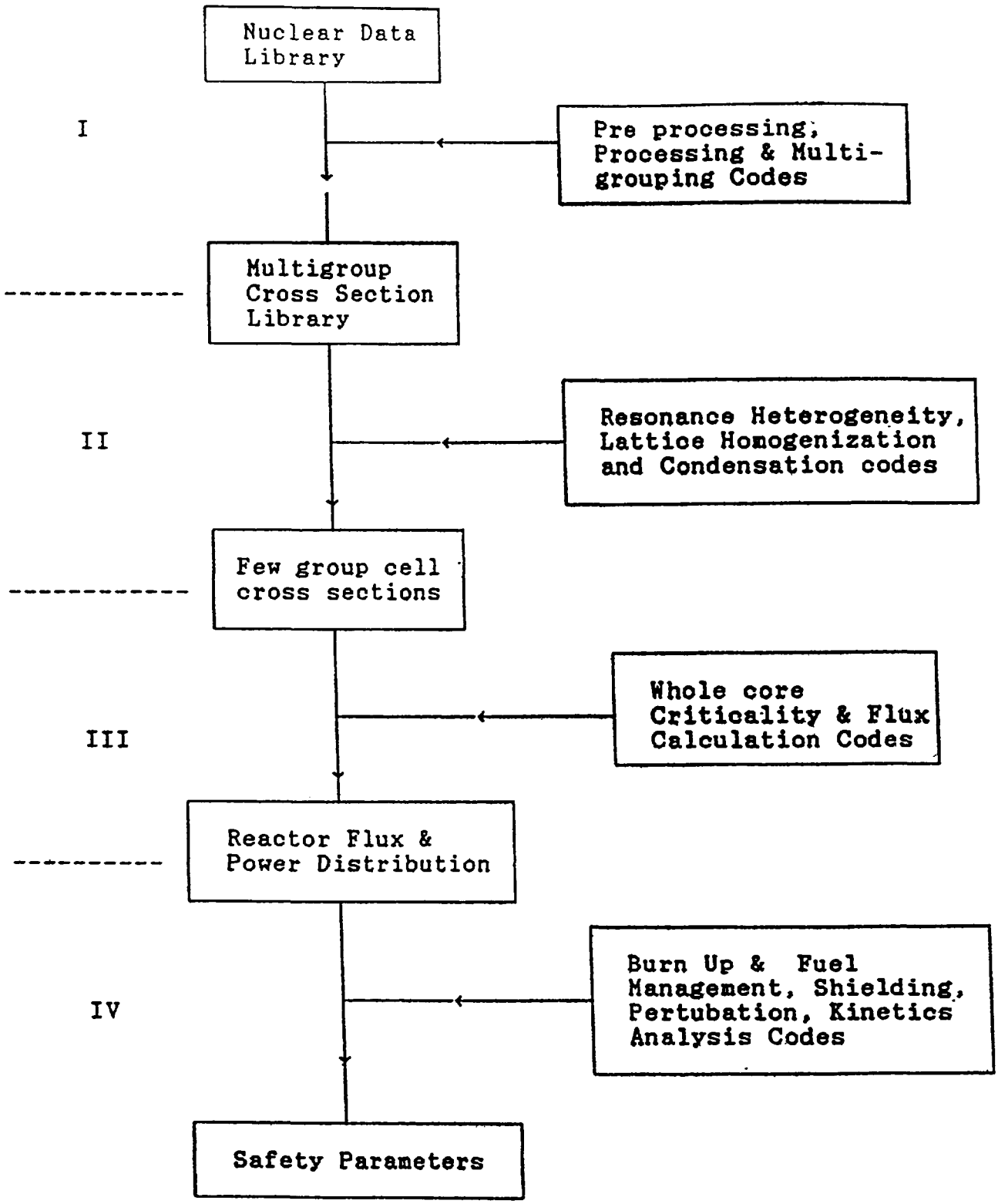
Nuclear Data Library
I
Pre processing, Processing & Multi-grouping Codes
Multigroup Cross Section Library
II
Resonance Heterogeneity, Lattice Homogenization and Condensation codes
Few group cell cross sections
III
Whole core Criticality & Flux Calculation Codes
Reactor Flux & Power Distribution
IV
Burn Up & Fuel Management, Shielding, Pertubation, Kinetics Analysis Codes
Safety Parameters

Figure-1

2. NUCLEAR ANALYSIS FLOW CHART

Fig. 1 presents a flow chart of the scheme of calculations for nuclear analysis. At the most basic level (Level I) is the nuclear data library containing neutron and photon cross sections, decay data etc for nuclides of interest to nuclear reactors. Only few countries produce these libraries through international efforts. Some examples are ENDF/B of USA, JENDL of Japan, JEF (Joint European File) etc.

In order to process the basic nuclear data library into homogeneous mixture multigroup cross sections for each nuclide as a function of temperature and potential (or background) cross section a set of pre-processing, processing and multi grouping routines are required. One such nuclear data processing computer code system is the NJOY code of USA for processing ENDF/B library into multi group cross sections.The Code can also produce coupled neutron gamma cross sections. The ENJOY-89 or later version is necessary for processing the latest ENDF/B-VI file. Similar to the NJOY code is the LINEAR/RECENT/SIGMA1/ GROUPIE code system available from IAEA.

At the second level (Level II) is the code system for correcting by equivalence principles or other methods the homogeneous mixture nuclide multigroup cross section for the fact that the actual mixture in the reactor occurs in a heterogeneous structure. The multigroup cross sections corrected for resonance heterogeneity and dependent on the reactor region composition and geometry are thus obtained for each nuclide. The mixture and nuclide (if needed) cross sections are further collapsed to few groups at this second level since the whole core and other computation for thermal reactors can be efficiently and accurately made in few groups. Further, it is not possible to take into account the detailed location of each pin in a fuel assembly in a whole core calculation. Thus at this second level of computation the spatial heterogeneity is properly homogenized by detailed flux weighting and "cell homogenized" mixture cross sections are obtained. Burn up

effects in cell homogenization also are taken into account at this second level of computation. The computation at this second level generally use the collision probability (CP) or, less frequently the DSN method. A well known lattice code system to handle the second level of computations is, for example, the WIMS code of UK, which has an in-built 69 group library obtained by first level computations from the UK Nuclear Data Library. Similiar code of US is for example LEOPARD which has in built multi group library derived earlier by first level computation from ENDF/B.

For the third and fourth level of computation involving whole core criticality, burn up, pertubation, shielding and kinetics analysis a large number of standard computer codes are available. Generally, the whole core calculations are made using two or three dimensional diffusion theory. Thermal-hydraulic coupling with neutronic analysis may be required in some reactors like BWRs. One and two dimensional calculations can be done in greater spatial detail and with more energy groups as compared to three dimensional calculations, whereas three dimensional calculations are useful for partial control rod insertion and other axial heterogeneity caloulations. Shielding calculations generally need one or two dimensional transport theory (DSN method) computer codes. Subsidiary burn up and pertubation theory codes may be required if the routines are not already incorporated in the whole core calculation code.

3. PROPOSED COMPUTER CODE SYSTEM

For the first level of computation it is suggested that NJOY latest version along with ENDF/B-VI be adopted. It is appreciated that this is a complex code system and considerable time and effort will be needed for its adaption to the available computing systems. However, as it is a comprehensive code system with facility for production of coupled neutron-gamma multigroup libraries it is worth the effort. Delay in the progress of this work will not

affect the flow of the calculations at the subsequent levels since it is proposed to retain WIMS 69 group library as the basis for the second . and subsequent level computations. The NJOY and ENDF/B utilization at the first level of computation shall serve only as a means of including materials which are unavailable in WIMS and for the purpose of updating the WIMS library as and when required.

Coming to the second level of computation the well known computer code WIMS with its inbuilt library is already available in BATAN and its research units. As already stated it is proposed that this code be continued in use as the main code for lattice cell caloulations. The PC version WIMSD4 is available and it may be necessary to add some more routines for multi cell or other calculations needed for different purposes.

For the whole core calculations it is recommended that CITATION be adapted as standard. This code is already available at the BATAN centre at Serpong and necessary modifications to accept cross sections in WIMS format may be necessary. As back up to CITATION it is recommended that 2DB/3DB codes be used for the main purpose of intercomparison and validation. These codes are already available in both versions PC and mainframe and working at BATAN Centres in Jakarta and Serpong. These codes can all perform core burn up calculations.

For shielding calculations one dimensional and two dimensional transport theory (DSN) codes are required. ANISN is already working in BATAN centres at Jakarta and Serpong and available for one dimensional neutron and gamma transport problems. It is recommended that DOT DSN computer code be obtained and adapted for two dimensional transport calculations.

4. PROPOSED STRATEGY

Codes which are not already available in BATAN will be obtained from sources such as IAEA, NEA Data Bank, Argonne Code Centre etc. Initally, all the codes will be worked on at the Nuclear Technology Assessment Centre, (BATAN) using powerful PCS (486) with extended memories. Necessary interfacing between modules so that the output from each level of computation is compatible with the input at the next level will have to be made and calls for sufficient programming expertise. It is also foreseen that the codes may have to be improved so that user-interactive input and output could be supported. Once the code system is established and properly validated it oould be distributed to the different BATAN centres for utilization and also used for the proposed power plant studies. It is considered that for each code one scientist should be available for the detailed study, modification and commissioning. Thus 6 scientists working with a commitment of about 12 man years will be needed.

5. PROGRESS TO BE ACHIEVED

A number of initiatives haven been taken to install various codes for this establishments. Table - 1 show the list of the codes available with their status. Effects have also been made to increase the number of staffs enganged and table - 2 displays computer Codes establishment and utilization. Technical assistances have also been requested from IAEA, starting with the installations of ENDF/B VI, a long with the processing codes such as LINEAR/RECENT/SIGMA1/GROUPIE, where Dr. Trukov and Dr. Ganesan have successfully completed their missions as experts assigned to the IAEA assistance. We have further requested IAEA technical assistance to oontinue the programme under a project title computer codes. In this project three experts have been proposed, one is Dr. Lee (IGCAR) (one of four authors of this article) who has

completed the mission successfully and others Dr. Lawson and Mr. Gopalakrisnan (IGCAR) to be expected to come in April 1993.

6. CONCLUSION

With the development of the nuclear power programme in Indonesia the need for a quality assured modular computer code system for nuclear analysis is felt. The paper has outlined a strategy which if followed could meet this need.

ACKNOWLEDGEMENT

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TABLE -1 CURRENT STATUS OF THE ANALYSIS CODES

NO	NAME OF CODES	OBJECTIVES	STATUS					
			0	I	II	III	IV	V
1	ENDF/B Nuclear Data Files	- Nuclear Data Library			X	X	X	
2	LINEAR/RECENT/ SIGMAI/GROUPIE Processing Codes	- Multigroup Cross Sections			X	X	X	
3	NJOY	- Multigroup Cross Section (Self Shielding)		X				
4	WIMS-D/IV (PC & Mainframe)	- DSN transport and Collision Probability Calculation - Macroscopic Cross Section for lattice - Burn up			X	X	X	X
5	IAFUEL	- 2 Dimensional diffusion theory - flux and criticality calculations - incore fuel Management			X	X	X	X
6	TRIGA P. (PC)	- One Dimensional diffusion theory - flux & criticality calculation - Incore Fuel Management			X	X	X	

TABLE -1 CURRENT STATUS OF THE ANALYSIS CODES

NO	NAME OF CODES	OBJECTIVES	STATUS					
			0	I	II	III	IV	V
7	CITATION (PC & Mainframe)	- 1/2/3 Dimensional Diffusion theory - flux & criticality calculation - Burn up			X	X		
8	2DB (PC & Mainframe)	- Two dimensional Diffusion theory - flux & criticality calculation - Burn up			X	X	X	
9	3DB (PC & Mainframe)	- Three Dimensional Diffusion theory - flux & criticality calculation			X	X		
10	ANISN (PC & Mainframe)	- One Dimensional DSN transport method for Neutron & gamma - Flux, criticality & source calculations - macroscopics cross section for cell - shielding calculation			X	X	X	

TABLE -1 CURRENT STATUS OF THE ANALYSIS CODES

NO	NAME OF CODES	OBJECTIVES	STATUS					
			0	I	II	III	IV	V
11	DOT	- Two dimensional DSN transport method for neutron & gamma - flux, criticality & source calculations - shielding calculation	X					
12	RELAP	- Safety Analysis	X					
13	EUREKA	- Safety Analysis			X	X	X	

NOTE :

0 = to be procured

I = being installed

II = has been installed

III = sample run

IV = usage experience

V = application for design & analysis

Comparison of WIMS Results Using Libraries Based on New Evaluated Data Files

Primerjava rezultatov programa WIMS z uporabo različnih knjižnic presekov na osnovi novejših evaluiranih knjižnic podatkov

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ABSTRACT - A number of selected benchmark experiments have been modelled with the WIMS-D/4 lattice code. Calculations were performed using multigroup libraries generated from a number of newly released evaluated data files. Data processing was done with the NJOY91.38 code. Since the data processing methods were the same in all cases, the results may serve to determine the impact on integral parameters due to differences in the basic data. The calculated integral parameters were also compared to the measured values. Observed differences were small, which means that there are no essential differences between the evaluated data libraries. The results of the analysis cannot serve to discriminate in terms of quality of the data between the evaluated data libraries considered. For the test cases considered the results with the new, unadjusted libraries are at least as good as those obtained with the old, adjusted WIMS library which is supplied with the code.

POVZETEK - Nekaj izbranih testnih kritičnih eksperimentov smo modelirali s programom WIMS-D/4. Izračuni so bili narejeni na osnovi grupnih konstant dobljenih iz več različnih novejših knjižnic evaluiranih nevtronskih podatkov. Za procesiranje podatkov je bil uporabljen program NJOY91.38. Ker je bila priprava podatkov izvedena na enak način, so rezultati primerni za določitev vpliva razlik med knjižnicami na integralne parametre. Izračunani integralni parametri so bili primerjani tudi z izmerjenimi vrednostmi. Ugotovljeno je bilo, da so odstopanja majhna, zato na osnovi teh rezultatov ni mogoče ugotoviti bistvenih razlik v natančnosti podatkov v različnih evaluiranih knjižnicah Za obravnavane testne primere smo ugotovili, da so rezultati z novimi, nepopravljenimi WIMS knjižnicami najmanj tako dobri kot tisti dobljeni s staro knjižnico, ki je bila popravljena na osnovi integralnih meritev.

1 Introduction

The WIMS package [1] originating from the Winfrith laboratory is widely used for reactor calculations of a variety of thermal reactors. It consists of a lattice transport code and the associated library. The lattice code exists in a version, which is available from the NEA Data Bank at Saclay as WIMS/D-4, and versions LWR-WIMS and WIMS-E which are distributed to the users through the ANSWERS service [2] on commercial terms. They all use basically the same multigroup data library [3], although the commercial versions allow some format extensions and their library is reported to include some further data adjustments [4] which improve the performance of WIMS.

The libraries that are supplied with different versions of the WIMS code are based on basic evaluated data files originating from the sixties. Considering that the source data base is very old and obsolete, the performance of WIMS has been shown to be remarkably good because of several adjustment to the multigroup data and particularly to the resonance integrals. The adjustments were performed through comparisons of the calculated integral parameters with measurements for a wide range of "benchmark" experiments.

With the welcomed release of several new evaluated data files [5,6] through the International Atomic Energy Agency (IAEA), such as JENDL-3.1, BROND-2, ENDF/B-VI, JEF-2 and CENDL-2, it was felt that the performance of WIMS for lattice calculations could be improved by updating its library from one of the new evaluated data files. With this objective in mind, the Nuclear Data Section of the IAEA has initiated the WIMS Library Update Project (WLUP) through international cooperation [7], which commenced in 1991. The work completed so far includes optimization of WIMS inputs to model some selected benchmark experiments as accurately as possible [8] and intercomparison of data entered into the WIMS library using different code systems [9]. Related to this work is also the upgrading of the WIMSR module of the NJOY code for nuclear data processing [10], so that it could be used reliably to prepare group constants for the WIMS library. The principal advantage of NJOY [11] is its versatility and its comprehensive capability to process data in ENDF-6 format, which is used for the ENDF/B-VI, JEF-2 and CENDL-2 evaluated data libraries.

A parametric study of the effect of different NJOY input options on integral results calculated by WIMS has been performed [12]. The ENDF/B-IV [13] library was used in the analysis. Using the updated library based on ENDF/B-IV data, WIMS calculations were done for a selection of benchmark lattices [14], considered in Stage-1 of WLUP. The results were compared to the highly accurate published results [15] for the same lattices (obtained mainly by Monte Carlo codes), which were used as numerical benchmarks. In this way a consistent set of NJOY91.38 input instructions to process evaluated nuclear data files and generate an updated WIMS library have been obtained.

Once the optimized data processing procedures for generating the WIMS library from ENDF/B-IV data with NJOY91.38 have been established, they were applied to other libraries (JENDL-3.1, ENDF/B-VI, JEF-1, JEF-2 and CENDL-2) that can be processed by the same code. The results of such analysis are presented in this work.

Table 1 Summary of WIMS results based on ENDF/B-IV data for selected benchmark lattices and comparison with reference results

LATTICE	k_{eff}	ρ^{28}	δ^{25}	δ^{28}	C^*
TRX-1	0 98760(32)	1 382(43)	0.0994(.60)	0.0966(63)	0 806(26)
ENDF/B-IV	0 98706(- 06)	1.380(-.14)	0.1002(+.80)	0.0928(-2.8)	0 804(- 25)
TRX-2	0 99350(.31)	0 863(.58)	0.0609(.49)	0.0676(.44)	0.647(31)
ENDF/B-IV	0 98916(- 44)	0 868(+ 58)	0.0615(+1.0)	0.0660(-2.4)	0.648(+ 25)
BAPL-1	0 99140(30)	1 433(2.0)	0.0835(1.6)	0 0735(96)	0.817(1 3)
ENDF/B-IV	0 99321(+ 18)	1 429(- 28)	0.0843(+.98)	0.0717(-2 5)	0.814(- 31)
BAPL-2	0 99320(09)	1 188(1.3)	0 0678(1.8)	0 0631(79)	0.742(81)
ENDF/B-IV	0 99325(+ 01)	1 191(+ 25)	0.0688(+1.4)	0 0617(-2 3)	0 743(+.08)
BAPL-3	0 99395(21)	0 936(1 7)	0.0522(38)	0.0516(.78)	0 664(1 1)
ENDF/B-IV	0 99373(- 02)	0 938(+ 21)	0 0529(+1.3)	0.0506(-2 0)	0 664(0 00)

NOTE For each lattice the reference solution and the % uncertainty is given in the first row The results using the updated ENDF/B-IV based library and the % difference from reference are given in the second row.

2 Data processing verification

The data processing details and the sensitivity of the results to the selection of the input parameters have been reported elsewhere [10,12,16,9], so they will not be repeated here. Integral measurements on thermal reactor lattices TRX-1, TRX-2, BAPL-UO_2-1, BAPL-UO_2-2 and BAPL-UO_2-3, which serve as standard benchmarks for testing nuclear data [14], have been selected for the analysis. For convenience, the comparison table of the calculated integral parameters [10] based on ENDF/B-IV data and the reference results is reproduced in Table 1. The measured integral parameters are listed below:

k_{eff} the effective multiplication factor,

ρ^{28} the ratio of the epithermal to thermal capture reaction rates in ^{238}U,

δ^{25} the ratio of the epithermal to thermal fission reaction rates in ^{235}U,

δ^{28} the ratio of the total fission reaction rates in ^{238}U and ^{235}U,

C^* the ratio of the capture reaction rates in ^{238}U and fission reaction rates in ^{235}U (i.e. the conversion ratio), measured in the TRX lattices only.

The thermal cutoff energy for the parameters defined above is 0.625 eV where applicable.

The reference results [15] were obtained by averageing the results of 10 different contributors, 6 of them using sophisticated Monte Carlo methods. Very good agreement between different contributions was observed, so we believe the reference results to be highly reliable

From the results in Table 1 it can be seen that WIMS predicts ρ^{28} and C^* quite well. The k_{eff} is generally predicted well, but there might be a trend to underpredict k_{eff} in lattices with metal fuel. Parameter δ^{25} lies at the upper end of the uncertainty interval of the reference results. This could be due to the differences

in the treatment of the unresolved resonance parameters, which are used to describe the ^{235}U cross sections over most of the resonance range There is a very clear underprediction of parameter δ^{28} of about 2 4 %. This parameter was found to be sensitive not only to the fission spectrum, but also to the cross section averageing spectrum. This is an indication that the WIMS 69-group structure is inadequate in the fast resonance range. This problem cannot be overcome within the present scope of analysis.

The conclusions reached with ENDF/B-IV data must be considered when analysing the results obtained with other evaluated libraries, discussed in the next section.

3 Comparison of integral results using different recently released evaluated libraries

Evaluated data libraries that have been included in the analysis are the following:

- CENDL-2 the Chinese evaluated nuclear data library,
- ENDF/B-VI the evaluated nuclear data files from USA,
- JEF-2 the revised joint European files,
- JEF-1 the original joint European files,
- JENDL-3.1 the Japanese evaluted nuclear data library.

The Russian BROND-2 library was also considered, but there arose problems with data processing. In the library some uncommon data formatting options are used to describe angular distributions which have not yet been incorporated into NJOY91.38, so a complete and consistent data set could not be produced.

The JENDL-3.1 and CENDL-2 libraries have no thermal scattering law data, therefore the ENDF/B-III file was used. The JEF-2 library also lacks thermal scattering law data, so the thermal scattering matrix for hydrogen bound in water processed from JEF-1 was used instead.

The results of all the calculations are summarized in Table 2. For each of the benchmark lattices the measured values of the integral parameters and the associated measurement % uncertainty are given. Then follow the calculated results using libraries generated from different evaluated data files. The calculated values and the % differences from the measurement are given. The results using the old WIMS library [3] are also presented. The result show that the maximum difference in k_{eff} never exceeds 0.58 %. With most of the new libraries the parameter ρ^{28} is predicted well within the uncertainty interval of the measurements for all lattices except for BAPL-2, where a systematic overprediction of 3 to 4 % is observed. This could be an indication of an error in the measurement. The agreement of the calculated values of δ^{25} is also very good. There is a large experimental uncertainty in the measurements of δ^{28}. The calculated values are generally within the uncertainty interval of the measurements, except for the well thermalized BAPL lattices, where some underprediction of δ^{28} is observed. Part of this underprediction can be attributed

Table 2: Summary of WIMS results using different evaluated data libraries for selected benchmark lattices and comparison with measured values

LATTICE	k_{eff}	ρ^{28}	δ^{25}	δ^{26}	C^*
TRX-1	1.00000(30)	1.320(1.6)	0.0987(1.0)	0.0946(4.3)	0.797(1.0)
CENDL-2	0.99997(- 03)	1.348(+2.1)	0.0980(- 72)	0.0960(+1.4)	0.789(-1.1)
ENDF/B-VI	0.99972(- 53)	1.345(+1.9)	0.0985(- 18)	0.0969(+2.5)	0.794(-.36)
JEF-1	0.99960(- 04)	1.336(+1.2)	0.0988(+ 12)	0.0978(+3.4)	0.793(- 53)
JEF-2	0.99982(- 32)	1.339(+1.4)	0.0985(- 23)	0.0968(+1.3)	0.798(+ 09)
JENDL-3.1	0.99957(- 55)	1.344(+1.8)	0.0970(-1.8)	0.0978(+3.3)	0.793(- 45)
WIMS-D/4	1.00227(+ 23)	1.279(-3.1)	0.0990(+ 30)	0.0965(+2.0)	0.780(-2.1)
TRX-2	1.00000(10)	0.837(1.9)	0.0614(1.3)	0.0593(5.1)	0.647(93)
CENDL-2	0.99911(- 09)	0.850(+1.6)	0.0603(-1.8)	0.0582(-1.6)	0.638(-1.4)
ENDF/B-VI	0.99429(- 58)	0.847(+1.2)	0.0606(-1.4)	0.0589(-.65)	0.643(- 68)
JEF-1	0.99844(- 16)	0.842(+ 60)	0.0608(- 99)	0.0599(+.61)	0.642(- 74)
JEF-2	0.99561(- 42)	0.844(+ 84)	0.0606(-1.3)	0.0584(-1.3)	0.646(- 12)
JENDL-3.1	0.99452(- 55)	0.847(+1.2)	0.0596(-2.9)	0.0597(+.55)	0.642(-.77)
WIMS-D/4	0.99654(- 35)	0.808(-3.5)	0.0610(- 64)	0.0595(+.30)	0.636(-1.7)
BAPL-1	1.00000(10)	1.390(72)	0.0840(2.4)	0.0780(5.1)	0.000
CENDL-2	1.00549(+ 54)	1.399(+ 63)	0.0826(-1.8)	0.0740(-5.2)	0.799
ENDF/B-VI	1.00052(+.05)	1.392(+.14)	0.0828(-1.4)	0.0747(-4.3)	0.804
JEF-1	1.00567(+ 56)	1.385(-.36)	0.0832(-.93)	0.0758(-2.8)	0.803
JEF-2	1.00286(+.28)	1.385(- 36)	0.0828(-1.4)	0.0741(-5.1)	0.808
JENDL-3.1	1.00137(+ 14)	1.392(- 14)	0.0816(-2.9)	0.0757(-3.0)	0.803
WIMS-D/4	1.00292(+ 29)	1.358(-2.3)	0.0840(+.06)	0.0756(-3.2)	0.800
BAPL-2	1.00000(10)	1.120(89)	0.0680(1.6)	0.0700(5.7)	0.000
CENDL-2	1.00424(+ 42)	1.167(+4.2)	0.0674(- 96)	0.0636(-9.1)	0.730
ENDF/B-VI	0.99936(- 06)	1.161(+3.7)	0.0676(- 56)	0.0642(-8.3)	0.734
JEF-1	1.00435(+.43)	1.156(+3.2)	0.0679(-.10)	0.0653(-6.7)	0.734
JEF-2	1.00157(+ 16)	1.155(+3.1)	0.0676(-.54)	0.0638(-8.8)	0.737
JENDL-3.1	1.00039(+ 04)	1.161(+3.7)	0.0665(-2.1)	0.0652(-6.9)	0.733
WIMS-D/4	1.00049(+.05)	1.133(+1.2)	0.0687(+1.0)	0.0652(-6.8)	0.732
BAPL-3	1.00000(10)	0.906(1.1)	0.0520(1.9)	0.0570(5.3)	0.000
CENDL-2	1.00347(+ 34)	0.919(+1.5)	0.0518(- 35)	0.0521(-8.5)	0.653
ENDF/B-VI	0.99885(- 12)	0.914(+.88)	0.0520(+.04)	0.0527(-7.6)	0.658
JEF-1	1.00341(+.34)	0.911(+.55)	0.0523(+ 48)	0.0536(-5.9)	0.657
JEF-2	1.00073(+.07)	0.910(+.44)	0.0520(+.08)	0.0524(-8.0)	0.661
JENDL-3.1	1.00000(+ 00)	0.915(+.99)	0.0512(-1.5)	0.0535(-6.2)	0.657
WIMS-D/4	0.99807(- 19)	0.894(-1.3)	0.0529(+1.7)	0.0538(-5.6)	0.657

NOTE: Measured values and their % uncertainty (in brackets) for each lattice are given in row-1. Then follow the results using libraries based on different evaluated data files and the old WIMS library, and in brackets the % difference from measurements.

Table 3: Summary of average measurement uncertainties and average differences between calculated and measured values over all the benchmark lattices.

LATTICE	k_{eff}	ρ^{28}	δ^{25}	δ^{28}	C^*
Average	0 16	1 32	1 69	6 12	0 97
CENDL-2	0 24(0.26)	1 46(0 63)	-1 13(0 69)	-4 60(4.04)	-1.22(0.15)
ENDF/B-VI	-0 25(0.26)	1 03(0 63)	-0 69(0.69)	-3 67(4 09)	-0.52(0.16)
JEF-1	0 23(0.28)	0 50(0 56)	-0 28(0 58)	-2 24(3.88)	-0 63(0 11)
JEF-2	-0 05(0 28)	0 58(0.64)	-0 68(0.59)	-4 39(3 88)	-0.02(0.11)
JENDL-3 1	-0.19(0.29)	1.03(0.59)	-2.23(0 57)	-2.43(3.91)	-0.61(0.16)
WIMS-D/4	0 00(0 24)	-2 55(0 84)	0 49(0 81)	-2 66(3 36)	-1.90(0.23)

NOTE Average uncertainty over all lattices for each of the parameters is given. Then follow the mean differences between the calculated values and the measurements and in brackets the standard deviation for each of the evaluated libraries considered.

to the coarseness of the energy mesh in the fast neutron range, as observed in the analysis with the ENDF/B-IV data summarized in the previous section. Parameter C^* is predicted well with all the new libraries.

To make the comparison clearer, averages over all the benchmark lattices were calculated and are presented in Table 3. The average measurement uncertainty was calculated as the root-mean-square of the uncertainties for each parameter over all the lattices. For each parameter, for each of the data libraries considered, the simple mean of the differences between the calculation and the measurement was calculated. As a measure of the spread of the results, the standard deviations were also calculated. For parameter ρ^{28} the results of the BAPL-2 lattice were excluded from averageing.

On average the claculated k_{eff} with the new libraries differs by up to 0.25% from criticality. The spread in the results is comparable to that obtained with the old WIMS library. This result is not disappointing, since we must consider, that the old WIMS library was adjusted to predict well the k_{eff}.

A significant improvement compared to the results with the old WIMS library is observed in predicting ρ^{28}. The average difference from the measured values lies within the uncertainty interval of the measurements with all libraries except CENDL-2 where it is slightly outside. The spread in the results is also reduced compared to the old WIMS library.

Parameter δ^{25} is predicted within the uncertainty interval of the measurements with all libraries except JENDL-3.1 where it is slightly outside. The spread of the results is smaller than with the old WIMS library.

On average, parameter δ^{28} is underpredicted by a few percent, but within the uncertainty interval of the measurements. The average difference from the measurements and the spread in the results are comparable to that obtained with the old WIMS library.

Since the measurements of parameter C^* are not available for the BAPL lattices, the comparison is limited to the two TRX lattices. Good agreement with measurements is observed with all the libraries except CENDL-2 where the difference slightly exceeds the measurement uncertainty The average difference from the

measurements and the spread of the results with the old WIMS library are considerably larger

4 Conclusions

Nuclear cross section data processing methods for generating a WIMS library from evaluated nuclear cross section data files have been established. They have been verified by comparing the results obtained with the ENDF/B-IV library to the published results of highly accurate Monte Carlo calculations, using the same basic data.

Using the same data processing methods, WIMS libraries have been generated from several recently available evaluated data files. A number of selected thermal reactor benchmark lattices have been analysed The results show that there are no significant differences related to thermal reactor applications between the libraries.

By comparing the calculated results for the benchmark lattices with the measured values, good agreement is observed. Almost all the parameters lie within (or very close to) the uncertainty interval of the measurements. Comparing the results with those obtained with the old WIMS library it can be concluded, that a significant improvement in predicting global lattice parameters has been achieved, even though the multigroup constants libraries for WIMS were generated from first principles. No empirical adjustments on the new data libraries were made. The work presented in this paper forms a sound basis for a systematic updating of the WIMS library.

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Due to adequate computer resources, the computationally intensive nuclear data processing was performed at the IAEA on the VAX4300 computer.

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NSLINK

Preparing AMPX MASTER (SCALE) library from NJOY output

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ABSTRACT

The INAS code system (IRI-NJOY-AMPX-SCALE) is a large system of nuclear data bases and computer programs for reactor physics computations used at the Interfaculty Reactor Institute (IRI) at the Delft University of Technology. NSLINK (NJOY-SCALE-LINK) is part of the INAS code system. It is a computer code to convert the data in GENDF format from the NJOY code into SCALE-4 AMPX MASTER format, retaining the Nordheim resolved resonance treatment option. Reich-Moore resonance parameters are converted to multi-level Breit-Wigner resonance parameters. The AMPX MASTER library is the basic library in the SCALE code system.

1. Introduction

The INAS code system (IRI-NJOY-AMPX-SCALE) is a large system of FORTRAN 77 programs and nuclear data bases for reactor physics computations, see figure 1. Most of the codes orignate from the Oak Ridge National Laboratory (ORNL) AMPX [1] and SCALE [2] code packages, and were adapted to run under the VAX/VMS operating system on a VAX cluster computer system. SCALE is a well-known code system, well documented, and widely used and accepted. It includes modules for resonance, cell, Monte-Carlo, shielding, and depletion calculations. The basic library in the SCALE code system is the AMPX MASTER library. This is a fine-group structured data library including resonance parameters, one- and two-dimensional cross sections for neutron and gamma reaction types, and some general information. The library can be generated from ENDF/B, JEF or JENDL evaluated data files using the NJOY [3] cross section processing code. The fine-group structured output data from NJOY can be converted to AMPX MASTER library format with the interface program NSLINK (NJOY-SCALE-LINK).

2. Overview of the code

NSLINK is an extensively revised version of the original MILER [4] code. The Netherlands Energy Research Foundation (ECN), in Petten, The Netherlands, contributed to the revision of MILER and tested the present version 4.1 of the code as well. ECN makes use of NSLINK in the PASC (PETTEN-AMPX-SCALE) code system. The 4.1 version of the code is also used to update the AMPX MASTER library at AECL Research in Pinawa, Canada.

NSLINK version 3.0 can be used to generate AMPX MASTER libraries for the obsolete SCALE-3 [5] code system. This version is distributed via NEA DATA BANK, Paris, France and RSIC, Oak Ridge, USA.

NSLINK version 4.0 can be used to generate AMPX MASTER libraries for the SCALE-4 code system. Due to an other representation of the resolved resonance data in the AMPX MASTER library version 4, NSLINK version 4.0 is much simpler than version 3.0. However, the 4.0 version of the code is not distributed. NSLINK version 4.1 includes necessary updates to use ENDF-6 formats and output of the NJOY code version 91. NSLINK version 4.1 will be distributed soon.

3. Functionality of the code

The following overview is a summary of the functionality and adaptations from MILER to the NSLINK version 4.1 code:

[1] Thermal scattering matrices for different temperatures as generated by NJOY can be included in the library.

[2] Bondarenko factors can be generated from NJOY output.

[3] 'Nordheim' resolved resonance parameters can be included in the library. The Breit-Wigner parameters are read from the evaluated data files as JEF, ENDF/B or JENDL.

[4] The fission matrix available in the GENDF output file of NJOY is converted to a fission spectrum, which can be used in the SCALE code system. Until NJOY89, a full fission matrix was given in the GENDF file. To reduce the size of the fission matrix, this matrix is split into 3 parts: a spectrum part, a production part, and a matrix part. This is done by

version 91 of the NJOY code. NSLINK can read and convert fission matrices to fission spectra from the NJOY89 and NJOY91 code version as well.

[5] In the ENDF-6 format of the evaluated data files, the Reich-Moore (RM) representation of the resolved resonance parameters is used for some structural materials and actinides. The Nordheim resonance treatment can only be used with Breit-Wigner (BW) resolved resonance parameters. NSLINK converts RM parameters to BW parameters for actinides only (developed by ECN, Petten [6]).

[6] A nuclide or a material with scattering or absorption cross sections only can be processed properly.

[7] The (n,2n) cross section of a nuclide is recognized by the SCALE code system and by most other post processing codes properly, if reaction type MT=16 is used as identification. In the JEF 1.1 library, the reaction types MT=6,7,8,9, and 46,47,48,49 are used for the (n,2n) cross sections. Reaction types MT=51 until 83 are used for the (n,2n) cross sections in the JEF 2.2 library. A patch in the NJOY code was needed to process the (n,2n) cross sections from JEF 2.2! NSLINK version 4.1 converts these (n,2n) cross sections from both JEF 1.1 and JEF 2.2 to reaction type MT=16. In the ENDF/B-VI library, reaction type MT=16 is used for the (n,2n) cross section. In future format specifications of ENDF, only reaction type MT=16 is allowed for the (n,2n) cross section.

[8] Some small errors in NJOY and evaluation problems (zero values for some levels) are circumvented in NSLINK.

[9] Reaction types available in the GENDF output file of NJOY, but not used in the SCALE code system and/or other post processing codes at present, are removed.

Restrictions:

[1] Gamma production and/or photon interaction data cannot be processed correctly yet.

[2] Consistency of redundant cross sections is not tested. The RADE module of the AMPX code package can be used.

[3] BCD (ASCII) format of the GENDF output file of NJOY can be processed at present.

4. Benchmarking

Validation and verification of nuclear data and code package that are used for reactor physics calculations are a continuous care of our group. A number of standard criticality PWR and BWR reactor benchmarks are carried out to validate our AMPX MASTER libraries (XMAS structure, 172 groups). These libraries are generated from the JEF 1.1 and JEF 2.2 library as well. The reactor benchmarks are partly done in cooperation with other Dutch institutes (ECN at Petten, GKN at Dodewaard, and KEMA at Arnhem, The Netherlands).

5. Conclusions

The NSLINK version 4.1 is a powerful code to generate cross section libraries in AMPX MASTER format as used in the SCALE-4 code system. Bondarenko factors and/or Nordheim resonance parameters can be included in the library. ENDF formats 3, 4, 5, and 6 can be read as well. Reich-Moore resonance parameters are converted to Breit-Wigner resonance parameters for the actinides only. Fission spectra can be generated from NJOY89 and

NJOY91 output as well. The option to process gamma production and/or photon interaction data, and GENDF output from NJOY in binary format, will be made operational in the NSLINK code.

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[1] N.M. Greene et al. AMPX - A Modular Code System for Generating Coupled Multi-group Neutron Gamma libraries from ENDF/B, (March 1976). Revised to level of AMPX II, ORNL/TM-3706 (December 1978).

[2] SCALE-4 - A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, CCC-545 (February 1990).

[3] R.E. MacFarlane and D.W. Muir, NJOY87 - A Code System for Producing Pointwise and Multigroup Neutron and Photon Cross Sections from ENDF/B Evaluated Nuclear Data, PSR-171 (November 1987).

[4] G.C. Panini, MILER - Master Interface Library makER, abstract NEA 1198 (July 1988).

[5] SCALE-3.1 - A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, CCC-466 (March 1985; July 1986).

[6] R.C.L. van der Stad, C. Chonghai and H. Gruppelaar, Generation of a 219-group neutron cross-section library based upon JEF2.2 for actinides and long-lived fission products. ECN Petten, Contribution to the seminar on NJOY91 and THEMIS (April 1992).

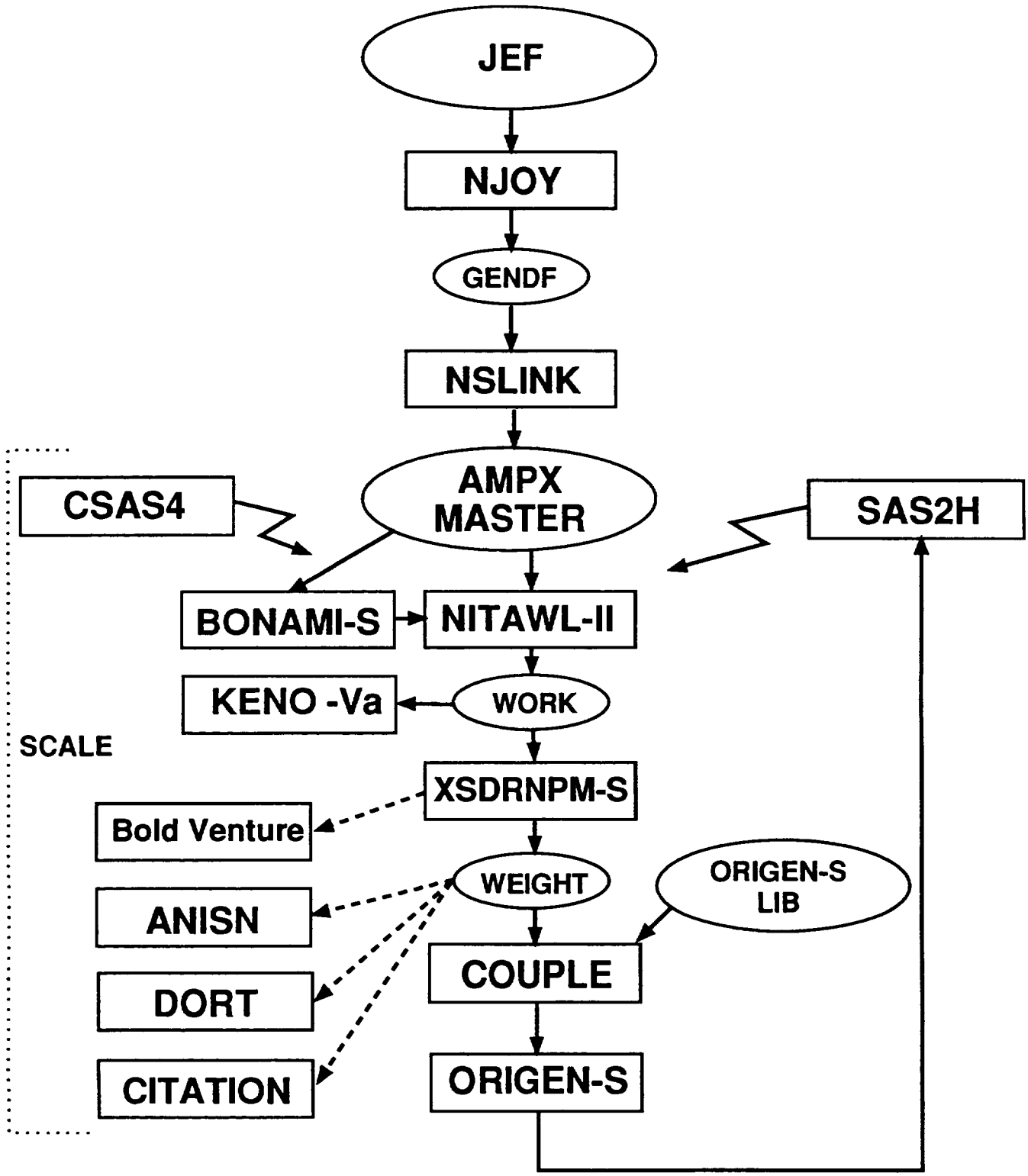

Fig. 1. Overview of the INAS code package

Cross Section Libraries Based on JENDL-3.1 and JENDL-3.2

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IAEA Specialists' Meeting on
Preparation of Processed Nuclear Data Libraries
for Thermal, Fast and Fusion Research
and Power Reactor Applications
December 8 - 10, 1993
IAEA Headquaters, Vienna

Abstract

This report presents a list of multigroup and pointwise libraries derived from JENDL-3.1 and JENDL 3.2 files. The libraries were processed in Japan for thermal, fast and fusion reactor applications. The characteristics of the group constants library are presented. The specifications used in processing are given. Comprehensive coupled neutron-photon calculations are performed in Japan using these working libraries. Several utility and interfacing routines are briefly mentioned in the paper.

Cross Section Libraries Based on JENDL-3.1 and JENDL-3.2

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History of JENDL-3

1983 Dec. **JENDL-3PR1** was released
for fusion neutronics application.
8 nuclides (^{6}Li,^{7}Li,^{9}Be,^{12}C,^{16}O,Cr,Fe,Ni)

1985 Mar. **JENDL-3PR2** was released.
3 nuclides(^{6}Li,^{7}Li,^{12}C) data were modified.

1987 Apr. **JENDL-3T Rev. 0** was released.
(1987.4.13 ~ 1988.3.17) , 32 nuclides.

1988 Mar. **JENDL-3T Rev. 1** was released.
(1988.3 ~ 1989.4.11)

1989 Dec. **JENDL-3.0** was released world-widely.

1990 Dec. **JENDL-3.1** was released.
addition of F. P. & revition of primitive errors

1993 Mar. **JENDL-3.2 and JENDL Fusion File**
will be released.

Cross Section Libraries based on JENDL-3.1

Group-wise type

Name	Number of group		Transport code
	neutron	gamma	
FUSION-J3 FUSION-40	125 42	40 21	ANISN, DOT, MORSE GMVP, etc.
JSSTDL	295	104	same as above
DDL-J3	125	---	ANISN-DD*1, DOT-DD*1 MORSE-DD*1,GMVP*2

*1 MORI T., et al. : JAERI-1314 (1988 Nov.).
*2 NAKAGAWA M., et al.: Nucl. Sci. Eng. 107, 58 (1991).

Point-wise type

Name	Transport code
FSXLIB-J3	MCNP
MVPLIB-J3#	MVP*1

Data sets are prepared for five temperature points. A generation code can make the data set of any temperature.

*1 MORI T., et al.: "MVP : A Continuous Energy Monte Carlo Code for Vector Supercomputers," Proc. Int'l Topical Mtg. Advances in Mathematics, Computations, and Reactor Physics, Apr. 28-May 2, 1991, Pittsburgh, PA, USA.

Referenses

FUSION-J3 / FUSION-40

* MAKI K., et al.:"Nuclear Group Constant Set Fusion-J3 for Fusion Reactor Nuclear Calculations Based on JENDL-3," JAERI-M 91-072 (1991 May) (In Japanese).

JSSTDL

* HASEGAWA A.:"JSSTDL-295n-104g; a Common Nuclear Group Cross-Section Library Based on JENDL-3 Nuclear Data File," JAERI-M 91-062, pp15-25 (1991 March) .
* HASEGAWA A.:"Development of a Common Nuclear Group Constants Library System: JSSTDL-295n-104 Based on JENDL-3 Nuclear Data Library," Proc. Int'l Conf. on Nuclear Data, Juelich, Germany, May 13-17, 1991.

DDL-J3

* NAKAGAWA M., MORI T., KANEKO K. "Production of Multi-group Double Differential Form Cross Section Library Using JENDL-3 and Benchmark Test of Fusion neutronics Problems," JAERI-M 90-097 (1990 June) (In Japanese).

FSXLIB-J3

* KOSAKO K., OYAMA Y., MAEKAWA H :"FSXLIB-J3 . MCNP Continuous Energy Cross Section Library Based on JENDL-3," JAER-M 91-187 (1991 Nov)

FUSION-J3 / FUSION-40

Type	:	group-independent format
Processing Code	:	MACS-N for neutron
		NJOY for gamma-ray
Photon Intraction Data	:	DLC-99
Weighting Function	:	1/E & Maxwell for thermal group
Temperature	:	300 K
σ_0 (b.g. cross section)	:	infinite dilution
Number of Nuclides	:	40
Anisotropic P_l Order	:	5

Table 2.2 Forty nuclide identification number available in FUSION-J3 nuclear group constant set.

No	Nuclide	MT No for ANISN input P0 - P5	No	Nuclide	MT No for ANISN input P0 - P5
1	^{1}H	1 – 6	21	Ca	121 – 126
2	^{2}D	7 – 12	22	Ti	127 – 132
3	^{3}He	13 – 18	23	^{51}V	133 – 138
4	^{4}He	19 – 24	24	Cr	139 – 144
5	^{6}Li	25 – 30	25	^{55}Mn	145 – 150
6	^{7}Li	31 – 36	26	Fe	151 – 156
7	^{9}Be	37 – 42	27	Co	157 – 162
8	^{10}B	43 – 48	28	Ni	163 – 168
9	^{11}B	49 – 54	29	Cu	169 – 174
10	^{12}C	55 – 60	30	Zr	175 – 180
11	^{14}N	61 – 66	31	^{93}Nb	181 – 186
12	^{16}O	67 – 72	32	Mo	187 – 192
13	^{19}F	73 – 78	33	Cd	193 – 198
14	^{23}Na	79 – 84	34	W	199 – 204
15	Mg	85 – 90	35	Pb	205 – 210
16	^{27}Al	91 – 96	36	^{209}Bi	211 – 216
17	Si	97 – 102	37	^{232}Th	217 – 222
18	^{31}P	103 – 108	38	^{235}U	223 – 228
19	S	109 – 114	39	^{238}U	229 – 234
20	K	115 – 120	40	^{239}Pu	235 – 240

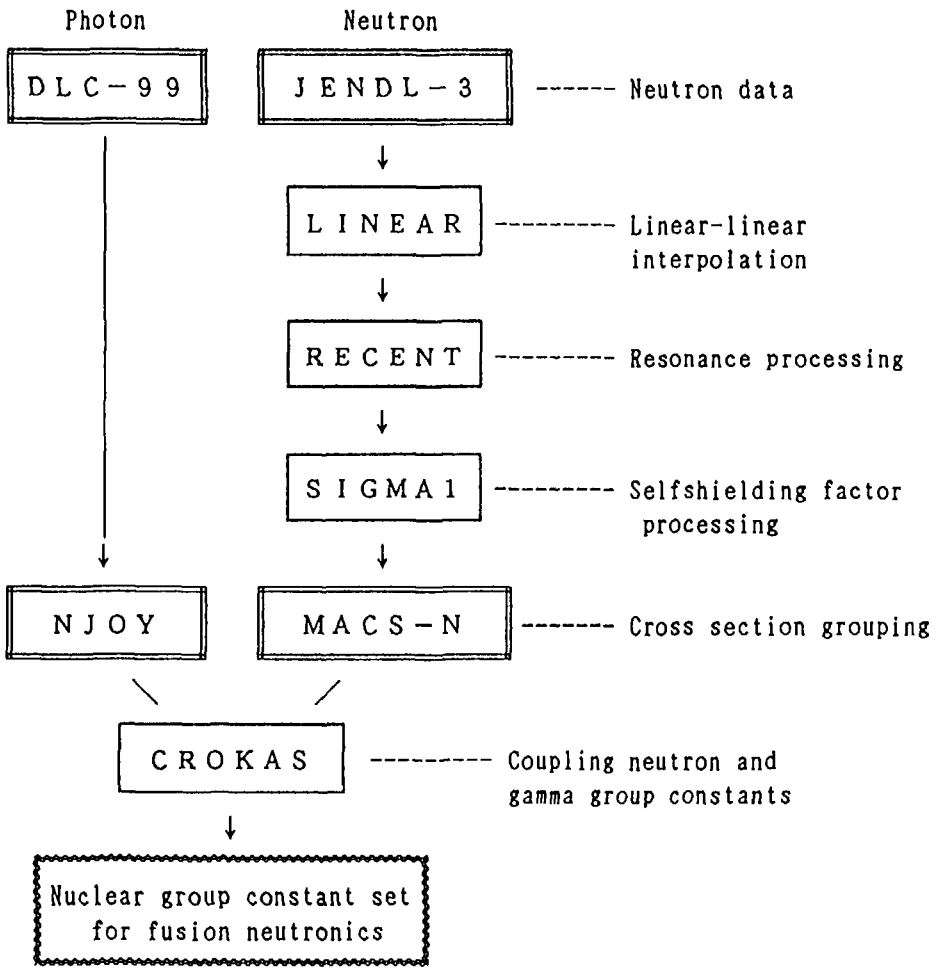


Fig. 2.1 Cross section processing flow for nuclear group constant set.

Characteristics of JSSTDL System

- **General Purpose Group Constants Library**
  - Shielding & Criticality Calculation
  - FBR and Fusion Application
  - Convertible to all group structures currently utilized in Japan.
  - Direct link to ANISN DOT MORSE codes.
- **JENDL-3 Based**
  - Latest Nuclear Data
- **Complete Package**
  - Libray and utility codes
- **Benchmark Test**
  - Shielding calculation
  - Small high enrichment FBR assemblies

Group Constants Specification

Group structure	neutron:295 20 MeV -- 1.0E-5 eV gamma:104 50 MeV -- 1 keV
Weighting spectrum	Maxwellian from 1.0E-5 to 0.3224 eV the rest is 1/E.
Resonance reconst-ruction tolerance	0.1%
Self-shielding factor Temperature grid σ_0 grid f-tab reactions	 300 600 900 2100 Kelvin 0 $_0$ 1778 1 10 10^2 10^3 10^4 10^5 10^6 barn total, elastic, capture, fission
Anisotropic P_l order	5
Photon production data	total= fission + capture + + otherthan fission and capture

Covered Group Boundaries

Table 1.2 Neutron group boundaries considered in JSSTDL system

Library name	groups
JSD-100	100
JSD-1000	100
BERMUDA-121	121
FNS-125	125
VITAMIN-C	171
VITAMIN-J(E+C)	175
GICX-42	42
ABBN-25	25
JFS-New	70
GAM-123*	92
MGCL-137*	91
WIMS-69*	28

n b *: fast group only

Table 1.3 Gamma group boundaries considered in JSSTDL system

Library name	groups
CSEWG-94	94
LANL-12	12
STEINER-21	21
STRAKER-22	22
LANL-48	48
LANL-24	24
BERMUDA-36	36
HONEYCOMB -15	15

Processing Out-Line.

process	CODE used	Process comments
pre-processing	LINAER RECENT-J SIGMA1	linearlized data generation resonance reconstruction for 0. K Doppler broadened cross-section
Processing	Prof. GROUCH-G/B	group averaging process self-shielding factor cal group transfer matrices cal
post-processing	GLIBMK	group library compilation

Data Sources

Data item	Source
Neutron Cross-section	JENDL-3
Photon Production Cross-section	JENDL-3
Photon Transport Cross-section	GAMLEG-JR

Developed Utility Codes

code		
neutron-lib.	gamma-lib.	comment
CONVJSS	CONVJGG	format conversion code Binary <==> EBCDIC
CONDNSJ	CONDNSJG	collapsing code to any broad group 295 <==> User specified group (JSD100, BERMUDA-125, etc)
NACROJ	MACROJG	region dependent macro-scopic cross section creation for ANISN,DOT,MORSE code.

Direct Link to ANISN DOT MORSE

- Direct link to ANISN DOT MORSE code
  via MACROJ code (neutron cal.)
  via MACROJG code (neutron+gamma cal.)

Processed Nuclides from JENDL-3

order	gamma	Z	ch	A	MAT	NCODE	comment
1	G	1	H	1	3011	117	
2		1	H	2	3012	127	
3		2	He	3	3021	237	
4		2	He	4	3022	247	
5	G	3	Li	6	3031	367	
6	G	3	Li	7	3032	377	
7	G	4	Be	9	3041	497	
8	G	5	B	10	3051	507	
9	G	5	B	11	3052	517	
10	G	6	C	12	3061	607	
11	G	7	N	14	3072	707	
12	G	8	O	16	3081	807	
13		9	F	19	3091	907	
14	G	11	Na	23	3111	1107	
15	G	12	Mg	0	3120	1207	
16	G	13	Al	27	3131	1307	
17	G	14	Si	0	3140	1407	
18		15	P	31	3151	1507	
19		16	S	0	3160	1607	
20		19	K	0	3190	1907	
21	G	20	Ca	0	3200	2007	
22	G	22	Ti	0	3220	2207	
23		23	V	51	3231	2307	
24	G	24	Cr	0	3240	2407	
25	G	25	Mn	55	3251	2557	
26	G	26	Fe	0	3260	2607	
27	G	28	Ni	0	3280	2807	
28	G	29	Cu	0	3290	2907	
29		31	Ga	0	3100	3108	ENDF/B-VI
30	G	40	Zr	0	3400	4007	
31	G	41	Nb	93	3411	4137	
32	G	42	Mo	0	3420	4207	
33		48	Cd	0	3480	4807	
34	G	63	Eu	0	3630	6307	
35	G	72	Hf	0	3720	7207	
36	G	73	Ta	81	3731	7307	
37	G	74	W	0	3740	7407	
38	G	82	Pb	0	3820	8207	
39	G	83	Bi	209	3831	8397	

order	gamma	Z	ch	A	MAT	NCODE	comment
40		90	Th	232	3905	9027	
41		92	U	233	3922	9237	
42		92	U	234	3923	9247	
43	G	92	U	235	3294	9257	
44		92	U	236	3925	9267	
45	G	92	U	238	3926	9287	
46		93	Np	237	3931	9377	
47		93	Np	239	3932	9397	
48		94	Pu	236	3941	9467	
49		94	Pu	238	3942	9487	
50	G	94	Pu	239	3934	9497	
51		94	Pu	240	3944	9407	
52		94	Pu	241	3945	9417	
53		94	Pu	242	3946	9427	
54		95	Am	241	3951	9517	
55		95	Am	242m	3953	9597	
56		95	Am	243	3954	9537	
57		96	Cm	242	3962	9627	
58		96	Cm	243	3963	9637	
59		96	Cm	244	3964	9647	
60		96	Cm	245	3965	9657	
61		96	Cm	246	3966	9667	
62		96	Cm	247	3967	9677	
63		96	Cm	248	3968	9687	

n.b.

gamma: G means data given
Z : atomic number
ch : chemical symbol
A . mass number
MAT : MAT number of JENDL-3
CODE . code number in JSSTDL lib.

Table A.1.1.3 Table of neutron energy group structure

TABLE OF NEUTRON ENERGY GROUP STRUCTURE

GROUP	UP-ENERGY	LETHAGY	DELTA-U	GROUP	UP-ENERGY	LETHAGY	DELTA-U	GROUP	UP-ENERGY	LETHAGY	DELTA-U
1	1.96403E+07	-3.38527E-01	1.25001E-01	41	1.08120E+07	2.58400E-01	1.55757E-02	81	4.06570E+06	1.23647E+00	1.62916E-02
2	1.73325E+07	-2.13526E-01	2.49969E-02	42	1.06449E+07	2.73976E-01	1.24969E-02	82	4.00000E+06	1.25276E+00	2.12109E-02
3	1.69046E+07	-1.88529E-01	2.50020E-02	43	1.05127E+07	2.86473E-01	3.11538E-03	83	3.91605E+06	1.27397E+00	6.24996E-02
4	1.64872E+07	-1.63527E-01	1.56613E-02	44	1.04800E+07	2.89588E-01	1.56757E-02	84	3.67879E+06	1.33647E+00	6.25010E-02
5	1.62310E+07	-1.47866E-01	1.55851E-02	45	1.03170E+07	3.05264E-01	1.56299E-02	85	3.45590E+06	1.39897E+00	3.74981E-02
6	1.59800E+07	-1.32281E-01	1.56411E-02	46	1.01570E+07	3.20894E-01	1.55780E-02	86	3.32871E+06	1.43647E+00	2.50012E-02
7	1.57320E+07	-1.16640E-01	3.11316E-03	47	1.00000E+07	3.36472E-01	5.00005E-02	87	3.24652E+06	1.46147E+00	2.49979E-02
8	1.56831E+07	-1.13526E-01	1.24989E-02	48	9.51229E+06	3.86472E-01	1.24996E-02	88	3.16637E+06	1.48647E+00	3.75033E-02
9	1.54883E+07	-1.01028E-01	1.56366E-02	49	9.39413E+06	3.98972E-01	8.56643E-03	89	3.04982E+06	1.52397E+00	1.24982E-02
10	1.52480E+07	-8.53910E-02	1.55985E-02	50	9.31400E+06	4.07538E-01	2.89340E-02	90	3.01194E+06	1.53647E+00	4.99986E-02
11	1.50120E+07	-6.97926E-02	6.26124E-03	51	9.04837E+06	4.36472E-01	2.49995E-02	91	2.86505E+06	1.58647E+00	5.00000E-02
12	1.49183E+07	-6.35314E-02	9.38141E-03	52	8.82497E+06	4.61471E-01	2.50001E-02	92	2.72532E+06	1.63647E+00	1.25021E-02
13	1.47790E+07	-5.41499E-02	1.56231E-02	53	8.60708E+06	4.86472E-01	3.75002E-02	93	2.69146E+06	1.64897E+00	3.74997E-02
14	1.45499E+07	-3.85268E-02	1.56477E-02	54	8.29029E+06	5.23972E-01	3.53932E-03	94	2.59240E+06	1.68647E+00	2.50014E-02
15	1.43240E+07	-2.28792E-02	9.34964E-03	55	8.26100E+06	5.27511E-01	8.96028E-03	95	2.52839E+06	1.71147E+00	1.12920E-02
16	1.41907E+07	-1.35295E-02	6.27024E-03	56	8.18731E+06	5.36472E-01	5.00000E-02	96	2.50000E+06	1.72277E+00	1.37055E-02
17	1.41020E+07	-7.25931E-03	1.56516E-02	57	7.78801E+06	5.86472E-01	5.00006E-02	97	2.46597E+06	1.73647E+00	3.32981E-02
18	1.38830E+07	8.39136E-03	3.08046E-03	58	7.40818E+06	6.36472E-01	1.25006E-02	98	2.38521E+06	1.76977E+00	4.20554E-03
19	1.38403E+07	1.14724E-02	8.73002E-03	59	7.31615E+06	6.48973E-01	3.74993E-02	99	2.37520E+06	1.77398E+00	4.19793E-03
20	1.37200E+07	2.02022E-02	3.76802E-03	60	7.04688E+06	6.86472E-01	2.50003E-02	100	2.36525E+06	1.77817E+00	8.29991E-03
21	1.36684E+07	2.39700E-02	1.25007E-02	61	6.87289E+06	7.11472E-01	2.49997E-02	101	2.34570E+06	1.78647E+00	1.66966E-02
22	1.34986E+07	3.64705E-02	3.16088E-03	62	6.70320E+06	7.36472E-01	1.66706E-02	102	2.30686E+06	1.80317E+00	3.33029E-02
23	1.34560E+07	3.96314E-02	1.55785E-02	63	6.59238E+06	7.53143E-01	1.41123E-02	103	2.23130E+06	1.83647E+00	4.99992E-02
24	1.32480E+07	5.52102E-02	1.56717E-02	64	6.50000E+06	7.67255E-01	6.71791E-03	104	2.12248E+06	1.88647E+00	1.25020E-02
25	1.30420E+07	7.08823E-02	1.56096E-02	65	6.45648E+06	7.73973E-01	1.24995E-02	105	2.09611E+06	1.89897E+00	3.74958E-02
26	1.28400E+07	8.64919E-02	1.56198E-02	66	6.37628E+06	7.86472E-01	4.99992E-02	106	2.01897E+06	1.93647E+00	2.50109E-02
27	1.26410E+07	1.02111E-01	9.36258E-03	67	6.06531E+06	8.36472E-01	5.00003E-02	107	1.96910E+06	1.96148E+00	2.49910E-02
28	1.25232E+07	1.11474E-01	6.26400E-03	68	5.76950E+06	8.86472E-01	1.25018E-02	108	1.92050E+06	1.98647E+00	2.61125E-02
29	1.24450E+07	1.17738E-01	1.56298E-02	69	5.69782E+06	8.98974E-01	3.74979E-02	109	1.87100E+06	2.01258E+00	1.13902E-02
30	1.22520E+07	1.33367E-01	3.10641E-03	70	5.48812E+06	9.36471E-01	2.50015E-02	110	1.84981E+06	2.02397E+00	1.24952E-02
31	1.22140E+07	1.36474E-01	1.24980E-02	71	5.35261E+06	9.61473E-01	2.49988E-02	111	1.82684E+06	2.03647E+00	5.00023E-02
32	1.20623E+07	1.48972E-01	1.56495E-02	72	5.22046E+06	9.86471E-01	2.35411E-02	112	1.73774E+06	2.08647E+00	4.99997E-02
33	1.18750E+07	1.64621E-01	1.56161E-02	73	5.09900E+06	1.01001E+00	1.39605E-02	113	1.65299E+06	2.13647E+00	1.25039E-02
34	1.16910E+07	1.80238E-01	6.23791E-03	74	5.02831E+06	1.02397E+00	1.24995E-02	114	1.63245E+06	2.14898E+00	3.74979E-02
35	1.16183E+07	1.86476E-01	9.36523E-03	75	4.96585E+06	1.03647E+00	4.99985E-02	115	1.57237E+06	2.18647E+00	2.50053E-02
36	1.15100E+07	1.95841E-01	1.56298E-02	76	4.72367E+06	1.08647E+00	4.49594E-02	116	1.53354E+06	2.21148E+00	2.49912E-02
37	1.13315E+07	2.11471E-01	1.56090E-02	77	4.51600E+06	1.13143E+00	5.04153E-03	117	1.49569E+06	2.23647E+00	3.75072E-02
38	1.11560E+07	2.27079E-01	9.39326E-03	78	4.49329E+06	1.13647E+00	1.25008E-02	118	1.44063E+06	2.27398E+00	1.24960E-02
39	1.10517E+07	2.36473E-01	6.23569E-03	79	4.43747E+06	1.14897E+00	6.24994E-02	119	1.42274E+06	2.28647E+00	1.61124E-02
40	1.09830E+07	2.42708E-01	1.56920E-02	80	4.16862E+06	1.21147E+00	2.49991E-02	120	1.40000E+06	2.30258E+00	3.38892E-02

TABLE OF NEUTRON ENERGY GROUP STRUCTURE

GROUP	UP-ENERGY	LETHAGY	DELTA-U
121	1.35335E+06	2.33647E+00	4.99972E-02
122	1.28735E+06	2.38647E+00	1.25065E-02
123	1.27135E+06	2.39898E+00	3.74978E-02
124	1.22456E+06	2.43648E+00	2.50046E-02
125	1.19432E+06	2.46148E+00	2.49933E-02
126	1.16484E+06	2.48647E+00	3.75066E-02
127	1.12196E+06	2.52398E+00	1.24935E-02
128	1.10803E+06	2.53647E+00	4.62034E-02
129	1.05800E+06	2.58268E+00	3.80691E-03
130	1.05398E+06	2.58648E+00	4.99869E-02
131	1.00259E+06	2.63647E+00	4.17018E-02
132	9.61640E+05	2.67817E+00	3.33083E-02
133	9.30137E+05	2.71148E+00	2.49910E-02
134	9.07180E+05	2.73647E+00	5.00004E-02
135	8.62936E+05	2.78647E+00	5.00002E-02
136	8.20850E+05	2.83647E+00	2.57287E-02
137	8.00000E+05	2.86220E+00	2.42709E-02
138	7.80817E+05	2.88647E+00	5.00002E-02
139	7.42736E+05	2.93647E+00	2.50094E-02
140	7.24391E+05	2.96148E+00	2.49911E-02
141	7.06512E+05	2.98647E+00	5.00000E-02
142	6.72055E+05	3.03647E+00	4.99992E-02
143	6.39279E+05	3.08647E+00	5.00000E-02
144	6.08101E+05	3.13647E+00	4.99993E-02
145	5.78444E+05	3.18647E+00	2.17477E-02
146	5.66000E+05	3.20822E+00	3.26331E-03
147	5.64156E+05	3.21148E+00	2.49908E-02
148	5.50232E+05	3.23647E+00	4.99998E-02
149	5.23397E+05	3.28647E+00	4.57322E-02
150	5.00000E+05	3.33220E+00	4.26713E-03
151	4.97871E+05	3.33647E+00	1.00001E-01
152	4.50492E+05	3.43647E+00	2.50098E-02
153	4.39365E+05	3.46148E+00	7.49903E-02
154	4.07622E+05	3.53647E+00	1.88758E-02
155	4.00000E+05	3.55535E+00	3.11244E-02
156	3.87742E+05	3.58647E+00	4.99990E-02
157	3.68832E+05	3.63647E+00	7.50102E-02
158	3.42178E+05	3.71148E+00	2.49898E-02
159	3.33733E+05	3.73647E+00	1.00000E-01
160	3.01974E+05	3.83647E+00	1.16012E-02
161	2.98491E+05	3.84807E+00	4.29748E-03
162	2.97211E+05	3.85237E+00	9.10222E-03
163	2.94518E+05	3.86147E+00	2.49977E-02
164	2.87247E+05	3.88647E+00	1.48956E-02
165	2.83000E+05	3.90137E+00	3.51074E-02
166	2.73237E+05	3.93647E+00	2.50104E-02
167	2.66488E+05	3.96148E+00	7.49900E-02
168	2.47235E+05	4.03647E+00	5.00009E-02
169	2.35177E+05	4.08647E+00	4.99968E-02
170	2.23708E+05	4.13647E+00	5.00030E-02
171	2.12797E+05	4.18647E+00	2.50098E-02
172	2.07541E+05	4.21148E+00	2.49891E-02
173	2.02419E+05	4.23647E+00	1.20225E-02
174	2.00000E+05	4.24850E+00	3.79771E-02
175	1.92547E+05	4.28647E+00	5.00020E-02
176	1.83156E+05	4.33647E+00	4.99965E-02
177	1.74224E+05	4.38647E+00	5.00000E-02
178	1.65727E+05	4.43647E+00	2.50136E-02
179	1.61633E+05	4.46148E+00	2.49891E-02
180	1.57644E+05	4.48647E+00	4.99975E-02
181	1.49956E+05	4.53647E+00	5.00040E-02
182	1.42642E+05	4.58647E+00	1.15782E-02
183	1.41000E+05	4.59805E+00	3.84165E-02
184	1.35686E+05	4.63647E+00	5.00040E-02
185	1.29068E+05	4.68647E+00	2.50103E-02
186	1.25880E+05	4.71148E+00	2.49920E-02
187	1.22773E+05	4.73648E+00	4.99940E-02
188	1.16786E+05	4.78647E+00	5.00025E-02
189	1.11090E+05	4.83647E+00	1.05171E-01
190	1.00000E+05	4.94164E+00	1.98291E-02
191	9.80366E+04	4.96147E+00	1.25002E-01
192	8.65169E+04	5.08647E+00	4.75003E-02
193	8.25034E+04	5.13397E+00	3.70987E-02
194	7.94987E+04	5.17107E+00	4.04109E-02
195	7.63501E+04	5.21148E+00	5.86896E-02
196	7.19981E+04	5.27017E+00	6.63011E-02
197	6.73794E+04	5.33647E+00	1.25010E-01
198	5.94615E+04	5.46148E+00	4.99882E-02
199	5.65622E+04	5.51147E+00	7.50002E-02
200	5.24752E+04	5.58647E+00	1.20888E-01
201	4.65000E+04	5.70736E+00	4.11168E-03
202	4.63092E+04	5.71147E+00	1.25001E-01
203	4.08677E+04	5.83647E+00	1.25012E-01
204	3.60652E+04	5.96148E+00	4.99878E-02
205	3.43067E+04	6.01147E+00	7.50005E-02
206	3.18278E+04	6.08647E+00	1.10397E-01
207	2.85011E+04	6.19687E+00	1.46145E-02
208	2.80876E+04	6.21148E+00	3.94876E-02
209	2.70001E+04	6.25097E+00	3.55005E-02
210	2.60584E+04	6.28647E+00	5.00007E-02
211	2.47875E+04	6.33647E+00	2.49998E-02
212	2.41755E+04	6.36147E+00	2.50002E-02
213	2.35786E+04	6.38647E+00	7.49998E-02
214	2.18749E+04	6.46147E+00	1.72869E-02
215	2.15000E+04	6.47876E+00	1.07715E-01
216	1.93045E+04	6.58647E+00	1.25010E-01
217	1.70360E+04	6.71148E+00	1.24988E-01
218	1.50344E+04	6.83647E+00	2.50000E-01
219	1.17088E+04	7.08647E+00	9.99962E-02
220	1.05946E+04	7.18647E+00	5.77595E-02
221	1.00000E+04	7.24423E+00	9.22447E-02
222	9.11882E+03	7.33647E+00	2.50001E-01
223	7.10174E+03	7.58647E+00	2.50000E-01
224	5.53084E+03	7.83647E+00	1.73473E-01
225	4.65000E+03	8.00994E+00	7.65277E-02
226	4.30742E+03	8.08647E+00	1.49998E-01
227	3.70744E+03	8.23647E+00	5.21363E-02
228	3.51910E+03	8.28861E+00	4.78639E-02
229	3.35463E+03	8.33647E+00	1.00002E-01
230	3.03539E+03	8.43647E+00	9.99979E-02
231	2.74654E+03	8.53647E+00	4.99999E-02
232	2.61259E+03	8.58647E+00	5.00011E-02
233	2.48517E+03	8.63647E+00	1.00002E-01
234	2.24867E+03	8.73647E+00	4.10863E-03
235	2.23945E+03	8.74058E+00	4.07625E-02
236	2.15000E+03	8.78134E+00	5.51293E-02
237	2.03468E+03	8.83647E+00	2.50000E-01
238	1.58461E+03	9.08647E+00	1.06096E-01
239	1.42510E+03	9.19257E+00	1.43900E-01
240	1.23410E+03	9.33647E+00	2.10342E-01

TABLE OF NEUTRON ENERGY GROUP STRUCTURE

GROUP	UP-ENERGY	LETHAGY	DELTA-U
241	1.00000E+03	9.54681E+00	3.96602E-02
242	9.61116E+02	9.58647E+00	5.80652E-02
243	9.06898E+02	9.64454E+00	1.91935E-01
244	7.48518E+02	9.83647E+00	2.49999E-01
245	5.82947E+02	1.00865E+01	2.26059E-01
246	4.65000E+02	1.03125E+01	2.39424E-02
247	4.53999E+02	1.03365E+01	2.12020E-01
248	3.67262E+02	1.05485E+01	3.79799E-02
249	3.53575E+02	1.05865E+01	2.50002E-01
250	2.75364E+02	1.08365E+01	2.49999E-01
251	2.14454E+02	1.10865E+01	2.50000E-01
252	1.67017E+02	1.13365E+01	1.15977E-01
253	1.48728E+02	1.14524E+01	1.34023E-01
254	1.30073E+02	1.15865E+01	2.50000E-01
255	1.01301E+02	1.18365E+01	1.29261E-02
256	1.00000E+02	1.18494E+01	2.37075E-01
257	7.88932E+01	1.20865E+01	4.39439E-02
258	7.55014E+01	1.21304E+01	2.06056E-01
259	6.14421E+01	1.23365E+01	2.45811E-01
260	4.80520E+01	1.25823E+01	4.18770E-03
261	4.78512E+01	1.25865E+01	2.86438E-02
262	4.65000E+01	1.26151E+01	2.21358E-01
263	3.72665E+01	1.28365E+01	2.49999E-01
264	2.90232E+01	1.30865E+01	4.66630E-02
265	2.77000E+01	1.31331E+01	2.03336E-01
266	2.26033E+01	1.33365E+01	5.00431E-02
267	2.15000E+01	1.33865E+01	1.99955E-01
268	1.76035E+01	1.35865E+01	9.75111E-02
269	1.59680E+01	1.36840E+01	1.52490E-01
270	1.37096E+01	1.38365E+01	2.50005E-01
271	1.06770E+01	1.40865E+01	6.55068E-02
272	1.00000E+01	1.41520E+01	1.23763E-02
273	9.87700E+00	1.41644E+01	1.72113E-01
274	8.31529E+00	1.43365E+01	2.50001E-01
275	6.47595E+00	1.45865E+01	2.49999E-01
276	5.04348E+00	1.48365E+01	8.12292E-02
277	4.65000E+00	1.49177E+01	1.50573E-01
278	4.00000E+00	1.50683E+01	1.81995E-02
279	3.92786E+00	1.50865E+01	2.50000E-01
280	3.05902E+00	1.53365E+01	2.49999E-01
281	2.38237E+00	1.55865E+01	1.02628E-01
282	2.15000E+00	1.56891E+01	1.47373E-01
283	1.85539E+00	1.58365E+01	2.49999E-01
284	1.44498E+00	1.60865E+01	2.50001E-01
285	1.12535E+00	1.63365E+01	1.18094E-01
286	1.00000E+00	1.64546E+01	1.31904E-01
287	8.76425E-01	1.65865E+01	2.50001E-01
288	6.82560E-01	1.68365E+01	2.50001E-01
289	5.31578E-01	1.70865E+01	1.33813E-01
290	4.65000E-01	1.72203E+01	1.16186E-01
291	4.13994E-01	1.73365E+01	2.50018E-01
292	3.22413E-01	1.75865E+01	4.05196E-01
293	2.15000E-01	1.79917E+01	7.65467E-01
294	1.00000E-01	1.87572E+01	4.60517E+00
295	1.00000E-03	2.33623E+01	4.60516E+00
296	1.00001E-05	2.79675E+01	0.0

GROUP	UP-ENERGY	LETHAGY	DELTA-U

Table A.1.1.4 Table of gamma energy group structure

TABLE OF GAMMA ENERGY GROUP STRUCTURE

GROUP	UP-ENERGY	LETHAGY	DELTA-U
1	5.00000E+07	-1.27297E+00	5.10826E-01
2	3.00000E+07	-7.62140E-01	4.05465E-01
3	2.00000E+07	-3.56675E-01	1.62519E-01
4	1.70000E+07	-1.94156E-01	1.94156E-01
5	1.40000E+07	0.0	1.54151E-01
6	1.20000E+07	1.54150E-01	8.70114E-02
7	1.10000E+07	2.41162E-01	3.70413E-02
8	1.06000E+07	2.78203E-01	5.82689E-02
9	1.00000E+07	3.36472E-01	4.60440E-02
10	9.55000E+06	3.82516E-01	5.93166E-02
11	9.00000E+06	4.41833E-01	5.71584E-02
12	8.50000E+06	4.98991E-01	6.06247E-02
13	8.00000E+06	5.59616E-01	3.17487E-02
14	7.75000E+06	5.91364E-01	3.27899E-02
15	7.50000E+06	6.24154E-01	3.39016E-02
16	7.25000E+06	6.58056E-01	3.50913E-02
17	7.00000E+06	6.93147E-01	3.63677E-02
18	6.75000E+06	7.29515E-01	3.77404E-02
19	6.50000E+06	7.67255E-01	3.92207E-02
20	6.25000E+06	8.06476E-01	4.08220E-02
21	6.00000E+06	8.47298E-01	4.25596E-02
22	5.75000E+06	8.89857E-01	4.44518E-02
23	5.50000E+06	9.34309E-01	1.83492E-02
24	5.40000E+06	9.52658E-01	3.77404E-02
25	5.20000E+06	9.90399E-01	3.92207E-02
26	5.00000E+06	1.02962E+00	6.18754E-02
27	4.70000E+06	1.09149E+00	4.34852E-02
28	4.50000E+06	1.13498E+00	2.24729E-02
29	4.40000E+06	1.15745E+00	4.65200E-02
30	4.20000E+06	1.20397E+00	4.87902E-02
31	4.00000E+06	1.25276E+00	2.53178E-02
32	3.90000E+06	1.27808E+00	2.59755E-02
33	3.80000E+06	1.30406E+00	4.02739E-02
34	3.65000E+06	1.34433E+00	4.19642E-02
35	3.50000E+06	1.38629E+00	4.88902E-02
36	3.33300E+06	1.43518E+00	5.14039E-02
37	3.16600E+06	1.48659E+00	5.38567E-02
38	3.00000E+06	1.54045E+00	5.72761E-02
39	2.83300E+06	1.59772E+00	6.07570E-02
40	2.66600E+06	1.65848E+00	6.42885E-02
41	2.50000E+06	1.72277E+00	6.91358E-02
42	2.33300E+06	1.79190E+00	3.62248E-02
43	2.25000E+06	1.82813E+00	3.80481E-02
44	2.16600E+06	1.86617E+00	7.97350E-02
45	2.00000E+06	1.94591E+00	6.45385E-02
46	1.87500E+06	2.01045E+00	6.89929E-02
47	1.75000E+06	2.07944E+00	5.27982E-02
48	1.66000E+06	2.13224E+00	3.68140E-02
49	1.60000E+06	2.16905E+00	6.45385E-02
50	1.50000E+06	2.23359E+00	5.48082E-02
51	1.42000E+06	2.28840E+00	3.22032E-02
52	1.37500E+06	2.32060E+00	3.32748E-02
53	1.33000E+06	2.35388E+00	6.20354E-02
54	1.25000E+06	2.41591E+00	4.08220E-02
55	1.20000E+06	2.45674E+00	6.45385E-02
56	1.12500E+06	2.52127E+00	1.17783E-01
57	1.00000E+06	2.63906E+00	1.05361E-01
58	9.00000E+05	2.74442E+00	3.96653E-02
59	8.65000E+05	2.78408E+00	4.73462E-02
60	8.25000E+05	2.83143E+00	3.07717E-02
61	8.00000E+05	2.86220E+00	6.45385E-02
62	7.50000E+05	2.92674E+00	6.89929E-02
63	7.00000E+05	2.99573E+00	3.63677E-02
64	6.75000E+05	3.03210E+00	3.77404E-02
65	6.50000E+05	3.06984E+00	3.92207E-02
66	6.25000E+05	3.10906E+00	4.08220E-02
67	6.00000E+05	3.14988E+00	4.25596E-02
68	5.75000E+05	3.19244E+00	4.44518E-02
69	5.50000E+05	3.23689E+00	4.65200E-02
70	5.25000E+05	3.28341E+00	9.56951E-03
71	5.20000E+05	3.29298E+00	1.55042E-02
72	5.12000E+05	3.30849E+00	3.91390E-03
73	5.10000E+05	3.31240E+00	1.98027E-02
74	5.00000E+05	3.33220E+00	1.05361E-01
75	4.50000E+05	3.43756E+00	5.71584E-02
76	4.25000E+05	3.49472E+00	6.06247E-02
77	4.00000E+05	3.55535E+00	6.45385E-02
78	3.75000E+05	3.61989E+00	6.89929E-02
79	3.50000E+05	3.68888E+00	7.41080E-02
80	3.25000E+05	3.76299E+00	8.00428E-02
81	3.00000E+05	3.84303E+00	1.43101E-01
82	2.60000E+05	3.98613E+00	1.67054E-01
83	2.20000E+05	4.15318E+00	9.53102E-02
84	2.00000E+05	4.24850E+00	5.12933E-02
85	1.90000E+05	4.29979E+00	1.71850E-01
86	1.60000E+05	4.47164E+00	6.45385E-02
87	1.50000E+05	4.53618E+00	6.89929E-02
88	1.40000E+05	4.60517E+00	1.54151E-01
89	1.20000E+05	4.75932E+00	1.82322E-01
90	1.00000E+05	4.94164E+00	1.05361E-01
91	9.00000E+04	5.04700E+00	1.17783E-01
92	8.00000E+04	5.16479E+00	6.45385E-02
93	7.50000E+04	5.22932E+00	1.43101E-01
94	6.50000E+04	5.37243E+00	8.00428E-02
95	6.00000E+04	5.45247E+00	8.70114E-02
96	5.50000E+04	5.53948E+00	2.00671E-01
97	4.50000E+04	5.74015E+00	1.17783E-01
98	4.00000E+04	5.85793E+00	1.33531E-01
99	3.50000E+04	5.99146E+00	1.54151E-01
100	3.00000E+04	6.14561E+00	4.05465E-01
101	2.00000E+04	6.55108E+00	2.87682E-01
102	1.50000E+04	6.83876E+00	4.05465E-01
103	1.00000E+04	7.24423E+00	6.93147E-01
104	5.00000E+03	7.93738E+00	1.60944E+00
105	1.00000E+03	9.54681E+00	0.0

Table 2.2 Identification number of nuclides in the 125-group JENDL-3 library

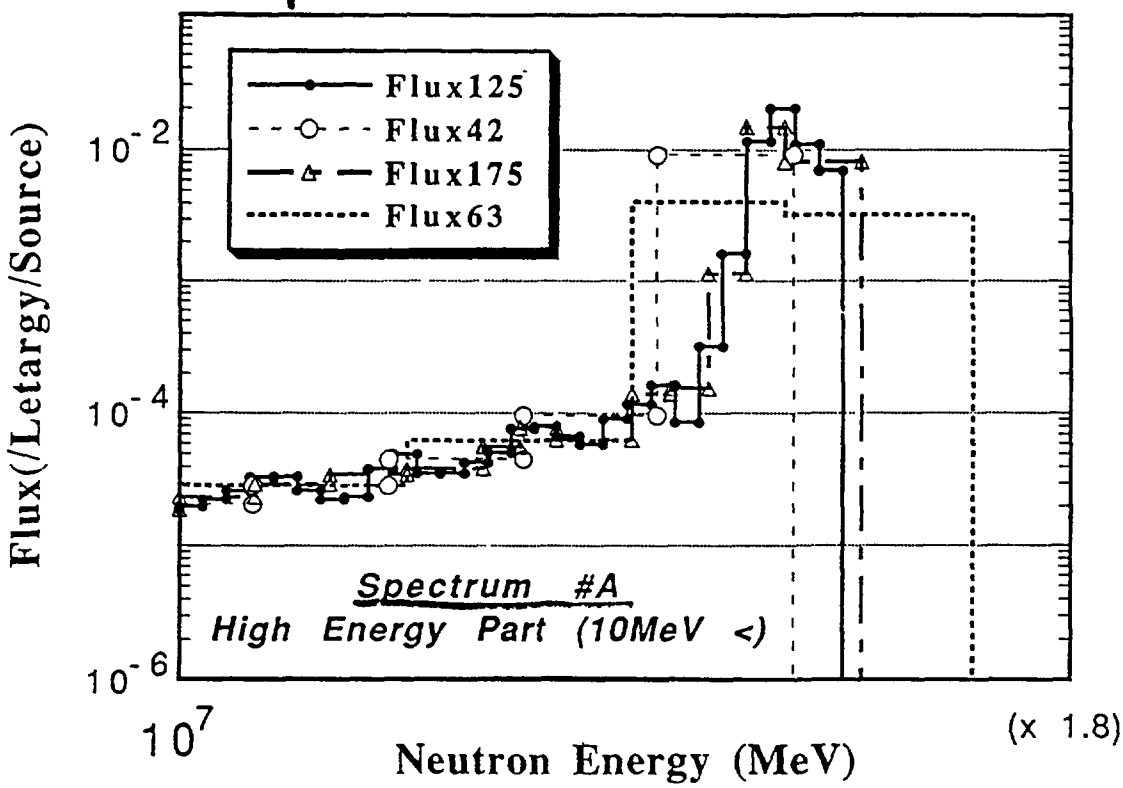
No.	Nuclide	ID number		
		JENDL-3	JENDL-3PR1	ENDF/B4
1	H	3011		1269
2	^{6}Li	3031	306	1271
3	^{7}Li	3032	307	1272
4	^{12}C	3061	612	1274
5	^{9}B	3071	419	1289
6	^{14}N	3041		1275
7	^{16}O	3081	816	1276
8	^{23}Na	3111		1156
9	Mg	3120		1280
10	^{27}Al	3130		1193
11	Si	3140		1194
12	K	3190		1150
13	Ca	3200		1195
14	Cr	3240	2400	1191
15	Fe	3260	2600	1192
16	Ni	3280	2800	1190
17	^{55}Mn	3251		1197
18	Cu	3290		1295
19	Mo	3420		1287
20	Pb	3820		1288

DDL-J3

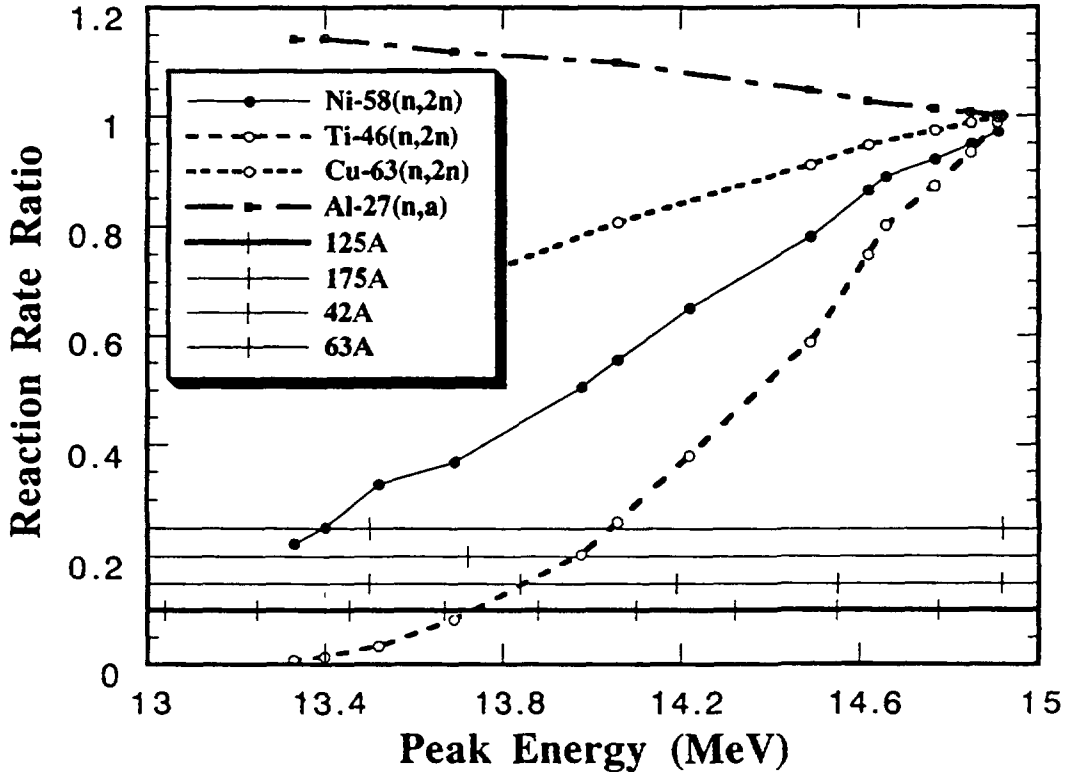
Type : DDX format
Processing Code : PROF-DD
Weighting Function : 1/E & Maxwell for thermal group
Angle : 20 ($\Delta\mu$ = 0.1)
Temperature : 300 K
Error of Point C. S. : 0.1 %

* Group Structure

Spectra in different energy structures



Sensitivity of Mean Neutron Peak Energy on Reaction rates



Associated Libraries based on JENDL-3.1

KERMA Library for FUSION-J3 / FUSION-40

* In order to avoid the probrem in the energy balance method, neutron KERMA factors were calculated by the direct method using the nuclear data in JENDL-3.1.

* Gamma-ray KERMA factors were obtained by the energy balance method, because there was no problem in calculating gamma-ray KERMA factor.

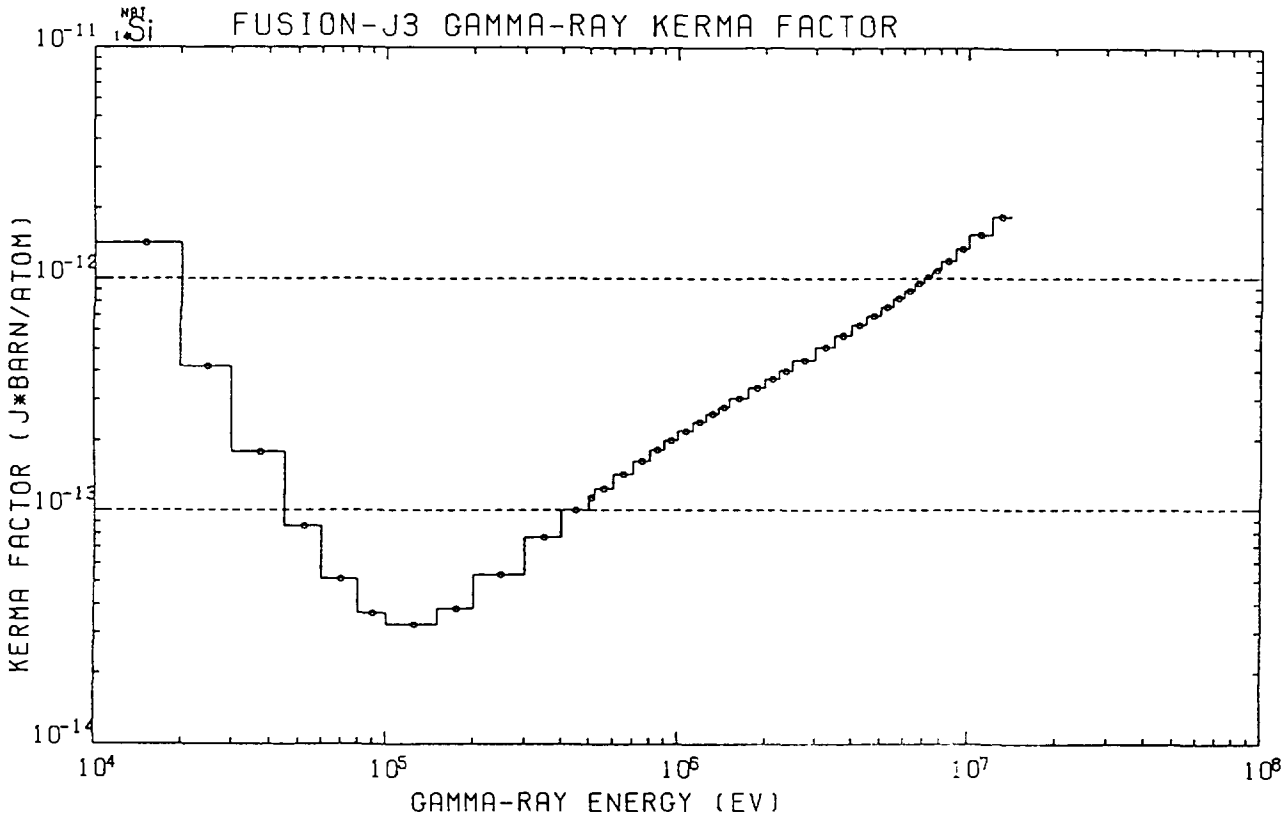
Reference

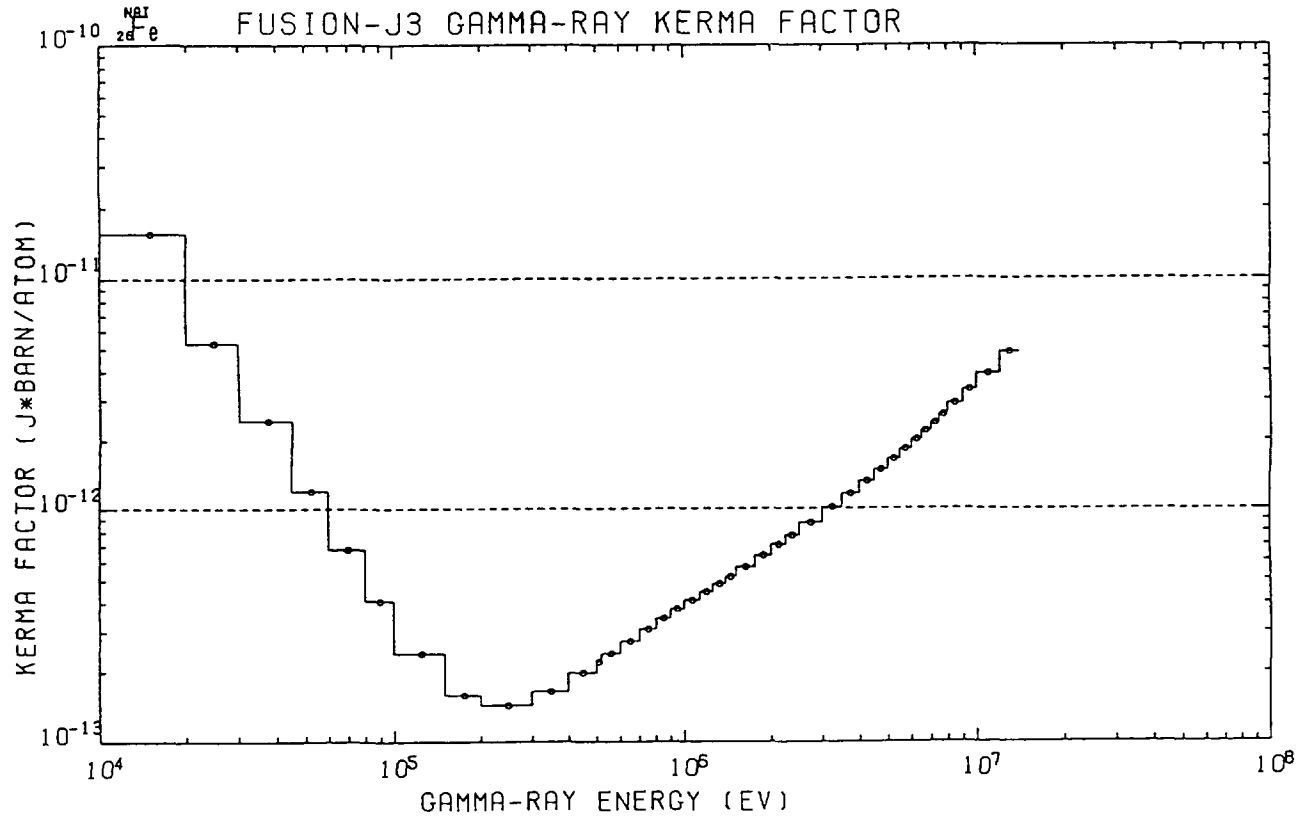
* MAKI K., et al.: "Nuclear Heating Constant KERMA Library -- Nuclear Heating Constant Library for Fusion Nuclear Group Constant Set FUSION-J3 --," JAERI-M 91-073 (1991 May).

Table 4.1 Forty nuclide identification number available in KERMA Library for FUSION-J3.

No	Nuclide	comment	No	Nuclide	comment
1	^{1}H	○	21	Ca	◎
2	^{2}D	◎	22	Ti	◎
3	^{3}He	◎	23	^{51}V	◎
4	^{4}He	◎	24	Cr	◎
5	^{6}Li	○	25	^{55}Mn	◎
6	^{7}Li	○	26	Fe	◎
7	^{9}Be	○	27	Co	◎
8	^{10}B	○	28	Ni	◎
9	^{11}B	◎	29	Cu	◎
10	^{12}C	○	30	Zr	◎
11	^{14}N	◎	31	^{93}Nb	◎
12	^{16}O	◎	32	Mo	◎
13	^{19}F	◎	33	Cd	◎
14	^{23}Na	◎	34	W	◎
15	Mg	◎	35	Pb	◎
16	^{27}Al	◎	36	^{209}Bi	◎
17	Si	◎	37	^{232}T	◎
18	^{31}P	◎	38	^{235}U	◎
19	S	◎	39	^{238}U	◎
20	K	◎	40	^{239}Pu	◎

◎ : Produced by direct method. ○ : Produced by energy balance method.





JAERI-M 91-073

FSXLIB-J3

Type : continuous cross section table
Processing Code : Modified NJOY-83/6 & macros
Directory File : FSXDIR
No. of Elements : 44
Temperature : 300 K

*The processed data were confirmed by MACROS code with the original data in JENDL-3.
*The same type library based on ENDF/B-IV is also available.

Module	Processing Parameters	Selected Condition
RECONR	precision to reconstruct point-wise cross sections	0.5 %
BROADR	precision to reconstruct with Doppler broading	0.5 %
	upper-limit energy of Doppler broading	upper-limit of resonance region or lowest threshold energy of inelastic scat-tering reactions
	processing temperature of cross sections	300 K
THERMR	upper-limit energy of thermal region	4.6 eV
	inelastic option of thermal region	free gas model
GROUPR	multigroup photon production cross sections	neutron 30-group, photon 94-group, 1/E weight, P_L=5
ACER	number of equal-probability angular intervals	32

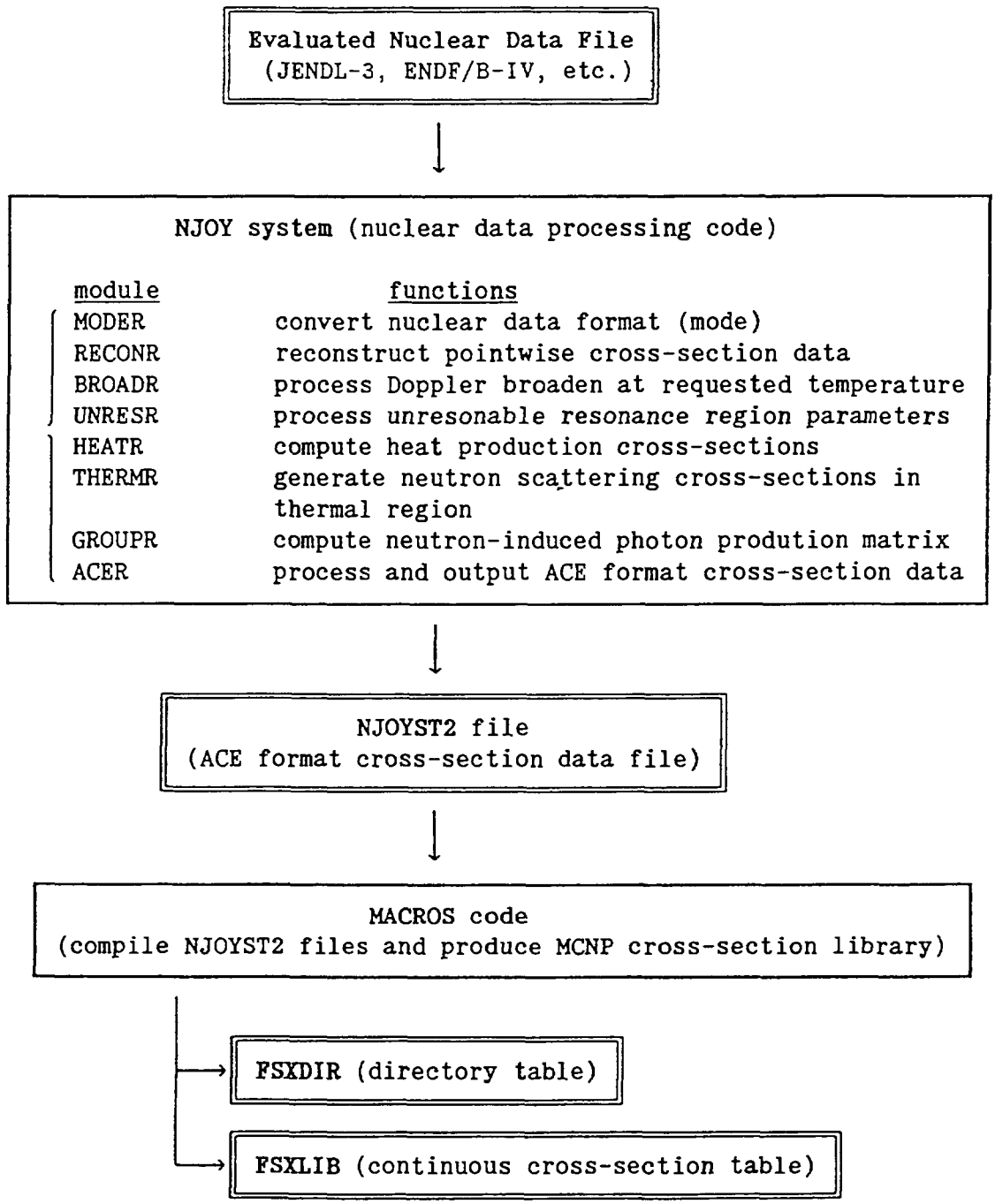


Fig. 3.1 Processing flow of an energy continuous cross-section library for the MCNP code at JAERI

Table 3.1 FSXLIB-J3: Cross sections contained in the MCNP continuous-energy cross section library based on JENDL-3

description of labels in table

ZAID : the nuclide identification number
FILE : the name of MCNP cross section libraries
SOURCE : the name of evaluated nuclear data file
"JENDL-3" is the Japanese Evaluated Nuclear Data Library, version 3.
MAT : the material identifier for a particular evaluation
TYPE : the type of a nuclear data table
"CONT" indicates a continuous-energy cross section set.
TEMP : the temperature (in Kelvin) at which the nuclear data were processed
GPD : the secondary gamma-ray production data
"YES" means that the pointwise energy-dependent gamma production cross section exist.
"NO" means that the gamma production cross section do not exist.
LENGTH : the total length of a particular nuclide in decimal
NUBAR : for fissionable material, NUBAR indicates the type of fission nubar data available
"PROMPT" means that only prompt nubar data are given.
"TOTAL" means that only total nubar data are given.
"BOTH" means that prompt and total nubar are given.
Standard : Does this nuclide include in the standard library FSXLIB?
"YES" means that nuclide includes in the standard and expansion FSXLIB.
"NO" means that nuclide includes only in the expansion FSXLIB.

nuclide [ZAID]	FILE	SOURCE	MAT	TYPE	TEMP (K)	GPD	LENGTH	NUBAR	Standard
(Hydrogen)									
1001.34C	FSXLIBJ3	JENDL-3	3011	CONT	300	YES	2422		YES
1002.34C	FSXLIBJ3	JENDL-3	3012	CONT	300	NO	6376		YES
(Helium)									
2003.34C	FSXLIBJ3	JENDL-3	3021	CONT	300	NO	2053		YES
2004.34C	FSXLIBJ3	JENDL-3	3022	CONT	300	NO	3319		YES
(Lithium)									
3006.34C	FSXLIBJ3	JENDL-3	3031	CONT	300	YES	11637		YES
3007.34C	FSXLIBJ3	JENDL-3	3032	CONT	300	YES	19526		YES
(Beryllium)									
4009.34C	FSXLIBJ3	JENDL-3	3041	CONT	300	YES	18522		YES
(Boron)									
5010.34C	FSXLIBJ3	JENDL-3	3051	CONT	300	YES	29462		YES
5011.34C	FSXLIBJ3	JENDL-3	3052	CONT	300	YES	42854		YES

Tabel 3.1 (continued)

nuclide [ZAID]	FILE	SOURCE	MAT	TYPE	TEMP (K)	GPD	LENGTH	NUBAR	Standard
(Carbon)									
6012.34C	FSXLIBJ3	JENDL-3	3061	CONT	300	YES	17833		YES
(Nitrogen)									
7014.34C	FSXLIBJ3	JENDL-3	3071	CONT	300	YES	37816		YES
7015.34C	FSXLIBJ3	JENDL-3	3072	CONT	300	YES	29147		YES
(Oxygen)									
8016.34C	FSXLIBJ3	JENDL-3	3081	CONT	300	YES	43263		YES
(Fluorine)									
9019.34C	FSXLIBJ3	JENDL-3	3091	CONT	300	NO	14188		YES
(Sodium)									
11023.34C	FSXLIBJ3	JENDL-3	3111	CONT	300	YES	43710		YES
(Magnesium)									
12000.34C	FSXLIBJ3	JENDL-3	3120	CONT	300	YES	44552		YES
12024.34C	FSXLIBJ3	JENDL-3	3121	CONT	300	NO	11943		NO
12025.34C	FSXLIBJ3	JENDL-3	3122	CONT	300	NO	16032		NO
12026.34C	FSXLIBJ3	JENDL-3	3123	CONT	300	NO	12175		NO
(Aluminum)									
13027.34C	FSXLIBJ3	JENDL-3	3131	CONT	300	YES	41884		YES
(Silicon)									
14000.34C	FSXLIBJ3	JENDL-3	3140	CONT	300	YES	86018		YES
14028.34C	FSXLIBJ3	JENDL-3	3141	CONT	300	YES	41020		NO
14029.34C	FSXLIBJ3	JENDL-3	3142	CONT	300	YES	37263		NO
14030.34C	FSXLIBJ3	JENDL-3	3143	CONT	300	YES	31540		NO
(Phosphorus)									
15031.34C	FSXLIBJ3	JENDL-3	3151	CONT	300	NO	21030		YES
(Sulfur)									
16000.33C	FSXLIBJ3	JENDL-3	3160	CONT	300	NO	67590		YES
(Potassium)									
19000.34C	FSXLIBJ3	JENDL-3	3190	CONT	300	NO	53108		YES
19039.34C	FSXLIBJ3	JENDL-3	3191	CONT	300	NO	28410		NO
19040.34C	FSXLIBJ3	JENDL-3	3192	CONT	300	NO	8822		NO
19041.34C	FSXLIBJ3	JENDL-3	3193	CONT	300	NO	23107		NO

Tabel 3.1 (continued)

nuclide [ZAID]	FILE	SOURCE	MAT	TYPE	TEMP (K)	GPD	LENGTH	NUBAR	Standard
(Calcium)									
20000.34C	FSXLIBJ3	JENDL-3	3200	CONT	300	YES	87797		YES
20040.34C	FSXLIBJ3	JENDL-3	3201	CONT	300	YES	54170		NO
20042.34C	FSXLIBJ3	JENDL-3	3202	CONT	300	NO	31071		NO
20043.34C	FSXLIBJ3	JENDL-3	3203	CONT	300	NO	24682		NO
20044.34C	FSXLIBJ3	JENDL-3	3204	CONT	300	NO	26534		NO
20046.34C	FSXLIBJ3	JENDL-3	3205	CONT	300	NO	3532		NO
20048.34C	FSXLIBJ3	JENDL-3	3206	CONT	300	NO	13615		NO
(Scandium)									
21045.34C	FSXLIBJ3	JENDL-3	3211	CONT	300	YES	79926		YES
(Titanium)									
22000.34C	FSXLIBJ3	JENDL-3	3220	CONT	300	YES	64044		YES
22046.34C	FSXLIBJ3	JENDL-3	3221	CONT	300	NO	29710		NO
22047.34C	FSXLIBJ3	JENDL-3	3222	CONT	300	NO	29725		NO
22048.34C	FSXLIBJ3	JENDL-3	3223	CONT	300	NO	20242		NO
22049.34C	FSXLIBJ3	JENDL-3	3224	CONT	300	NO	20441		NO
22050.34C	FSXLIBJ3	JENDL-3	3225	CONT	300	NO	16522		NO
(Vanadium)									
23051.34C	FSXLIBJ3	JENDL-3	3231	CONT	300	NO	39191		YES
(Chromium)									
24000.34C	FSXLIBJ3	JENDL-3	3240	CONT	300	YES	117591		YES
24050.34C	FSXLIBJ3	JENDL-3	3241	CONT	300	NO	47312		NO
24052.34C	FSXLIBJ3	JENDL-3	3242	CONT	300	NO	39983		NO
24053.34C	FSXLIBJ3	JENDL-3	3243	CONT	300	NO	31758		NO
24054.34C	FSXLIBJ3	JENDL-3	3244	CONT	300	NO	24481		NO
(Manganese)									
25055.34C	FSXLIBJ3	JENDL-3	3251	CONT	300	YES	201069		YES
(Iron)									
26000.34C	FSXLIBJ3	JENDL-3	3260	CONT	300	YES	144778		YES
26054.34C	FSXLIBJ3	JENDL-3	3261	CONT	300	YES	62131		NO
26056.34C	FSXLIBJ3	JENDL-3	3262	CONT	300	YES	72810		NO
26057.34C	FSXLIBJ3	JENDL-3	3263	CONT	300	YES	60593		NO
26058.34C	FSXLIBJ3	JENDL-3	3264	CONT	300	YES	67076		NO
(Cobalt)									
27059.34C	FSXLIBJ3	JENDL-3	3271	CONT	300	NO	79549		YES

Tabel 3.1 (continued)

nuclide [ZAID]	FILE	SOURCE	MAT	TYPE	TEMP (K)	GPD	LENGTH	NUBAR	Standard
(Nickel)									
28000.34C	FSXLIBJ3	JENDL-3	3280	CONT	300	YES	237991		YES
28058.34C	FSXLIBJ3	JENDL-3	3281	CONT	300	YES	82463		NO
28060.34C	FSXLIBJ3	JENDL-3	3282	CONT	300	YES	103280		NO
28061.34C	FSXLIBJ3	JENDL-3	3283	CONT	300	NO	44481		NO
28062.34C	FSXLIBJ3	JENDL-3	3284	CONT	300	NO	42650		NO
28064.34C	FSXLIBJ3	JENDL-3	3285	CONT	300	NO	41593		NO
(Copper)									
29000.34C	FSXLIBJ3	JENDL-3	3290	CONT	300	YES	170023		YES
29063.34C	FSXLIBJ3	JENDL-3	3291	CONT	300	YES	144647		NO
29065.34C	FSXLIBJ3	JENDL-3	3292	CONT	300	YES	115286		NO
(Zirconium)									
40000.34C	FSXLIBJ3	JENDL-3	3400	CONT	300	YES	160023		YES
40090.34C	FSXLIBJ3	JENDL-3	3401	CONT	300	NO	53750		NO
40091.34C	FSXLIBJ3	JENDL-3	3402	CONT	300	NO	67096		NO
40092.34C	FSXLIBJ3	JENDL-3	3403	CONT	300	NO	53115		NO
40094.34C	FSXLIBJ3	JENDL-3	3405	CONT	300	NO	44262		NO
40096.34C	FSXLIBJ3	JENDL-3	3407	CONT	300	NO	30050		NO
(Niobium)									
41093.34C	FSXLIBJ3	JENDL-3	3411	CONT	300	YES	119152		YES
(Molybdenum)									
42000.34C	FSXLIBJ3	JENDL-3	3420	CONT	300	YES	165448		YES
42092.34C	FSXLIBJ3	JENDL-3	3421	CONT	300	NO	48160		NO
42094.34C	FSXLIBJ3	JENDL-3	3422	CONT	300	NO	41872		NO
42095.34C	FSXLIBJ3	JENDL-3	3423	CONT	300	NO	39992		NO
42096.34C	FSXLIBJ3	JENDL-3	3424	CONT	300	NO	46780		NO
42097.34C	FSXLIBJ3	JENDL-3	3425	CONT	300	NO	42827		NO
42098.34C	FSXLIBJ3	JENDL-3	3426	CONT	300	NO	65873		NO
42100.34C	FSXLIBJ3	JENDL-3	3428	CONT	300	NO	67388		NO
(Silver)									
47000.34C	FSXLIBJ3	JENDL-3	3470	CONT	300	YES	183283		NO
(Cadmium)									
48000.34C	FSXLIBJ3	JENDL-3	3480	CONT	300	YES	153335		YES
(Antimony)									
51000.34C	FSXLIBJ3	JENDL-3	3510	CONT	300	NO	76427		NO
(Europium)									
63000.34C	FSXLIBJ3	JENDL-3	3630	CONT	300	YES	56669		NO
63151.34C	FSXLIBJ3	JENDL-3	3631	CONT	300	NO	42862		NO
63153.34C	FSXLIBJ3	JENDL-3	3633	CONT	300	NO	39469		NO

Tabel 3.1 (continued)

nuclide [ZAID]	FILE	SOURCE	MAT	TYPE	TEMP (K)	GPD	LENGTH	NUBAR	Standard
(Hafnium)									
72000.34C	FSXLIBJ3	JENDL-3	3720	CONT	300	YES	85740		YES
(Tantalum)									
73181.34C	FSXLIBJ3	JENDL-3	3731	CONT	300	YES	166522		YES
(Tungsten)									
74000.34C	FSXLIBJ3	JENDL-3	3740	CONT	300	YES	199104		YES
74182.34C	FSXLIBJ3	JENDL-3	3741	CONT	300	NO	105858		NO
74183.34C	FSXLIBJ3	JENDL-3	3742	CONT	300	NO	67010		NO
74184.34C	FSXLIBJ3	JENDL-3	3743	CONT	300	NO	95032		NO
74186.34C	FSXLIBJ3	JENDL-3	3744	CONT	300	NO	86861		NO
(Lead)									
82000.34C	FSXLIBJ3	JENDL-3	3820	CONT	300	YES	137640		YES
82204.34C	FSXLIBJ3	JENDL-3	3821	CONT	300	YES	55113		NO
82206.34C	FSXLIBJ3	JENDL-3	3822	CONT	300	YES	107986		NO
82207.34C	FSXLIBJ3	JENDL-3	3823	CONT	300	YES	71243		NO
82208.34C	FSXLIBJ3	JENDL-3	3824	CONT	300	YES	38091		NO
(Bismuth)									
83209.34C	FSXLIBJ3	JENDL-3	3831	CONT	300	YES	70146		YES
(Thorium)									
90232.34C	FSXLIBJ3	JENDL-3	3905	CONT	300	NO	96180	BOTH	YES
(Protactinium)									
91233.34C	FSXLIBJ3	JENDL-3	3913	CONT	300	NO	13592	BOTH	NO
(Uranium)									
92233.34C	FSXLIBJ3	JENDL-3	3922	CONT	300	NO	33892	BOTH	NO
92234.34C	FSXLIBJ3	JENDL-3	3923	CONT	300	NO	80600	TOTAL	NO
92235.34C	FSXLIBJ3	JENDL-3	3924	CONT	300	YES	62587	BOTH	YES
92236.34C	FSXLIBJ3	JENDL-3	3925	CONT	300	NO	77518	TOTAL	NO
92238.34C	FSXLIBJ3	JENDL-3	3926	CONT	300	YES	235117	BOTH	YES
(Neptunium)									
93237.34C	FSXLIBJ3	JENDL-3	3931	CONT	300	NO	47171	BOTH	YES
(Plutonium)									
94239.34C	FSXLIBJ3	JENDL-3	3943	CONT	300	YES	128048	BOTH	YES
94240.34C	FSXLIBJ3	JENDL-3	3944	CONT	300	NO	134534	BOTH	YES
94241.34C	FSXLIBJ3	JENDL-3	3945	CONT	300	NO	25107	BOTH	NO
94242.34C	FSXLIBJ3	JENDL-3	3946	CONT	300	NO	58687	TOTAL	NO

Tabel 3.1 (continued)

nuclide [ZAID]	FILE	SOURCE	MAT	TYPE	TEMP (K)	GPD	LENGTH	NUBAR	Standard
(Americium)									
95241.34C	FSXLIBJ3	JENDL-3	3951	CONT	300	NO	50430	BOTH	NO
95242.34C	FSXLIBJ3	JENDL-3	3952	CONT	300	NO	13996	BOTH	NO
95243.34C	FSXLIBJ3	JENDL-3	3954	CONT	300	NO	57829	BOTH	NO
(Californium)									
98252.34C	FSXLIBJ3	JENDL-3	3984	CONT	300	NO	32666	BOTH	NO

Cross section Libraries Based on JENDL-3.2

JENDL-3.1	JENDL-3.2	
FUSION-J3	FUSION-J3.2*	
FUSION-40	FUSION-40/J3.2*	
JSSTDL	JSSTDL-J3.2	
DDL-J3	DDL-J3.2	
FSXLIB-J3	FSXLIB-J3.2	for MCNP
MVPLIB-J3	MVPLIB-J3.2	for MVP

*Colapsed from JSSTDL-J3.2 including F-table.

Cross Section Libraries for Fission Reactor Design Based on JENDL-3,1

Name	nuclides	F. P.	Remarks
SRACLIB-J3	59	190	prepared for SRAC code system and for thermal reactor analysis
JFS-3-J3	59	190	for fast reactor analysis EXPANDA-G-S SLAROM-3

See Appendix 1 and Appendix 2.

Appendix 1

THERMAL REACTOR GROUP CROSS SECTION LIBRARY, SRACLIB-J3 and SRACLIB-J3.2

Hideki TAKANO

Reactor System Laboratory, Depertment of Reactor Engneering, JAERI

Abstract

1. **NAME OF DESIGNATION OF LIBRARY - SRACLIB-J3, SRACLIB-J3.2**
2. **COMPUTER FOR WHICH LIBRARY IS DESIGNED AND OTHER MACHINE VERSION PACKAGES AVAILABLE -**

Library-name	Package-ID	Orig. Computer	Test Computer
SRACLIB-J3		FACOM VP-2000, FACOM M-780	
SRACLIB-J3.2			

3. **DESCRIPTION OF LIBRARY OF FUSION -**

This library is designed so as to be used for JAERI thermal reactor analysis and design code system, SRAC. This library contains the 74-group constants for fast energy region from 0.414 eV to 10 MeV and 48-group ones for thermal region below 3.93 eV. The following 59 nuclides are included:

H-1, He-4, Be-9, B-10, B-11, C-12, N-14, O-16, Na-23, Al-27, Si, Ar, Ti, V, Cr, Mn-55, Fe, Ni, Cu, Zr, Nb-93, Mo, Eu-151, Eu-153, Gd, Gd-155, Gd-156, Gd-157, Gd-158, Gd-160, Ta-181, W, Th-228, Th-230, Th-232, Th-233, Th-234, Pa-233, U-233, U-234, U-235, U-236, U-238, Np-237, Np-239, Pu-236, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-242m, Am242g, Am-243, Gm-242, Cm-243, Cm-244, Cm-245.

For high burnup reactor calculation, group cross sections of 65 explicit fission product nuclides and a lumped fission nuclide have been generated by using 190-FP nuclides of JENDL-3.1 and JENDL-3.2.

In resonance energy region, ultra-fine enrgy group cross section library of 4000 points for primary important heavy resonant nuclides are prepared.

4. METHOD OF SOLUTION -

Group constants are generated with the data processing code system TIMS-PGG, MCROSS-2, GASKET and HEXSCAT.

The group constants in the unresolved resonance region are produced on the basis of random sampling resonance generation method. The ultra-fine group calculation method is used for resonance region.

The resonance shielding factors are tabulated for 8 background potential scattering cross sections (0, 1, 10, 100, 1000, 10000, 100000 and 1000000 barns), 4 temperatures (300, 800, 2100 and 4500) and 4 resonance interference parameters.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROGRAM -

6. TYPICAL RUNNING TIME -

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS -

9. STATUS -

10. REFERENCES -

Hideki TAKANO, Hiroshi AKIE and Kunio KANEKO
"Production and Benchmark Tests of Thermal Reactor Group Cross Section Library SRACLIB-J3"
To be published in JAERI-M report

11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED -

13. OPERATING SYSTEM UNDER WHICH PROGRAM IS EXECUTED -

14. OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -

Hideki TAKANO, Hiroshi AKIE and Kunio KANEKO
Department of Reactor Engineering
JAERI
Japan

16. MATERIAL AVAILABLE -

17. CATEGORY -

Appendix 2

FAST REACTOR GROUP CONSTANTS SET, JFT-3-J3 and JFS-3-J3.2

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Reactor System Laboratory, Depertment of Reactor Engneering, JAERI

Abstract

1. **NAME OF DESIGNATION OF LIBRARY - SRACLIB-J3, JFS-3-J3.2**
2. **COMPUTER FOR WHICH LIBRARY IS DESIGNED AND OTHER MACHINE VERSION PACKAGES AVAILABLE -**

Library-name	Package-ID	Orig. Computer	Test Computer
JFS-3-J3		FACOM VP-2000, M-780	
JFS-3-J3.2			

3. **DESCRIPTION OF LIBRARY OF FUSION -**

This library is designed so as to be used for JAERI fast reactor analysis and design code system. This library contains the 70-group constants with quater-lethargy width for the following 59 nuclides and 12 lumped fission products :

H-1, He-4, Be-9, B-10, B-11, C-12, N-14, O-16, Na-23, Al-27, Si, Ar, Ti, V, Cr, Mn-55, Fe, Ni, Cu, Zr, Nb-93, Mo, Eu-151, Eu-153, Gd, Gd-155, Gd-156, Gd-157, Gd-158, Gd-160, Ta-181, W, Th-228, Th-230, Th-232, Th-233, Th-234, Pa-233, U-233, U-234, U-235, U-236, U-238, Np-237, Np-239, Pu-236, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241,Am-242m, Am-242g, Am-243, Gm-242, Cm-243, Cm-244, Cm-245, and 12 LFPs for 4 mother nuclides (U-235, U-238, Pu-239 and Pu-241) and 3 burnup days (180, 1080 and 1800).

The group contants for minor actinides such as Np, Am and Cm have been produced so as to be able to use for TRU-transmutation calculations.

For high burnup reactor calculation, burnup depending lumped FP group cross sections for mother fission isotopes of U-235, U-238, Pu-239 and Pu-241 have been generated by using 190-FP nuclides of JENDL-3.

4. **METHOD OF SOLUTION -**

Group constants are generated with the data processing code system TIMS-PGG. In this code, the collision denity spectrum for a typical large LMFBR core spectrum is used as the weithing function. The group constants in the unresolved resonance region are produced

on the basis of random sampling resonance generation method. The ultra-fine group calculation method is used for resonance region.

The resonance shielding factors are tabulated for 8 background potential scattering cross sections (0, 1, 10, 100, 1000, 10000, 100000 and 1000000 barns), 4 temperatures (300, 800, 2100 and 4500) and 4 resonance interference parameters.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROGRAM -

6. TYPICAL RUNNING TIME -

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS -

JFS3CARD : JFS-3 user library is transfered from binary to card mode.

JFS3BINA : JFS-3 user library is transfered from card to binary mode.

DUMP70 : Table of group constants are printed out.

EXPANDA-G-S : One-dimensional diffusion code uses the JFS-3-J3 and JFS-3-J3.2 libraries.

SLAROM-3 : Heterogeneous cell calculation code uses the JFS-3-J3 and JFS-3-J3 libraries.

9. STATUS -

10. REFERENCES -

Hideki TAKANO and Kunio KANEKO

"Production and Benchmark Tests of Fast Reactor Group Constant Set JFS-3-J3"

To be published in JAERI-M report

11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED -

13. OPERATING SYSTEM UNDER WHICH PROGRAM IS EXECUTED -

14. OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -

Hideki TAKOANO, Hiroshi AKIE and Kunio KANEKO

Department of Reactor Engineering

JAERI

Japan

16. MATERIAL AVAILABLE -

17. CATEGORY.

AMPX77 Modules for Checking and Validating Processed Multigroup Cross Sections

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IAEA Consultants Meeting
Preparation of Processed Nuclear Data Libraries for Thermal, Fast and Fusion Research and Power Reactor Applications
December 8-10, 1993
Vienna, Austria

Abstract

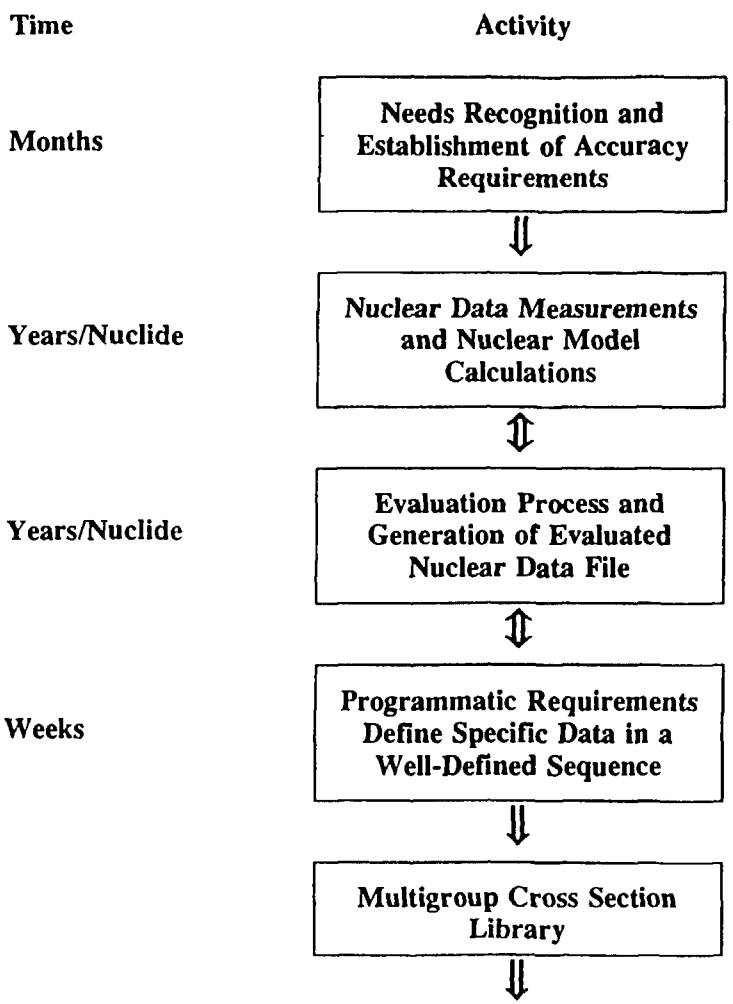
Cross Section data required for applications usually evolve from a national effort to provide state-of-the-art data based on established needs and uncertainties. The process rquires an iterative sequence of events to yield a quality assured transport cross-section library. This paper outlines the general approach used to develop the VITAMIN libraries in Oak Ridge. The use of RADE in checking multigroup data is illustrated. The use of benchmark calculations used to test the performance of the multigroup data and preparation of data for distribution are discussed.

Overview of Presentation

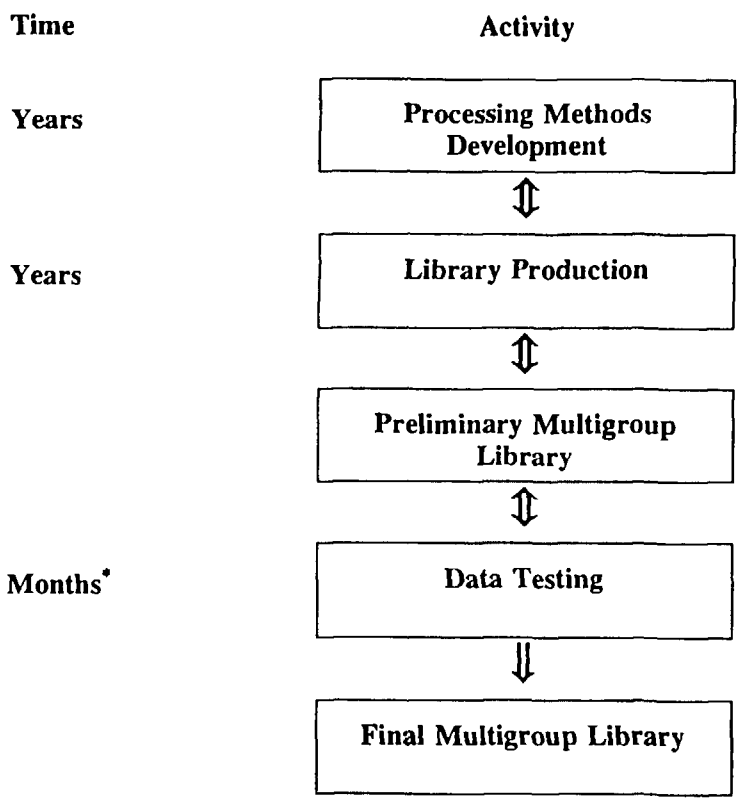
- Outline the general approach used to develop the VITAMIN libraries in Oak Ridge
- Illustrate the use of RADE in checking multigroup data
- Benchmark calculations used to test the performance of the multigroup data
- Preparation of data for distribution

NUCLEAR DATA FOR RADIATION TRANSPORT (CROSS SECTIONS)

Cross Section data required for applications usually evolve from a national effort to provide state-of-the-art data based on established needs and uncertainties. The process requires an iterative sequence of events to yield a quality assured transport cross-section library.



11/93:afr



*Comprehensive efforts require years.

Critical Elements of the Production Process

STAGE 1: Generate pointwise data

STAGE 2: Set-up calculational procedure and input data for the production task

STAGE 3: Generate multigroup data

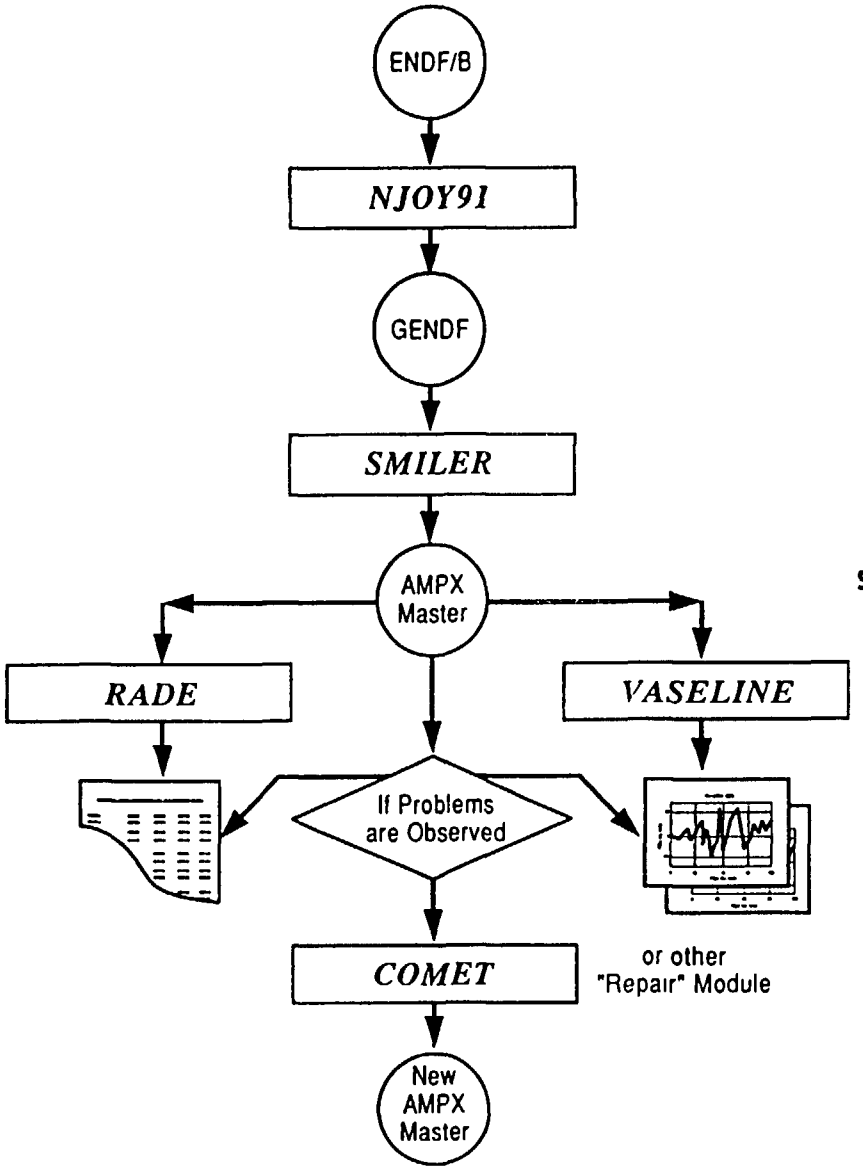
STAGE 4: Preliminary testing of multigroup data

STAGE 5: Build library and perform validation studies

STAGE 6: Re-process library

STAGE 7: Preparation of data for distribution

ORNL-DWG 91Z-14372A



Procedure used to generate multigroup cross sections

Stage 1: Generate Pointwise Data

- Obtain evaluated data from NNDC
- Independently run checking codes (CHECKER, FIZCON) to resolve data discrepancies
- Reconstruct pointwise data from resonance parameters using NJOY system
- For known troublesome nuclides, make spot comparisons of total, elastic, fission, and capture reactions with point data from independent source (e.g., RECENT)
- If major discrepancy is observed, expand investigation to include other independent methods (e.g., SAMMY)

Stage 1: Generate Pointwise Data (continued)

- Make visual and automated examination of output (resolve warning messages and other problem indicators)
- Check continuity at resonance region breakpoints
- Look for non-physical indicators
  - negative cross sections
  - zero cross section values
  - check energy mesh to ensure that it represents an increasing function of energy

```
1**************************************************************************
 *                                                *                       *
 *   $$   $$        $$   $$$$$   $$    $$        *                       *
 *   $$$  $$        $$  $$$$$$$   $$  $$         *  nuclear              *  vers. 91.94  *
 *   $$$$ $$        $$  $$   $$    $$$$          *  cross section        *  ran at lanl  *
 *   $$ $$$$  $$   $$  $$   $$     $$           *  processing           *  on mach r    *
 *   $$  $$$  $$$$$$$  $$$$$$$     $$           *  system               *  on 09/24/93  *
 *   $$   $$   $$$$$    $$$$$      $$           *                       *  at 09:33:02  *
 *                                                *                       *
 **************************************************************************

reconr...reconstruct pointwise cross sections in pendf format                0.0s

unit for endf/b tape .................          21
unit for pendf tape ..................          22

label for pendf tape
----------------------------------------
 pendf tape for fe-56 mat 2631 mod1 evaluation

tape label
----------------------------------------
 Fe-56 (neutron) revision 1 from ENDF/B-VI tape 123

                                                              storage  15/100000

material to be processed .............       2631
reconstruction tolerance .............      0.002
reconstruction temperature ...........        0.k
no. significant figures ..............          7
resonance-integral-check tolerance ...      0.040
max resonance-integral error .........  2.000E-07

descriptive cards for pendf tape
----------------------------------------
  fe-56, mod1 evaluation
  processed by the njoy system
  r.q. wright,  9-17-93


processing mat   2631
----------------------------------------
  26-FE- 56 ORNL       EVAL-MAR89 FU, C.PEREY, HETRICK, F.PEREY


---message from rdfil2---calculation of angular distribution not installed.

mat has no unresolved resonance parameters

points in initial unionized grid = 3470
points added by linearization = 225                                          4.0s

             estimated maximum error due to
             resonance integral check (errmax,errint)
             and significant figure truncation (ndigit)

   upper      elastic    percent error    capture    percent error
   energy     integral  res-int sig-fig   integral  res-int sig-fig
 1.00E-05
 3.32E-05    1.44E+01   0.000   0.000     1.17E+02   0.000   0.000
 1.12E-04    1.47E+01   0.000   0.000     6.52E+01   0.000   0.000
 4.30E-04    1.62E+01   0.000   0.000     3.81E+01   0.000   0.000
 1.64E-03    1.61E+01   0.000   0.000     1.94E+01   0.000   0.000
 6.33E-03    1.63E+01   0.000   0.000     9.99E+00   0.000   0.000
 2.06E-02    1.42E+01   0.000   0.000     4.60E+00   0.000   0.000
 7.13E-02    1.50E+01   0.000   0.000     2.66E+00   0.000   0.000
 2.53E-01    1.53E+01   0.000   0.000     1.45E+00   0.000   0.000
 8.67E-01    1.48E+01   0.000   0.000     7.52E-01   0.000   0.000
 2.76E+00    1.39E+01   0.000   0.000     3.88E-01   0.000   0.000
 9.00E+00    1.42E+01   0.000   0.000     2.20E-01   0.000   0.000
 2.92E+01    1.41E+01   0.000   0.000     1.21E-01   0.000   0.000
 1.03E+02    1.49E+01   0.000   0.000     6.85E-02   0.000   0.000
 3.41E+02    1.35E+01   0.000   0.000     3.18E-02   0.000   0.000
 1.14E+03    1.21E+01   0.000   0.000     1.48E-02   0.012   0.000
 3.77E+03    9.33E+00   0.000   0.000     1.82E-01   0.003   0.000
 1.38E+04    6.40E+00   0.000   0.000     9.81E-04   0.070   0.000
 4.78E+04    1.12E+01   0.000   0.000     1.18E-02   0.166   0.000
 1.65E+05    5.40E+00   0.002   0.000     1.22E-02   0.212   0.017
```

```
 5.34E+05     4.20E+00    0.010    0.000     6.55E-03    0.236    C.027

points added by resonance reconstruction = 20866
points affected by resonance integral check = 14236
points affected by significant figure reduction =   180
points removed by backthinning =   396
final number of resonance points = 21851

---message from emerge---negative elastic cross sections found.

***number of nonpositive cross sections removed =     1

                                                        usage  27088/100000
                                                                    58.2s
********************************************************************************

broadr...doppler broadening of endf/b data                              58.2s

unit for input pendf tape .....          22
unit for output pendf tape .....         23
material to be processed .....         2631
number of final temperatures .....        4
restart (0 no, 1 yes) .....               0
bootstrap (0 no, 1 yes) .....             1
starting material temperature .....     0.k
thinning tolerance .....              0.002
max. energy .....                  1.00000E+06
errmax for thinning .....          4.00000E-02
errint for thinning .....          2.00000E-07
final temperatures .....           3.00E+02
                                   6.00E+02
                                   1.00E+03
                                   2.10E+03

                                                  storage  10/ 30000

output   0.0000E+00k from restart

max energy for broadening and thinning =   8.50000E+05
                                                                    69.8s
broadened mat2631 from  0.0000E+00 to  3.0000E+02 k
        points in= 24495  points out= 19584
        mt   2 102
                                                                    96.5s
broadened mat2631 from  3.0000E+02 to  6.0000E+02 k
        points in= 19584  points out= 18756
        mt   2 102
                                                                   118.4s
broadened mat2631 from  6.0000E+02 to  1.0000E+03 k
        points in= 18756  points out= 18226
        mt   2 102
                                                                   139.7s
broadened mat2631 from  1.0000E+03 to  2.1000E+03 k
        points in= 18226  points out= 17328
        mt   2 102
                                                                   162.4s
********************************************************************************

heatr...prompt kerma                                                    162.4s

                                                    storage  15/ 12000

input endf/b unit .....                  21
input pendf unit .....                   23
output pendf unit .....                  24
mat to be processed .....              2631
no. temperatures (0=all) .....            0
gamma heat (0 nonlocal, 1 local) .....    0
print option (0 min, 1 more, 2 chk) ..    0
                                                      temp 1      173.3s
                                                      temp 1      191.3s
                                                      temp 2      260.4s
                                                      temp 2      277.8s
                                                      temp 3      345.1s
                                                      temp 3      362.3s
                                                      temp 4      428.2s
                                                      temp 4      444.5s

                                                    usage  17508/ 12000
```

```
                                                                                497.7s
********************************************************************************

thermr...compute thermal scattering cross sections and matrices          497.7s

                                                          storage  10/ 50000

unit for endf/b tape ..............          0
unit for input pendf tape .........         24
unit for output pendf tape ........         25
material to be processed (endf) ...          0
material to be processed (pendf) ..       2631
number of angle bins ..............          8
number of temperatures ............          4
inelastic option ..................          1
elastic option ....................          0
number of principal atoms .........          1
reference mt ......................        221
print option (0 min, 1 max) .......          0
temperatures (kelvin) ............. 3.0000E+02
                                    6.0000E+02
                                    1.0000E+03
                                    2.1000E+03
tolerance ......................... 1.0000E-02
max energy for thermal treatment ..... 5.0435E+00

wrote thermal data for temp= 3.00E+02                                     532.4s

wrote thermal data for temp= 6.00E+02                                     566.3s

wrote thermal data for temp= 1.00E+03                                     599.4s

wrote thermal data for temp= 2.10E+03                                     631.3s

                                                        usage  12665/ 50000
                                                                          631.3s
********************************************************************************
```

```
points added by resonance reconstruction = 20866
points affected by resonance integral check = 14236
points affected by significant figure reduction =   180
points removed by backthinning =   396
final number of resonance points = 21851

---message from emerge---negative elastic cross sections found.

***number of nonpositive cross sections removed =     1
```

```
 1.187000+6 2.229800+0 1.188000+6 2.537100+0 1.188700+6 2.762800+02631 3  2 181
 1.189000+6 2.785300+0 1.189800+6 2.858400+0 1.190000+6 3.069900+02631 3  2 181
 1.190500+6 3.597100+0 1.191000+6 2.447100+0 1.191400+6 1.529300+02631 3  2 181
 1.192000+6 1.138100+0 1.192300+6 9.429600-1 1.193000+6 7.495700-12631 3  2 181
 1.193300+6 6.576900-1 1.193500+6 7.339400-1 1.194000+6 4.224900-12631 3  2 181
 1.194600+6 4.508000-1 1.194800+6 4.155200-1 1.194900+6 5.602000-12631 3  2 181
 1.195000+6 5.207100-1 1.195700+6 2.244900-1 1.196000+6 3.499900-12631 3  2 182
 1.196200+6 1.555300-1 1.196600+6 2.170100-1 1.197000+6 3.926200-22631 3  2 182
 1.197100+6 0.000000+0 1.197300+6 8.537900-2 1.197500+6 3.621100-32631 3  2 182
 1.197600+6 2.256600-1 1.198000+6 4.730800-1 1.198200+6 6.027800-12631 3  2 182
 1.198400+6 8.943000-1 1.198500+6 1.333000+0 1.199000+6 2.529400+02631 3  2 182
 1.199400+6 3.488100+0 1.200000+6 3.361600+0 1.200300+6 3.301800+02631 3  2 182
 1.201000+6 1.833200+0 1.201400+6 1.009200+0 1.202000+6 7.848600-12631 3  2 182
 1.202700+6 1.850400+0 1.203000+6 1.477700+0 1.203200+6 1.219800+02631 3  2 182
 1.204000+6 1.091100+0 1.205000+6 1.270900+0 1.205100+6 1.290500+02631 3  2 182
 1.206000+6 2.858300+0 1.206900+6 6.214400+0 1.207000+6 5.903600+02631 3  2 182
 1.207400+6 4.654100+0 1.208000+6 3.228200+0 1.208100+6 2.991700+02631 3  2 183
 1.209000+6 2.410500+0 1.209100+6 2.351700+0 1.210000+6 3.331600+02631 3  2 183
 1.210200+6 3.553300+0 1.210700+6 4.829500+0 1.211000+6 4.224700+02631 3  2 183
 1.212000+6 2.209400+0 1.212500+6 1.201000+0 1.213000+6 8.500300-12631 3  2 183
 1.213100+6 7.794600-1 1.213500+6 1.092700+0 1.214000+6 1.658900+02631 3  2 183
 1.214200+6 1.882300+0 1.214600+6 1.770800+0 1.214900+6 1.319100+02631 3  2 183
 1.215000+6 1.276500+0 1.215800+6 9.257100-1 1.216000+6 1.035900+02631 3  2 183
:
```

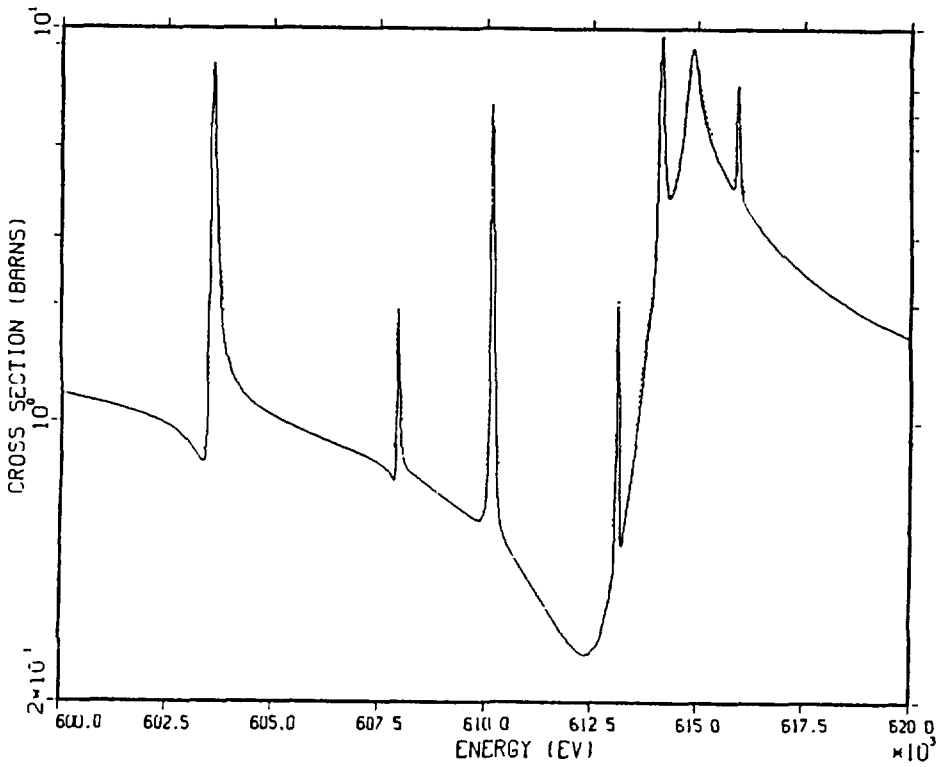
^{56}Fe endf data

```
1.168000+6 5.119300-1 1.168300+6 2.192600-1 1.168500+6 2.177000-12631 3  21250
1.168600+6 9.701600-2 1.169000+6 2.867100-1 1.169200+6 3.795000-12631 3  21250
1.169900+6 1.711700+0 1.170000+6 1.955700+0 1.170200+6 2.447200+02631 3  21250
1.171000+6 1.971800+0 1.171900+6 1.464100+0 1.172000+6 1.487700+02631 3  21250
1.172800+6 1.701100+0 1.173000+6 1.878800+0 1.173700+6 2.502900+02631 3  21250
1.174000+6 2.041300+0 1.174400+6 1.421100+0 1.175000+6 1.222300+02631 3  21250
1.175800+6 9.383500-1 1.176000+6 9.392400-1 1.177000+6 9.309300-12631 3  21250
1.177200+6 9.258600-1 1.177500+6 8.239700-1 1.177700+6 9.156300-12631 3  21251
1.177900+6 6.907800-1 1.178000+6 7.666000-1 1.179000+6 1.520200+02631 3  21251
1.179800+6 3.491900+0 1.180000+6 3.392300+0 1.180400+6 3.198100+02631 3  21251
1.181000+6 4.251200+0 1.181200+6 4.596700+0 1.182000+6 2.709800+02631 3  21251
1.182100+6 2.476600+0 1.183000+6 1.710300+0 1.183500+6 1.920000+02631 3  21251
1.184000+6 2.733400+0 1.184400+6 3.384200+0 1.185000+6 2.153000+02631 3  21251
1.185100+6 1.948100+0 1.185800+6 1.867800+0 1.186000+6 1.929800+02631 3  21251
1.187000+6 2.229800+0 1.188000+6 2.537100+0 1.188700+6 2.762800+02631 3  21251
1.189000+6 2.785300+0 1.189800+6 2.858400+0 1.190000+6 3.069900+02631 3  21251
1.190500+6 3.597100+0 1.191000+6 2.447100+0 1.191400+6 1.529300+02631 3  21251
1.192000+6 1.138100+0 1.192300+6 9.429600-1 1.193000+6 7.495700-12631 3  21252
1.193300+6 6.576900-1 1.193500+6 7.339400-1 1.194000+6 4.224900-12631 3  21252
1.194600+6 4.508000-1 1.194800+6 4.155200-1 1.194900+6 5.602000-12631 3  21252
1.195000+6 5.207100-1 1.195700+6 2.244900-1 1.196000+6 3.499900-12631 3  21252
1.196200+6 1.555300-1 1.196600+6 2.170100-1 1.197000+6 3.926200-22631 3  21252
1.197100+6 1.000000-8 1.197300+6 8.537900-2 1.197500+6 3.621100-32631 3  21252
 :
```


^{56}Fe pointwise tape

^{56}Fe

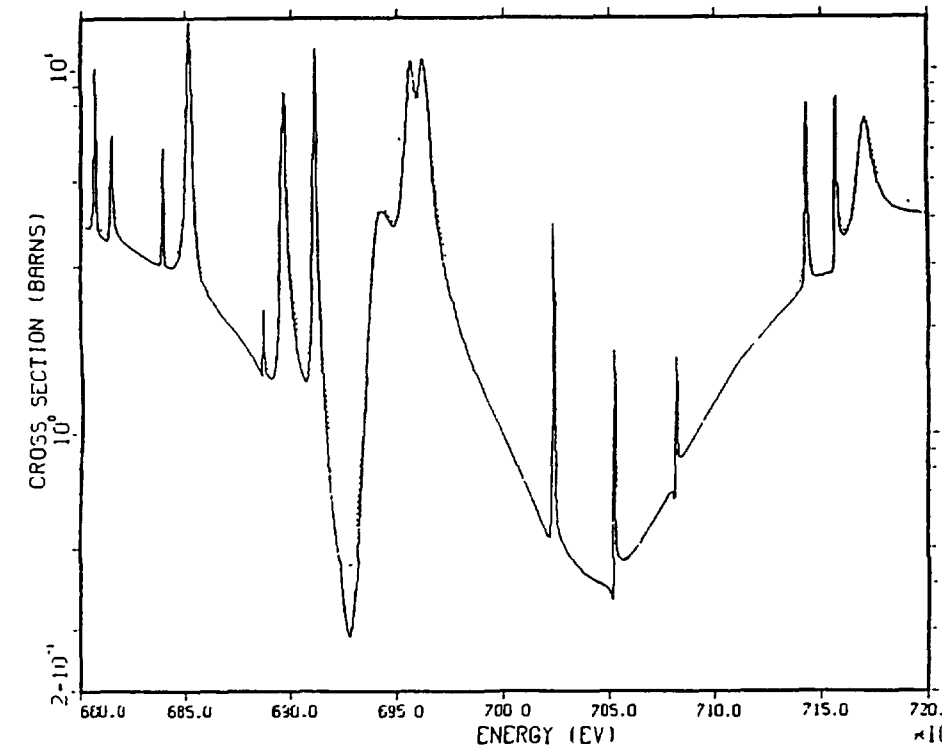
COMPARE RECONR (91.V69) VS RECENT ELASTIC XSEC
PLOT 2



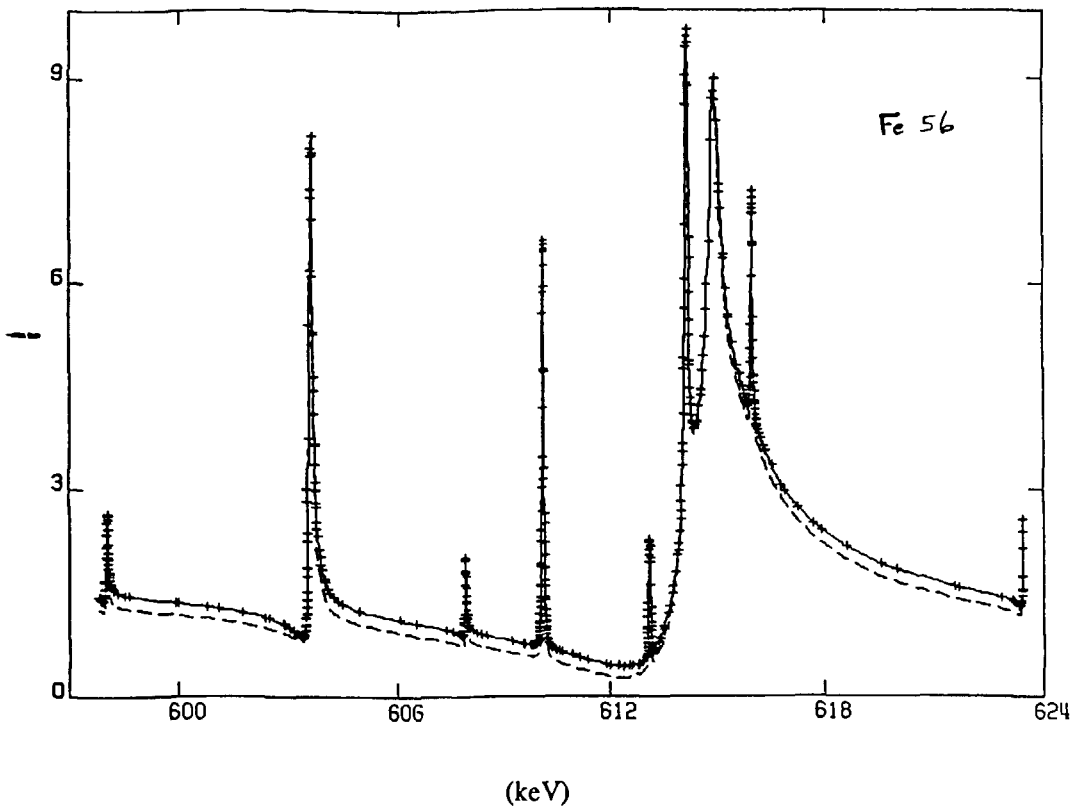
SYM	MT	SOURCE	MAT
	2	ENDF	2631
	2	ENDF	2631

^{56}Fe

COMPARE RECONR (91.V69) VS RECENT ELASTIC XSEC
PLOT 2



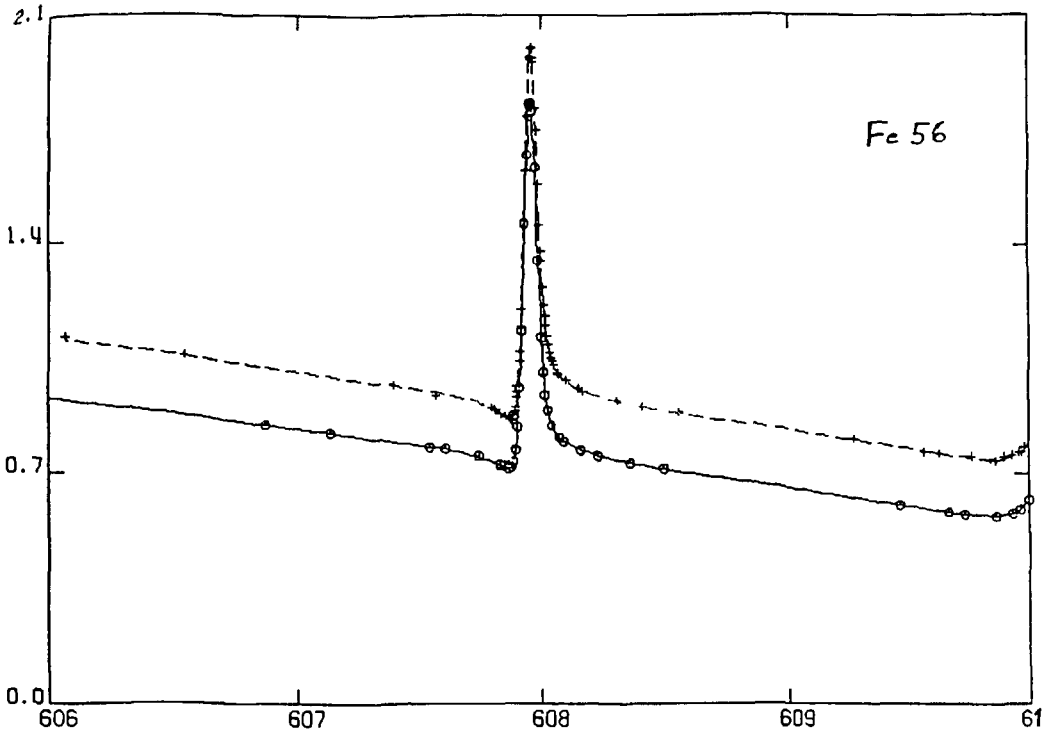
SYM	MT	SOURCE	MAT
	2	ENDF	2631
	2	ENDF	2631



+ RECENT calculation

——— SAMMY incorrect calculation (one radius)

- - - - - SAMMY calculation with two radii



+ RECENT calculation

o NJOY

——— SAMMY 2 radii

- - - - - SAMMY 1 radius

Stage 2: Set-up calculational procedure and input data for production task

- Select test nuclides to verify input deck for production task

- Check output edits against the library specifications (energy group structure; weighting spectrum, etc.)

- Note size of files, execution times for representative nuclides

  - CPU utilization
  - storage requirements for both scratch and permanent files

Stage 3: Generate Multigroup Data

- Review output and note all messages

- Resolve serious problem indicators

```
UNICOS 6.1.6 on a Cray Y-MP 8/64        *****       NOTICE       ******

getting endf tape
  got xnjoy92m:/605016/unix/xnjoy92m  (7162624 bytes   written:93/08/04
        14:16)  08/05 17:42
  got tape21:/endf/6u/neutron/fe/56a  (1366065 bytes   written:92/09/21
        22:00)  08/05 17:42
  got tape23:/605023/u6/point/fe56  (13847436 bytes   written:93/04/06 07:15)
         08/05 17:42
running njoy

 ---message from getmf6---bad grids for corresponding-point interpolation-
                          changing to unit-base interpolation

 ---message from getmf6---bad grids for corresponding-point interpolation-
                          changing to unit-base interpolation

 ---message from getmf6---bad grids for corresponding-point interpolation-
                          changing to unit-base interpolation

 ---message from getmf6---bad grids for corresponding-point interpolation-
                          changing to unit-base interpolation

 ---message from getmf6---bad grids for corresponding-point interpolation-
                          changing to unit-base interpolation

 ---message from getmf6---bad grids for corresponding-point interpolation-
                          changing to unit-base interpolation

 ---message from getmf6---bad grids for corresponding-point interpolation-
                          changing to unit-base interpolation

 ---message from getmf6---bad grids for corresponding-point interpolation-
                          changing to unit-base interpolation

 ---message from getmf6---bad grids for corresponding-point interpolation-
                          changing to unit-base interpolation

 ---message from getmf6---bad grids for corresponding-point interpolation-
                          changing to unit-base interpolation
CPU limit exceeded
TB001 - BEGINNING OF TRACEBACK
      - $TRBK     WAS CALLED BY f_sig     AT  372621b (LINE NUMBER      102)
      - f_sig     WAS CALLED BY ~handlr   AT  272306b
      - ~handlr   WAS CALLED BY GETFF     AT  750644d
      - GETFF     WAS CALLED BY PANEL     AT  737445c (LINE NUMBER     2660)
      - PANEL     WAS CALLED BY GROUPR    AT  725033b (LINE NUMBER      522)
      - GROUPR    WAS CALLED BY $SEGCALL  AT    1513d (LINE NUMBER      180)
      - $SEGCALL  WAS CALLED BY NJOY      AT     163a (LINE NUMBER      180)
      - NJOY      WAS CALLED BY $SEGRES   AT    1460d
      - $SEGRES   WAS CALLED BY $START$   AT     672a
TB002 - END OF TRACEBACK
Cpu limit exceeded
  saved tape24:/605016/ub6/fe56g.92m  (5392008 bytes)  08/05 18:25
  saved output:/605016/ub6/fe56o.92m  (3892359 bytes)  08/05 18:25
njoy run completed
logout
```

Stage 4: Preliminary testing of multigroup data

- Convert groupwise data to AMPX format
- Run RADE to expose processing problems
- Plot point versus multigroup data to check consistency

Checks Performed by RADE

1. $\sigma_t = \sigma_a + \sigma_s$

2. $\sigma_{in} = \sum \sigma_{in}^{partial}$

3. $\sigma_a = \sigma_c + \sigma_f$

4. $\sigma_c = \sigma_{n\gamma} + \sigma_{n\alpha} + \sigma_{np} + \sigma_{nd} + \ldots$

5. $\sigma_{el}^{g} = \sum_{g'} \sigma_{el,0}(g \rightarrow g')$ (also made on any process with a scattering matrix)

6. $\sigma_0(g \rightarrow g') > 0$

7. $\sigma_T,\ \sigma_a, \sigma_f, \sigma_{n\gamma}, \sigma_{np}, \ldots > 0$

8. $f_\ell^{\min} \leq f_\ell(g \rightarrow g') \leq 1.0,$

where $f_\ell(g \rightarrow g') = \dfrac{\sigma_\ell(g \rightarrow g')}{(2\ell + 1)\sigma_0(g \rightarrow g')}$

and $f_\ell^{\min} = -1.0$ for all odd ℓ, and is given in the following table for even ℓ

ℓ	$f_\ell^{\min}$
2	-0.5
4	-0.433
6	-0.419
8	-0.414

AN EXAMPLE OF RADE OUTPUT

```
2-D NEUT PROCESSES
        MT     LENGTH      ORDER      TEMPS
 1       2       1255          5          0
 2      51        508          5          0
 3      52        503          5          0
 4      53        497          5          0
 5      54        508          5          0
 6      55        522          5          0
 7      56        526          5          0
 8      57        541          5          0
 9      58        528          5          0
10      59        513          5          0
11      60        524          5          0
12      61        509          5          0
13      62        513          5          0
14      63        496          5          0
15      64        500          5          0
16      65        501          5          0
17      66        476          5          0
18      67        476          5          0
19      68        476          5          0
20      91       1053          0          0
21    1007       6126          3         10
MT=   4 GROUP    1 PO MATRIX SUM=  4.19004E-01 VERSUS AVERAGE VALUE=  4.19146E-01,FRACT DEV= -3.39017E-04
MT=   4 GROUP    2 PO MATRIX SUM=  4.95239E-01 VERSUS AVERAGE VALUE=  4.95431E-01,FRACT DEV= -3.86552E-04
MT=   4 GROUP    3 PO MATRIX SUM=  4.68027E-01 VERSUS AVERAGE VALUE=  4.68357E-01,FRACT DEV= -7.04657E-04
MT=   4 GROUP    4 PO MATRIX SUM=  4.23654E-01 VERSUS AVERAGE VALUE=  4.24218E-01,FRACT DEV= -1.32974E-03
MT=   4 GROUP    5 PO MATRIX SUM=  3.99918E-01 VERSUS AVERAGE VALUE=  4.00051E-01,FRACT DEV= -3.30615E-04
MT=   4 GROUP    6 PO MATRIX SUM=  4.22426E-01 VERSUS AVERAGE VALUE=  4.22648E-01,FRACT DEV= -5.24901E-04
MT=   4 GROUP    7 PO MATRIX SUM=  3.08307E-01 VERSUS AVERAGE VALUE=  3.08324E-01,FRACT DEV= -5.58689E-05
MT=   4 GROUP    9 PO MATRIX SUM=  1.22541E-01 VERSUS AVERAGE VALUE=  1.22669E-01,FRACT DEV= -1.04031E-03
```

Indicates reasonable transfer matrix processing

AN EXAMPLE OF RADE OUTPUT (continued)

```
MT=1007 P3 COEFFICIENT FOR SCATTER FROM GRP 193 TO GRP 217 IS UNREASONABLE--COEFF=  -2.599E+01
                    P3 VALUE=  -2.623E-08
                    P0 VALUE=   1.442E-10
P3 TRANSFER FROM 192 TO 217 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 191 TO 217 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 190 TO 217 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 189 TO 217 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 188 TO 217 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 187 TO 217 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 186 TO 217 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 195 TO 218 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 194 TO 218 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 193 TO 218 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 192 TO 218 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 191 TO 218 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 190 TO 218 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 189 TO 218 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 188 TO 218 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 187 TO 218 HAS NO P0 MATCH--PROCESS--> 1007
MT=1007 P3 COEFFICIENT FOR SCATTER FROM GRP 196 TO GRP 219 IS UNREASONABLE--COEFF=  -1.293E+00
                    P3 VALUE=  -6.499E-06
                    P0 VALUE=   7.179E-07
P3 TRANSFER FROM 195 TO 219 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 194 TO 219 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 193 TO 219 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 192 TO 219 HAS NO P0 MATCH--PROCESS--> 1007
MT=1007 P3 COEFFICIENT FOR SCATTER FROM GRP 196 TO GRP 220 IS UNREASONABLE--COEFF=  -2.064E+01
                    P3 VALUE=  -2.242E-06
                    P0 VALUE=   1.552E-08
P3 TRANSFER FROM 195 TO 220 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 194 TO 220 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 193 TO 220 HAS NO P0 MATCH--PROCESS--> 1007
P3 TRANSFER FROM 192 TO 220 HAS NO P0 MATCH--PROCESS--> 1007
MT=1007 P3 COEFFICIENT FOR SCATTER FROM GRP 196 TO GRP 221 IS UNREASONABLE--COEFF=  -1.187E+01
```

Other useful information provided by RADE/SMILER

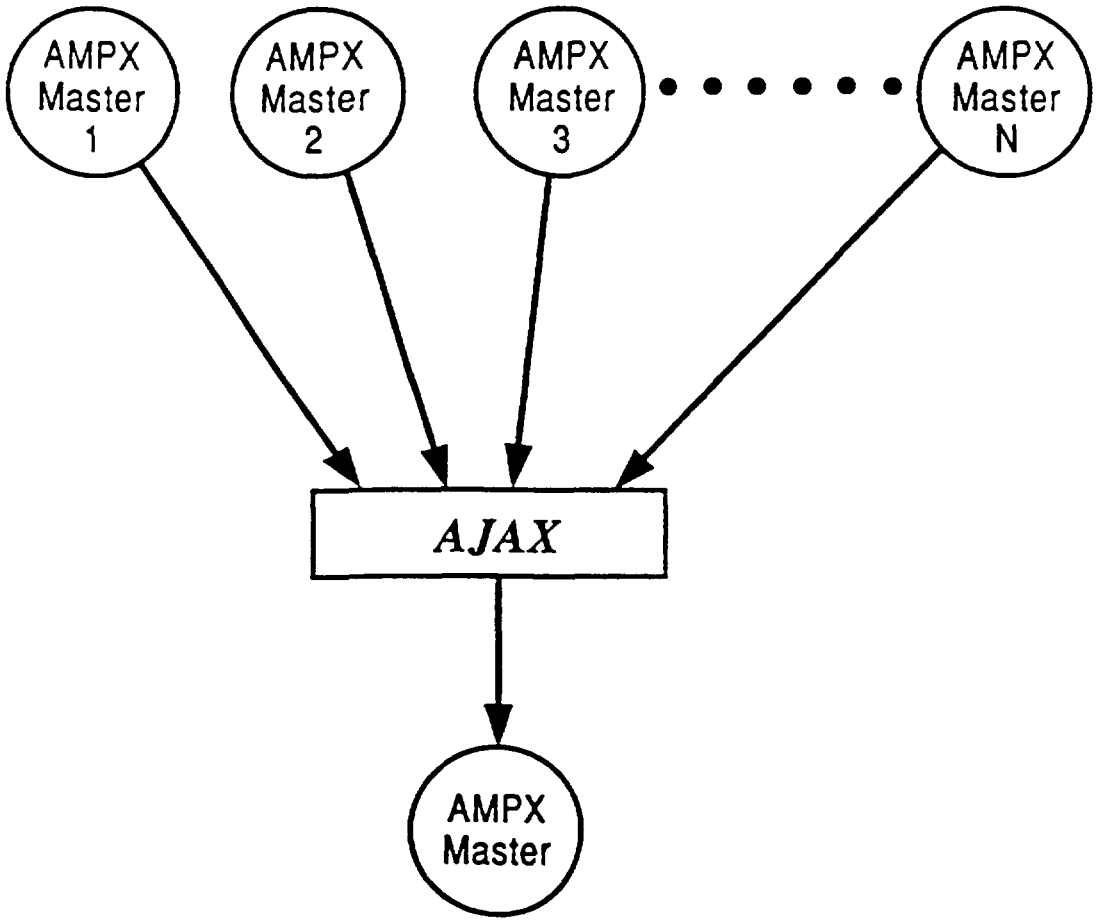
- Average energy released for photon production
- Minimum/Maximum f-factor value
- ELO, EHI for f-factors
- Normalization of spectrum for IG2L0 case

Stage 5: Build library and perform validation studies

- Calculate CSEWG benchmarks and compare with measurements and previously reported results

- Calculate other benchmarks (Winfrith Iron, NESDIP2, etc.)

ORNL-DWG 91Z-14374A



Procedure used to build final Libraries

CSEWG Data Testing Benchmarks

1. Thermal Reactor Benchmarks

ORNL-1, -2, -3, -4, and -10: unreflected spheres of U-235 (as uranyl nitrate) in H_2O. These benchmarks are useful for testing H_2O fast scattering data, U-235 absorption, and neutron capture in hydrogen.

TRX-1 and TRX-2: H_2O moderated uranium lattices. These lattices directly test U-235 resonance fission integral and thermal fission cross section. Also, U-238 shielded resonance capture and thermal neutron capture will be tested. These benchmarks are sensitive to U-235 fission spectrum and U-238 fast fission and inelastic scattering cross section.

BAPL-1, -2, and -3: H_2O moderated uranium oxide critical lattices in a triangular pattern.

PNL-3: unreflected plutonium sphere.

PNL-6, and -11: homogeneous aqueous plutonium nitrate spheres and cylinders. These benchmarks are useful for testing H_2O scattering data, cross sections for resonance and thermal fission of Pu-239 and the Pu-239 fission spectrum.

ORNL-7, -8, 9, -10, -11: reflected and unreflected spheres of U-235 (as uranyl flouride) in H_2O.

CSEWG Data Testing Benchmarks (continued)

2. Fast Reactor Benchmarks

JEZEBEL: a bare sphere of plutonium metal.

JEZEBEL-PU: a bare sphere of plutonium metal containing 20.1% Pu-240.

JEZEBEL-23: a bare sphere of U (98.13 at % U-233) metal.

GODIVA: a bare sphere of enriched uranium metal.

FLATTOP-25: a reflected sphere of enriched uranium metal.

BIG TEN: a reflected cylinder of uranium containing 10% U-235.

ZPR-3/11: fertile to fission uranium metal ratio of 7:1 with U-238 reflector.

ZPR-3/12: U-fueled assembly with uranium-graphite ratio of 4:1, U-238 blanket.

ZPR-6/6A: a uranium oxide fueled fast critical assembly.

ZPR-6/7: a large (3100 liter) plutonium oxide fueled fast critical assembly.

CSEWG Data Testing Benchmarks (continued)

3. Shielding Benchmarks

SDT1-5: "Broomstick" Experiments for Iron, Oxygen, Nitrogen, Sodium, and Stainless Steel.

SDT11: ORNL Benchmark for Iron and Stainless Steel.

SB2: Secondary Gamma-Ray Production for Thermal Neutron Spectrum.

SB3: Secondary Gamma-Ray Production for Fast Neutron Spectrum.

SB5: ORNL 14-MeV Stainless Steel/Borated Polyethylene Slab Experiment.

SB6: ORNL 14-MeV Iron Duct Experiment.

Other Shielding Benchmarks

Winfrith Iron Benchmark Experiment

University of Illinois Iron Sphere Benchmarks (14-MeV and Cf-252)

PCA-PV "Blind Test" Benchmark

Winfrith NESDIP2 and NESDIP3 Radial Shield and Cavity Experiments

PWR Shielding Benchmark (Computational)

LMFBR Shielding Benchmark (Computational)

LWR Shielding Benchmark (Computational)

CTR Standard Blanket Benchmark (Computational)

Stage 6: Re-process library

- Systematic errors will be found; the probability increases when the following conditions exist

  - Newly released evaluated data
  - Untested representation of the physics which the formats allow
  - Processing code has not been tested for rigorous application

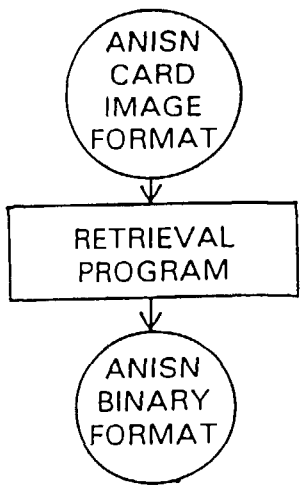
Stage 7: Preparation of data for distribution

- Assemble material with user needs as the primary driver
  - Format of data
  - Supporting library utility tools
- For fine-group libraries, modules for performing energy group collapse, self-shielding and temperature correction, etc.
- Prepare sample problems to illustrate how to use the data
- Clean-up ID label on each material to uniquely identify a particular data set
  (include source of data and processing code version)
- Documentation of data package
- Submit data to a center authorized to disseminate all elements of data package for its utilization

A DLC Package in "ANISN" Format Usually Contains at Least Three Parts Written on Separate Tape Files

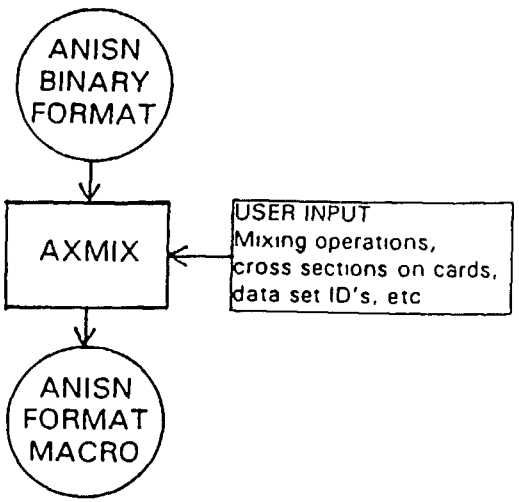
- The Multigroup Cross Sections in Fixed FIDO Format (Card Image).
- A Retrieval Program Whose Only Function is to Convert the Multigroup Cross Sections to Binary Form.
- Printed Output from a Run of the Retrieval Code (Lists the IDs Used to Identify the Materials on the Binary Library).

Typical Setup for a Multigroup DLC



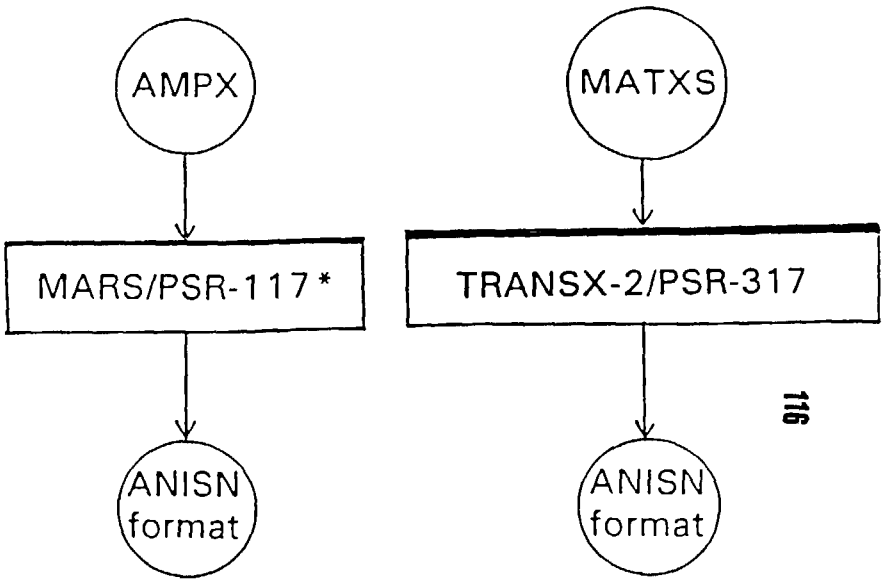
ANISN Binary Format Can Be Read Directly Into Codes Like ANISN, DOT, and MORSE.

PSR-75/AXMIX-GIP Can Also Be Used for Mixing and Other Useful Operations



* Free Form FIDO Input
* Merge, Delete, Update
* Macroscopic XSEC Production
* Transport Correction
* Nuclide or Group Oriented Libraries

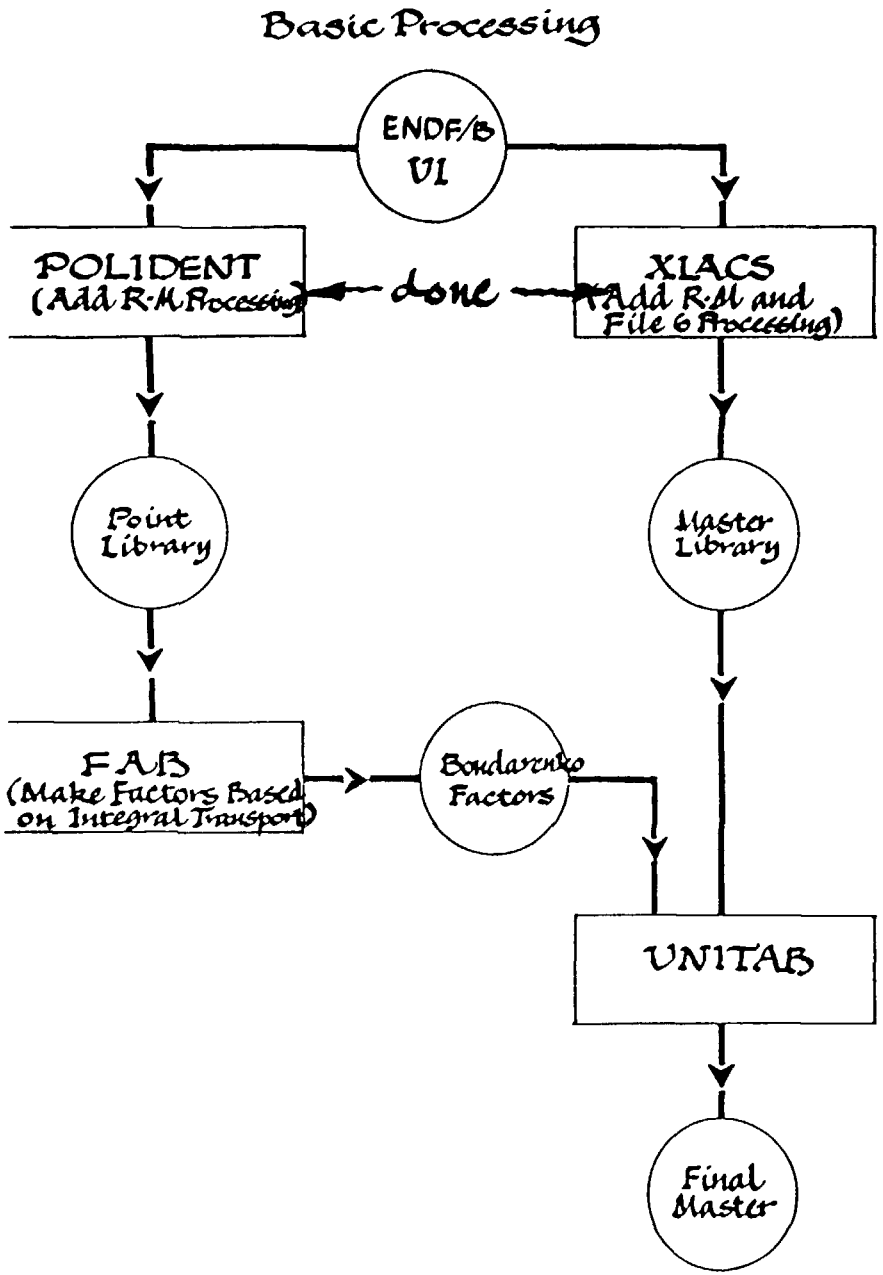
Fine-Group Libraries in a Flexible Format can be Translated into the ANISN Working Format

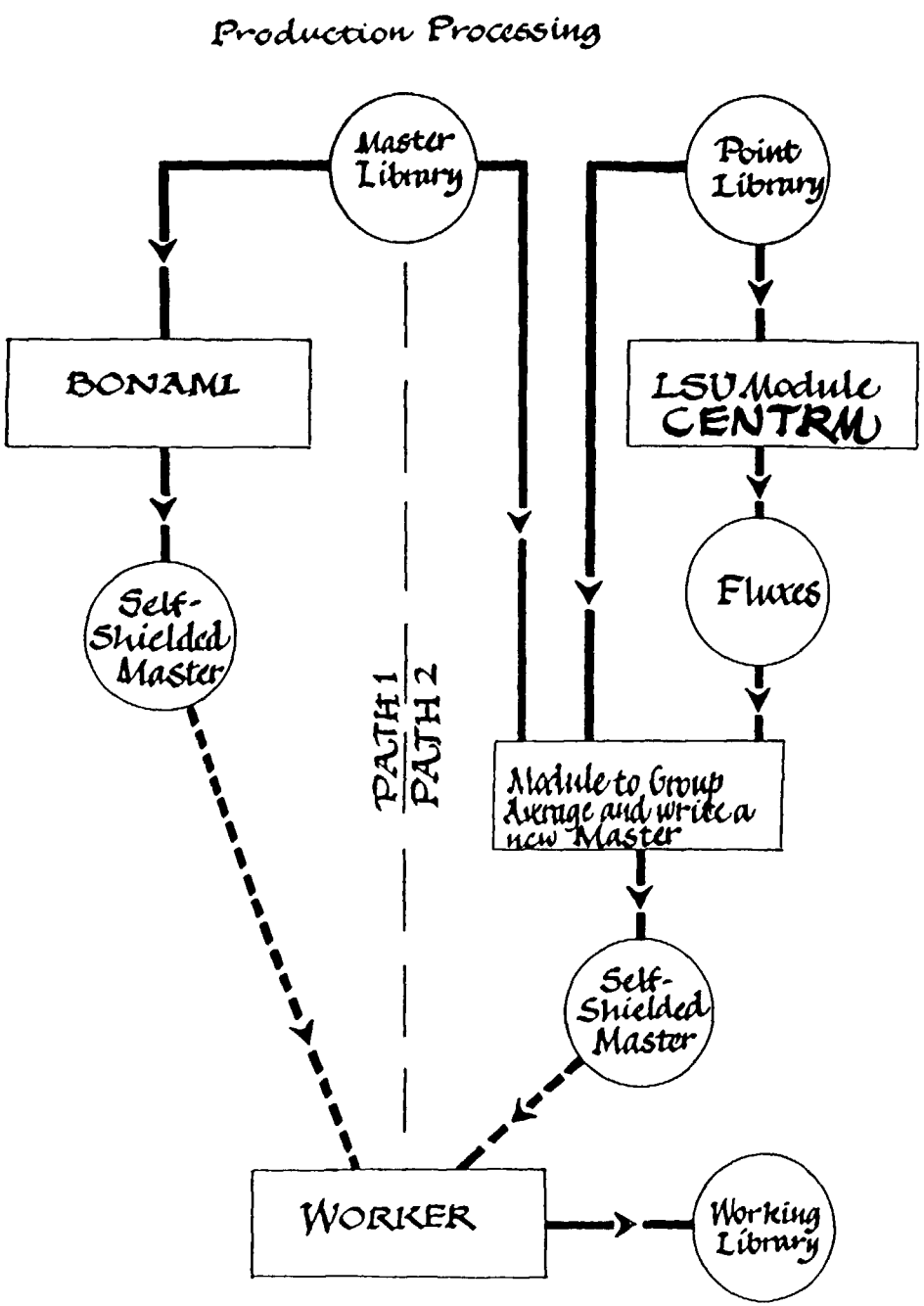


* Effect of temperature and resonance self-shielding can be treated by the f-factor method; group collapsing and other data manipulations.

Status of AMPX77 Processing

N. M. Greene





Chacteristics of LSV Module

- Discrete Ordinates (or alternate on arbitrary point mesh)
- Coupling to multigroup above and below resolved ranges
- Isotropic scattering in CM
- Newly developed efficient procedure for calculating scattering sources

Production of application libraries for WIMS

C.J. Dean

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Safety Engineering Systems Division
AEA Technology - Reactor Services
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Dorset DT2 8DH
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Abstract

The WIMS codes have been developed over a period of 30 years to include sophisticated modelling of many thermal reactor lattices. WIMS6 is currently being developed to include the flexibility of WIMSE together with the case of use seen in LWRWIMS. A new cross-section library is to be released with WIMS6. For the first time since the original WIMSD was written, the library is being generated from completely new differentual data based mainly on JEF2.2.

Quality Assurance into the XMAS library is achieved in three levels; through general procedures during assembly, by validation studies and by verification. The main general procedure involves forming overall physics quantities (Thermal average cross sections, resonance integrals and fission spectrum average values) from the point evalutions and comparing them with similar quantities from the XMAS library. Examples of validation are studies optimising the background cross-sections at which resonance integrals are tabulated and comparing reaction rates from homogeneous calculations with those formed in Monte Carlo and deterinistic codes which represent energy variation of cross-section on a hyperfine energy mesh. Verification involves performing benchmark calculations and comparing results with experiment and calculations with the Apollo code. The current XMAS library contains 140 nuclides based on JEF2.2 evaluations.

The WIMS XMAS Library (Summary)

C.J. Dean

The WIMS codes have been developed over a period of 30 years to include sophisticated modelling of many thermal reactor lattices. (MAGNOX, ARG, PWR, SGHWR, HTR, BWR, CAWDU, RBMK, VVER ...). The widly distributed WIMSD code has been enhanced dramatically to form the modular WIMSE code and the LWR-WIMS code which is optimised for use with light water assemblees. WIMS6 is currently being developed to include the flexibility of WIMSE together with the case of use seen in LWRWIMS. A new cross-section library is to be released with WIMS6. For the first time since the original WIMSD was written, the library is being generated from completely new differentual data based mainly on JEF2.2.

Modern Evaluated data are comprehensively validated internationally and have been distributed through data centres including the IAEA. Our task in generating the new XMAS library includes representing the improved features in the evaluated point data within a practical group scheme. This is achieved by preserving energy dependent reaction rates. Initial steps involved designing the group structure by combining boundries from the existing WIMS 69 and APOLLO 99 group schemes and extending the resultant structure to represent data features such as centring resonances and describing reaction thresholds. The resultant 172 energy group structure was validated by showing improved results over existing structures relative to 2000 group calculations. Studies of within group weighting spectra (collision density) were also validated by comparison with similar spectra used for VITAMIN-J libraries.

The NJOY code has been used to process the evaluated data into group constants. Point cross-sections are generated from resonance parameters preserving the shape of resonances within a tollerance of 0.1%. This requires NJOY modules to generate the energy grid using 9 significant figures and double precision coding invoked using computer options on SUN work stations. During this procedure enhancements were made to NJOY. These enhancements, together with other improvements, particularly those connected with solving slowing down equations have been communicated to the NJOY authors for inclusion in internationally improved code versions. This arrangement has required the design of Quality Procedures for maintaining NJOY which are similar to those used in "ANSWERS" for marketing commercial code packages.

We have improved the WIMSR module of NJOY to input current spectra, improve the definition of scattering power (ξ) and to form resonance integrals consistent with WIMS definitions.

A code called WILT (WIMS Improved Library Translation) has been written to assemble the group constants from the NJOY WIMSR output, on WIMS libraries. A new direct access library structure has been designed by labelling each data type using a concept known as a Datagram. Quantities are then collected into items of the same type (reals, characters or integers). By quoting a datagram label it is possible for standard accession routines to obtain physics quantitites directly, either by random access I/O reads or by transfering from the library kept in computer memory.

It is very important to build quality Assurance into the XMAS library. This is achieved in three levels; through general procedures during assembly, by validation studies and by verification. The main general procedure involves forming overall physics quantities (Thermal average cross sections, resonance integrals and fission spectrum average values) from the point evalutions and comparing them with similar quantities from the XMAS library. Examples of validation are studies optimising the background cross-sections at which resonance integrals are tabulated and comparing reaction rates from homogeneous calculations with those formed in Monte Carlo and deterinistic codes which represent energy variation of cross-section on a hyperfine energy mesh. Verification involves performing benchmark calculations and comparing results with experiment and calculations with the Apollo code.

The current XMAS library contains 140 nuclides based on JEF2.2 evaluations. We aim to have the first version of the library released to users for practical testing in March 1994. There will follow a year of benchmarking and consolidation work culminating in an unadjusted library for customer use by March 1995. We will then make any adjustment necessary and produce the WIMS7 product library optimising code and library to give the best results for thermal reactor calculations. The target date for this product is March 1996.

WIMS
"XMAS" LIBRARY

INTRODUCTION

- WIMS Status - interface with "XMAS" library.
- Evaluated Nuclear Data Base.
- Group Boundaries + Weighting Spectra.
- Processing Route + NJOY developments.
- Quality Assurance.
- Conclusions.

WIMS
CODE STATUS

SOPHISTICATED MODELLING CAPABILTY FOR

UK Programme	MAGNOX	AGR	PWR
Other UK Designs	SGHWR	HTR	
US/Canada	BWR	CANDU	
Russian	RBMK	VVER	

STORAGE, REPROCESSING, TRANSPORT

ENDLESS BENCHMARK CONFIGURATIONS

LINKS WHOLE CORE CALCULATIONS - PANTHER

AEA Technology

WIMS
CODES

WIMSD
- Simple models - benchmarking.
- Simple routes - easy to set up.

WIMSE
- Modular.
- Sophisticated methods.
- Main development code.
- More difficult to use.
- MAGNOX, AGR.

LWRWIMS
- Optimised for Light Water calculations.
- Standard routes.
- Less flexible than WIMSE.
- Easier to use than WIMSE.
- PWR, BWR.

1994 WIMS6
- WIMSE + LWRWIMS + XMAS Library

1995 WIMS7
- General Code.
- Simplified input - samples.
- Including Monte Carlo - MONK5W.
- Optimised XMAS Library.
- Single code to maintain.
- Supplied by module to suit customer needs.

8-10 December 1993 WIMS "XMAS" LIBRARY IAEA Vienna

History of the Differential Nuclear Data in WIMS Libraries

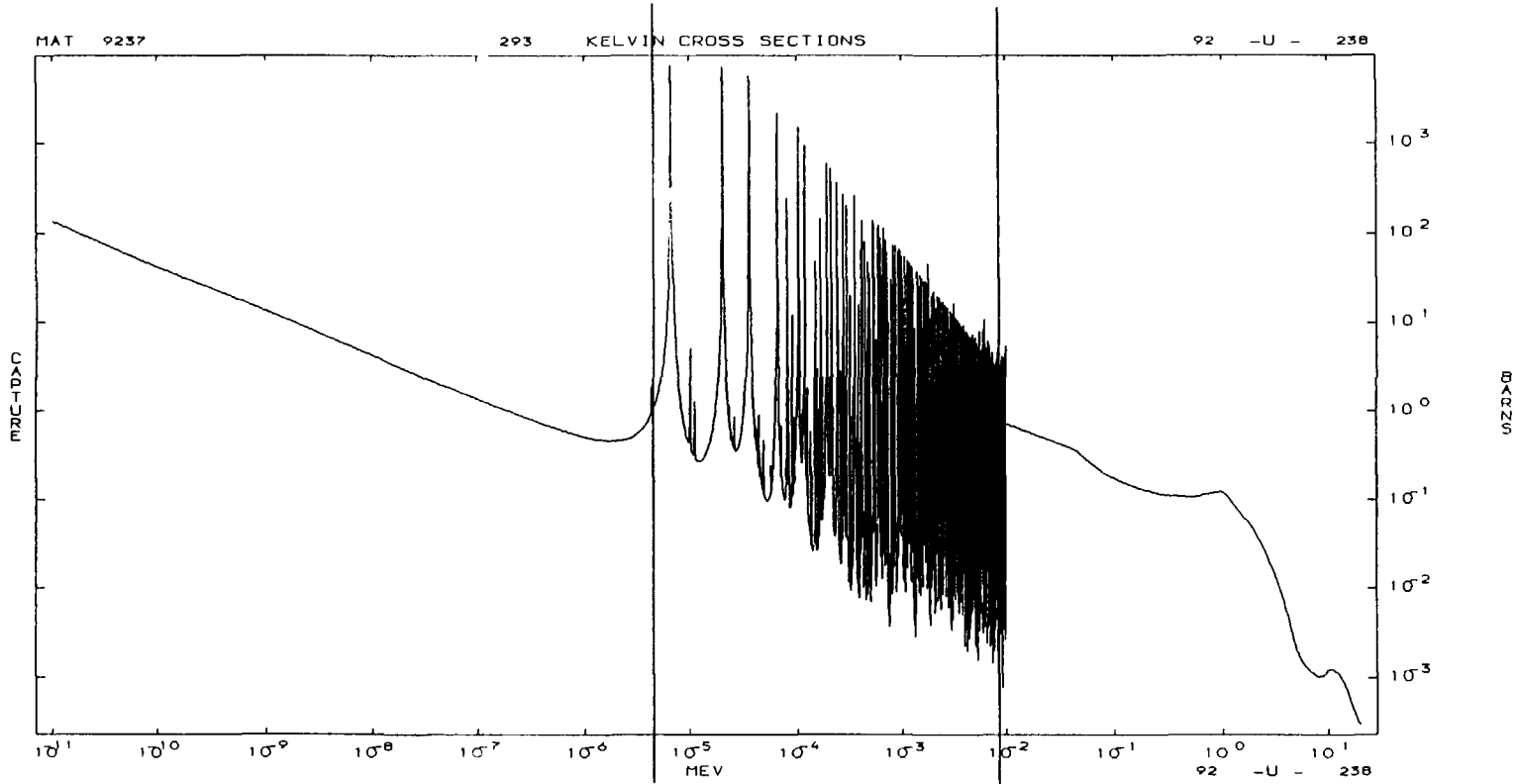
YEAR	UK EVALUATED DATA	FORMAT	OTHER MAJOR DATA CHOICE	PROCESSING	REMARKS
1964	UKNDL	UKNDL	-	GALAXY3 PIXIE GENEX/SDR	The UKNDL probably contained the best consistent nuclear data in the world at this time
1975	UKNDL	UKNDL	ENDF/B III	GALAXY6 PIXIE GENEX/SDR WIMRES-ERIC2	Data of equal quality UKNDL has better QA than ENDF/B III
1981	UKNDL	UKNDL	ENDF/B IV	GALAXY MURAL	Fewer new evaluations in UKNDL (Am^{241}, Am^{243}, Hf isotopes) than ENDFB IV ENDFB IV QA improved.
1986	JEF1	ENDF-5	Parts of ENDF/B V	NJOY 6/83	Only secondary isotopes and standards available from ENDF/B V. JEF1 contains selection of the best data from. ENDF/B IV, ENDF/B V, JENDL-2, ENEA,KEDAK, etc UKNDL for some parts of Hf evaluations
October 1986	JEF1.1	ENDF-5	-	NJOY87	Improvements to JEF1 due to experiences in processing
1990	JEF2.1	ENDF-6	ENDF/B VI JENDL-3	NJOY89	New European evaluations UK evaluations U^{238}
1993	JEF2.2 ENDFB-VI (revision 2) JENDL3.1 EFF	ENDF-6	-	NJOY89.62W	Access to all 3 libraries + further coordinated evaluation.

WHY GENERATE A REVISED LIBRARY ?

- 1986 WIMS Library based on 1970's differential data (secondary isotopes from later evaluations) e.g. Ti, Mo, Gd^{154}, Gd^{156}, Gd^{158}, U^{237}, Np^{237}, Np^{239}, Pu^{238}, Pu^{241}, Pu^{242}, Am^{241}, Am^{242m}, Am^{243} from JEF1.
- Integral adjustments made to give the present quality of data.
- New integral data considered as experimental results became available.
- Gives good results for Uranium fuelled systems but we could improve results for other systems, e.g. MOX.
- C/E values often given relative to later differential data (difficult to compare calculated results with those from other centres).
- Much new differential data to be considered.
- Still very necessary to benchmark resultant library and to make adjustments.
- Benchmark calculations require a quality assured cell code so C/E differences are associated with data differences rather than method approximations.

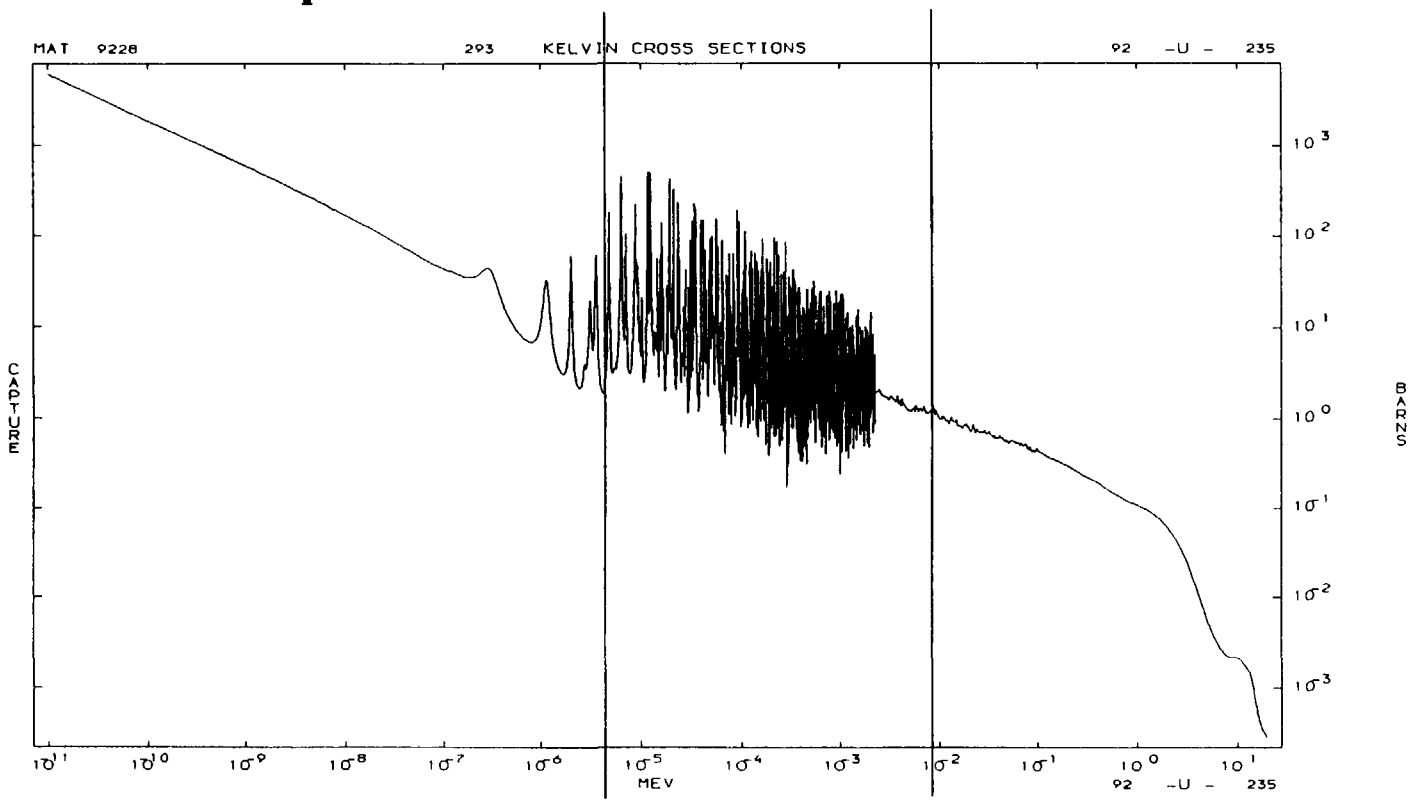
WIMS
ENERGY REGIONS

U238 Capture Cross Sections at 293.16K



WIMS
ENERGY REGIONS

U235 Capture Cross Sections at 293.16K from JEF 2.2



RANGE OF RESONANCE PARAMETERS

		Resolved		Unresolved
		Bottom	Join	Top
U^{235}	UKNDL	1.0E-5 eV	# 0.147 KeV	25 KeV
	JEF 1	1.0 eV	0.082 KeV	25 KeV
	JEF 2	0.15 eV	2.25 KeV	25 KeV
U^{238}	UKNDL	1.0E-5 eV	5 KeV	25 KeV
	JEF 1	1.0E-5 eV	4 KeV	50 KeV
	JEF 2	1.0E-5 eV	10 KeV	300 KeV
Pu^{239}	UKNDL	1.0E-5 eV	.6 KeV	25 KeV
	JEF 1 *	1.0 eV	.65 KeV	30 KeV
	JEF 2	1.0E-5 eV	1.0 KeV	30 KeV
	ENDF/B-VI		2.0 KeV	

* Defined by energy, cross section pairs up to 201.5 eV

BNL 325, 3rd Edition 1973

AEA Technology

FUTURE LIBRARY DEVELOPMENT

1. **JEF and EFF** — Merge best files from JEF and EFF to become J/EFF
   Projects to continue in parallel

2. **ENDF/B-VI** — Experimental programme and evaluation will be drastically cut.
   Support for major new releases of ENDF are limited!
   Revision 2 released in June.

3. **JENDL3** — Revision 2 expected soon.
   Improvements to heavy nuclides and gamma ray production data.

<u>Benchmarking continuing on all 3 libraries above.</u>

4. **EAF** — Extensions and improvements in version 4 at ECN Petten. Potential source for FISPIN cross sections.

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PROCESSING EVALUATED DATA

JEF evaluated point Energy data

e.g.

$$n_E \rightarrow U^{235} \rightarrow \sigma_E^f \rightarrow (I^{135} + Y^{98})_{E_K} + n_{E'_1} + n_{E'_2} + n_{E'_3} + \gamma_{E'_4}$$

↓

Processing Code NJOY89.62 generates Grouped data preserving reaction rates

Neutron Loss	Neutron Gain
$\sigma_G = \frac{\int_G \sigma_E \phi_E dE}{\int_G \phi_E dE}$	$\sigma^{G'-G} = \frac{\int_{G'} \sigma_{E'} \phi_{E'} \int_G P(E' \rightarrow E)\, dE dE'}{\int_{G'} \phi_{E'} dE'}$
(absorption, total)	(fission, scatter)

(ϕ required)

↓

WIMS Library

e.g. $n_G \rightarrow U^{235} \rightarrow \sigma_G^f \rightarrow (I^{135} + \text{Pseudo})_{E_{K'}} + \bar{\nu}_G n_{G'} + \gamma_{g'}$

Grouped Data (in Energy)

AEA Technology

"XMAS" GROUP SCHEME

172 Neutron Groups

(2 added in range 3.3 - 1.2 eV for Pu^{242} capture)

19.64 MeV -> 0.00011eV

- **Includes all 99 APOLLO + 69 WIMS group bounds**
- **Reduce dependence on weighting function (both flux and current)**
- **Represent thresholds - U^8 fission, (n,2n)**
- **Centre intermediate resonances of heavy nuclides in a single group.**
- **Detailed representation of**
  - **Wide resonance in light (O^{16}) and structural (Fe) materials.**
  - **Thermal resonances of heavy absorbers e.g. U^{235}, Pu^{239}, Pu^{240}, Pu^{241}.**
  - **Elastic scattering matrices.**
  - **Transport cross-section.**

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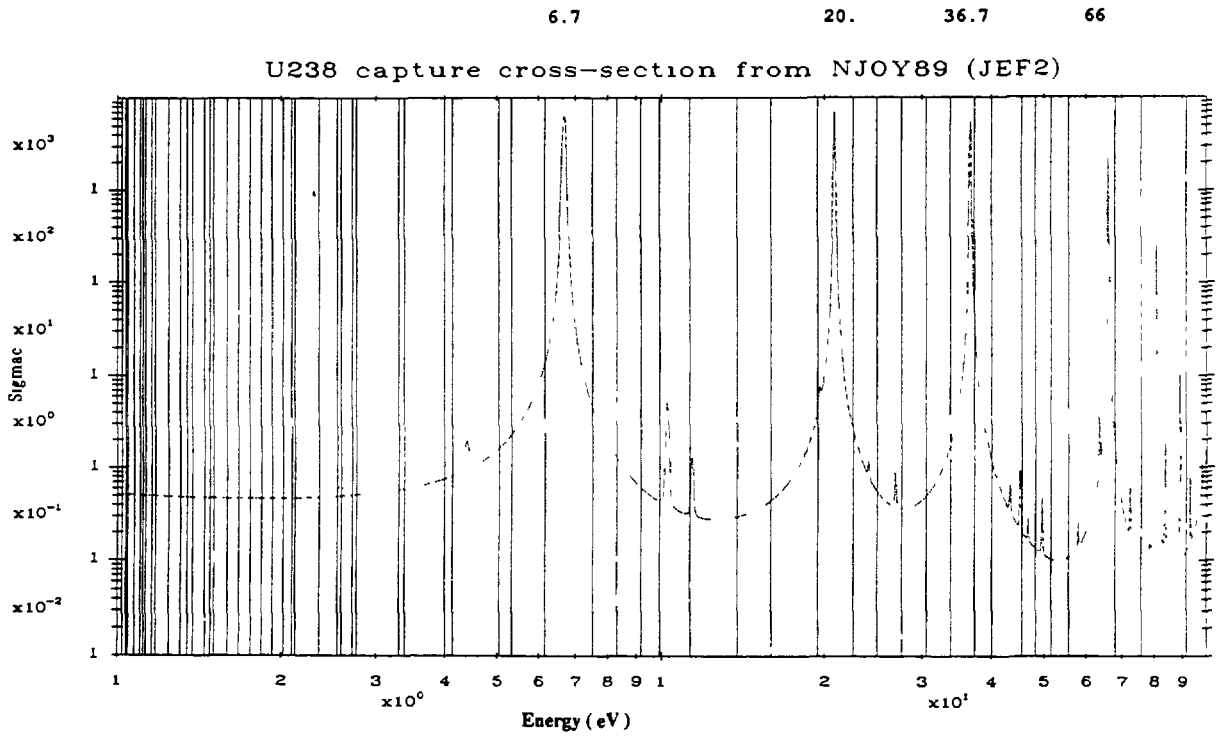
GROUP BOUNDS



AEA

VALIDATION OF RESONANCE SHIELDING "XMAS" GROUP STRUCTURE

69 Groups → 172 Groups

14 Fast	19.64 MeV	45	9 KeV
13 Resonance	9 KeV	47	4 eV
42 Thermal	4 eV	80	1.1E-4 eV

"XMAS" GROUP SCHEME
Validation by CEA (G. Rimpault)

- ECCO Cell Code above 0.414 eV
- FGL5 Library (2240 groups)
- FGL5 Condensed to APOLLO (99 groups)
- FGL5 Condensed to 170 groups.

% Reactivity Differences from 2240 groups		
CELL	99 Groups	170 Groups
ERASME S	-0.340	-0.117
ERASME R	-0.419	-0.167
ERASME L	-0.475	-0.211
(11% Pu MOX Fuel)		
ERASME S	-0.034	-0.184 *
(Voided Cell)		
CAMELEON	-0.475	-0.201
(PWR Cell)		

* Cancelling effects from Fe and O

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WEIGHTING SPECTRA
(C_E)

- Form correct reaction rates - $\sigma_g \phi_g$
- $\left(\int_g \sigma(E)\,\phi(E)\,dE\right) / \left(\int_g \phi(E)\,dE\right)$
- $\left(\int_g \sigma(E)\,\frac{C(E)}{\sum_T(E)}\,dE\right) / \left(\int_g \frac{C(E)}{\sum_T(E)}\,dE\right)$
- $C(E) = \phi(E) \sum_T(E)$ - collision density
- Smooth function of E
- No resonance dips
- Shielding and slowing down separate

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WEIGHTING SPECTRA

Original Spectrum	
(NJOY Option 4)	
Up to 0.2 eV	Maxwellian (T=0.0253 eV)
0.2 eV to 0.82085 MeV	1/E
0.82085 MeV to 20 MeV	Maxwellian fission spectrum (T=1.3539 MeV)

Vitamin J Spectrum	
(NJOY Option 11)	
Up to 0.414 eV	Maxwellian (T=0.0253 eV)
0.414 eV to 2.21255 MeV	1/E
2.21225 MeV to 10 MeV	Maxwellian fission spectrum (T=1.415 MeV)
10 MeV to 12.52 MeV	1/E
12.52 MeV to 15.68 MeV	Fusion Peak
Above 15.68 MeV	1/E

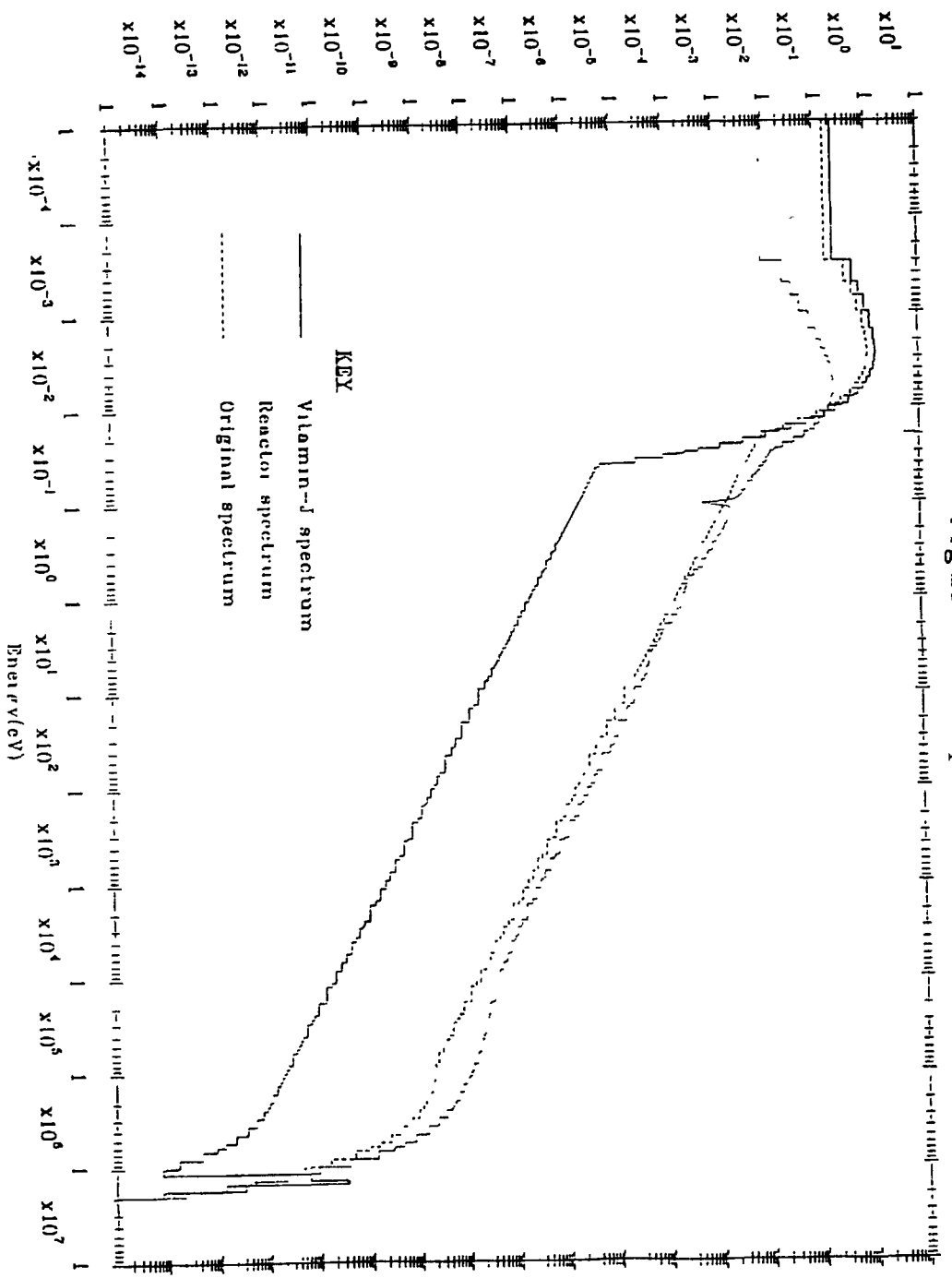

Figure 1. Spectra

INFINITE DILUTE CROSS SECTIONS

U238 Infinite Dilute Cross Sections (barns)				
Reaction		Overall Cross section Vitamin-J	Overall Cross section Original	Percentage Difference
Name	MT			
Total	1	1.959296E+01	1.959278E+01	0.0009
Elastic	2	1.275994E+01	1.275985E+01	0.0007
Inelastic (total)	4	6.190315E-01	6.190686E-01	0.0060
(n,2n)	16	2.626312E-03	2.629749E-03	0.1307
(n,3n)	17	2.575788E-05	1.341147E-05	92.0586
Fission	18	6.431512E-02	6.436598E-02	0.0790
(n,n')L1	51	2.587817E-01	2.586752E-01	0.0412
.				
.				
.				
(n,n')L25	75	7.110717E-04	7.151663E-04	0.5725
(n.n') continuum	91	2.066075E-01	2.068258E-01	0.1055
(n,γ)	102	6.147019E+00	6.146847E+00	0.0028

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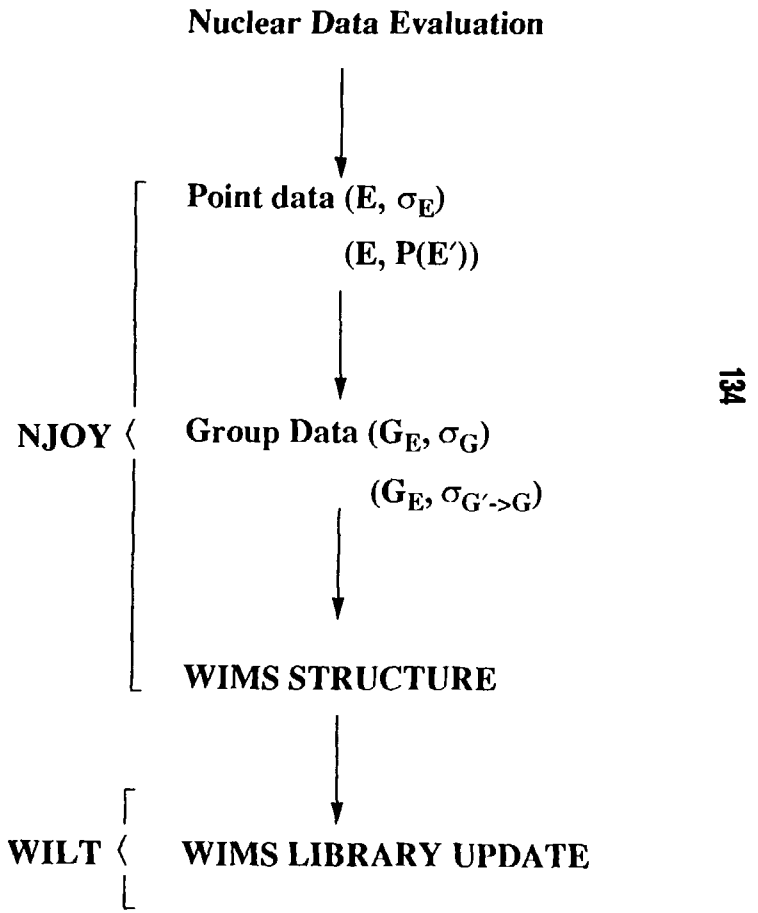
PROCESSING ROUTE INTRODUCTION



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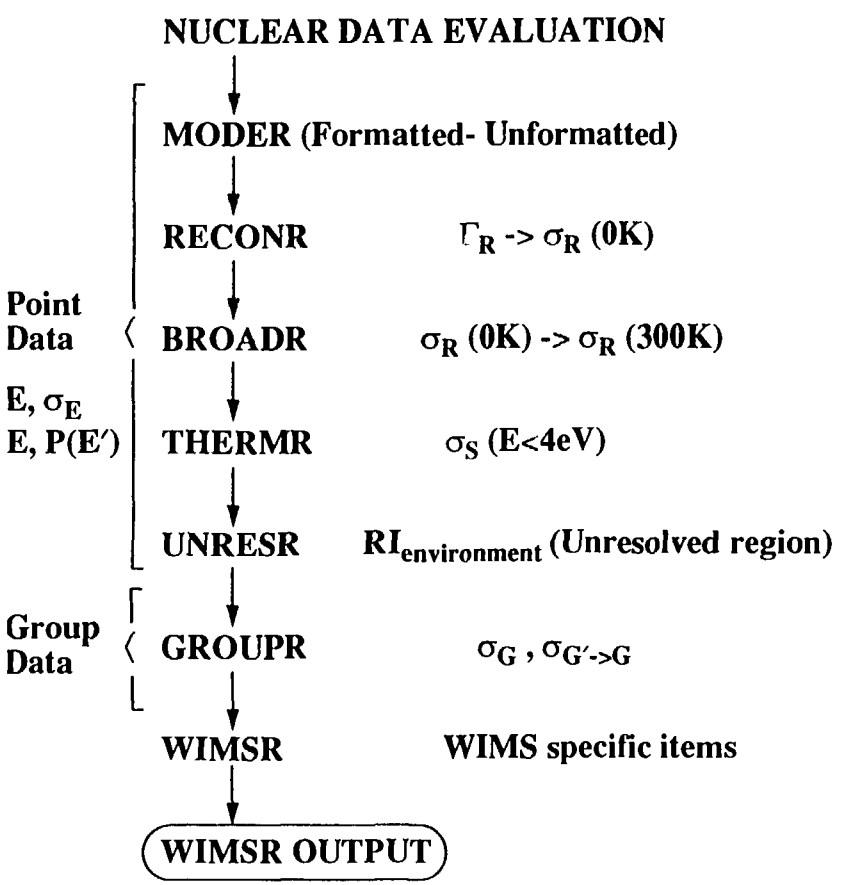
FUNCTIONS OF NJOY (WIMS LIBRARY PRODUCTION)



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NJOY DEVELOPMENT PRECISION

- Single precision IBM based version failed to reconstruct resonances from parameters.
- Test double precision version of RECONR, BROADR, GROUPR made using TOOLPACK (E. Webster).
- Removed resonance reconstruction problems (9 significant figures for energy grid for U^{238}). 0.1% error at intermediate energy points.
- Source code now very different from original.
- CDC, CRAY Single precision source used.
- -r8 option of SUN FORTRAN 77 compiler Version 1.3.1
  (Double precisions integers and reals. Double precision code with internationally compatible source. Significant changes to storage of character strings in output modules.)
- Removal of routine DIGITS from BROADR. BROADR input tolerances as RECONR.

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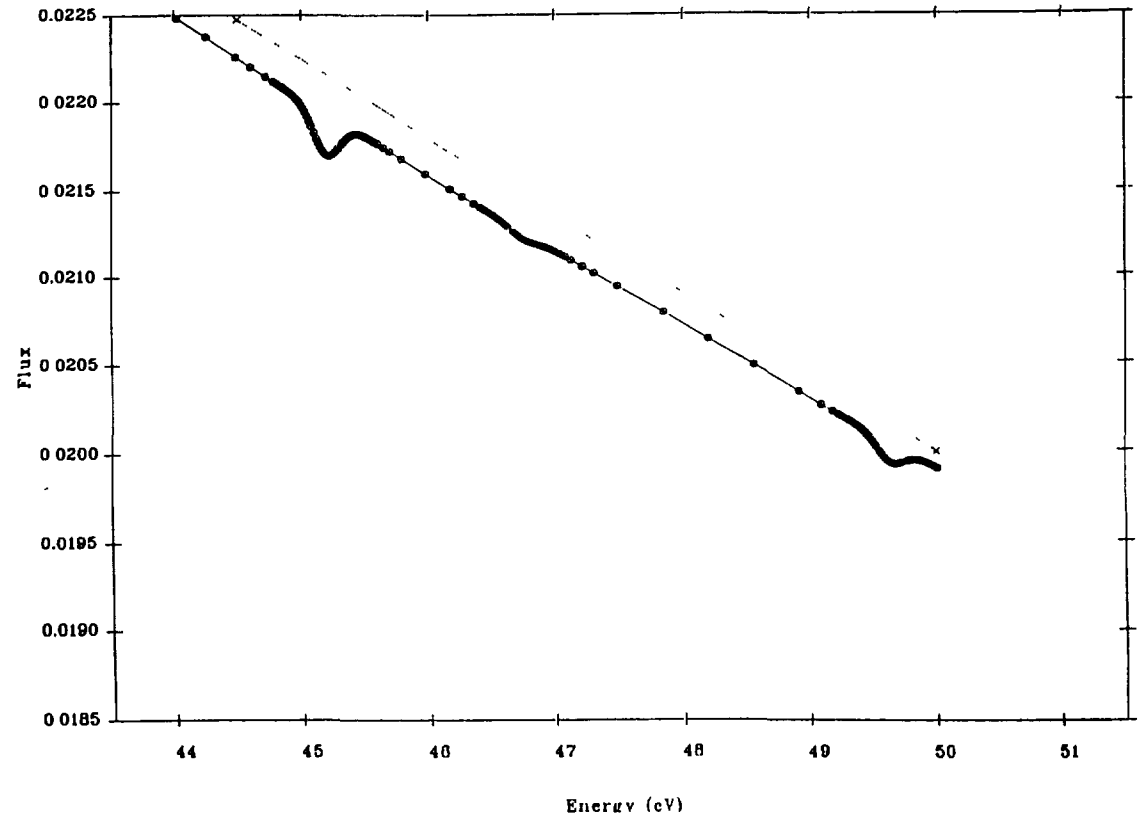
FLUX FROM NJOY FLUX CALCULATOR (OPTION -4) CORRECTED



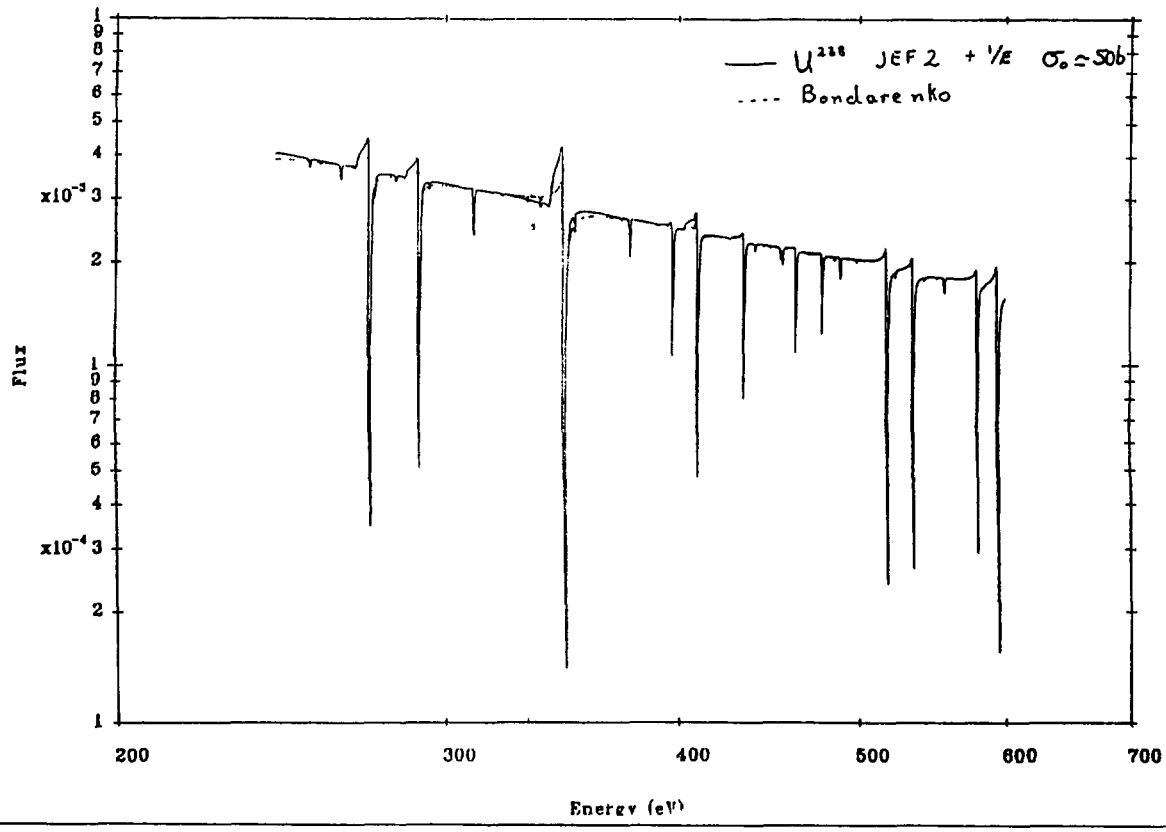
SLOWING DOWN FLUX

- Energy range 4 eV -> 500 eV (U^{238})
- Calculated by NJOY89.62
- Problems resolved
  - (a) Incorrect structure due to missing source
  - (b) Dimension limitations reducing the range
- NJOY options
  - i) absorber + 1/E source (APOLLO)
  - ii) absorber + H (WIMS)
  - (previously GENEX/SDR)

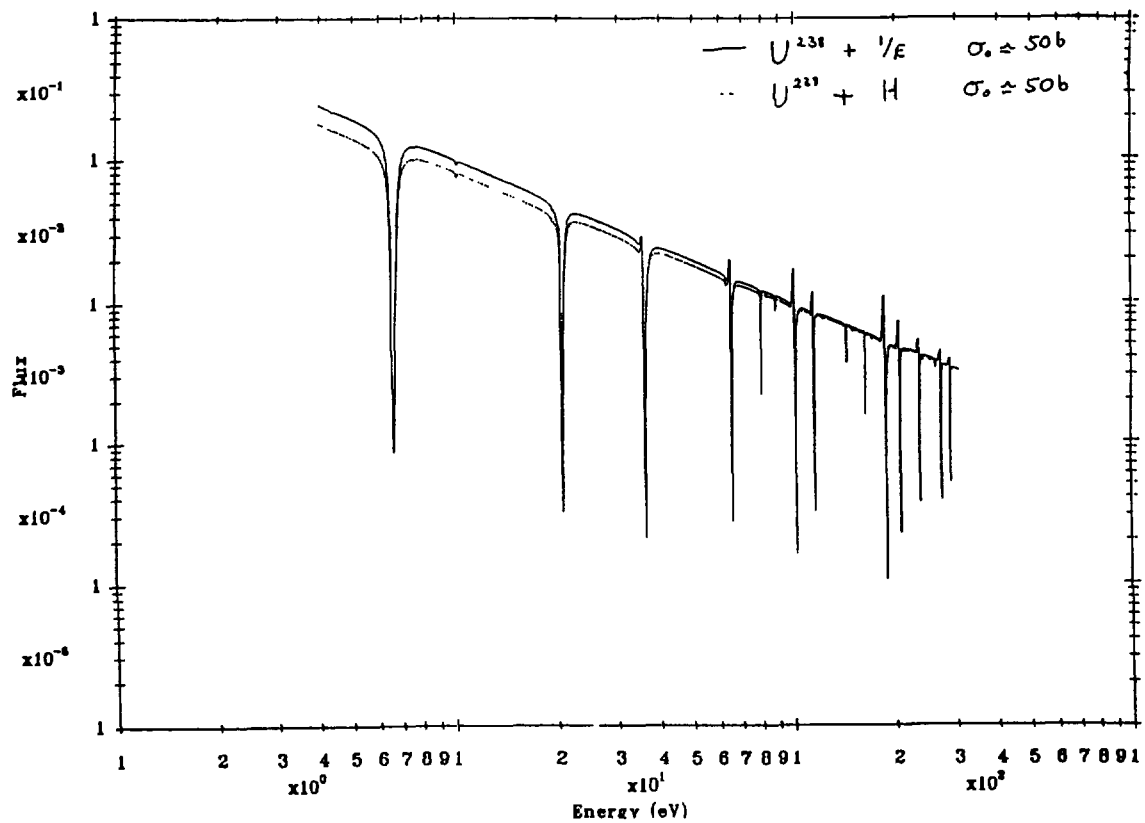
σ_0 is a measure of the amount of H.

$\sigma_0 = 50$ is ~ 2.5 : 1 ~ H : U^8 ($\lambda_h = 1$, $\sigma_{pot} \approx 20$)

COMPARISON OF FLUX FROM BONDARENKO AND NJOY (HIGH ENERGY)



COMPARISON OF FLUX HOMOG. - ADMIXED MOD. - GAMMA = 1.0



EUROPEAN PROCEDURES FOR NJOY MAINTENANCE

- OECD NEA NJOY Seminars

- OECD NEA NJOY User Group

  Connected with Sub-group B (Evaluated data Formats and Processing for Application Libraries) of NEANSC Working Party on International Evaluation Co-operation.

  Coordinates European development.

  (a) Meets twice a year.

  (b) Formal papers describing user developments.

  (c) Developments discussed and sent to MacFarlane.

  (d) European evaluators aware of developments before Los Alamos issues an update.

  (e) Data base of test cases.

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LOCAL PROCEDURES FOR NJOY DEVELOPMENT

- Source - international base version + international updates.
  e.g. NJOY89.62 = NJOY89 + 62 updates

- Files of UPDATE changes kept by type
  e.g. "update.win"- Winfrith international improvements.
  "update.win.sun4" - Changes to suit local workstation environment
  "update" files maintained using SCCS

- Maintained in a reference area.
  /eagle2/jeflib/codes/njoy89

- Command to extract fortran to a user area - "getnjoy"

- Full database of test cases.

- ANSWERS database for logging problems.

- Detailed QA notes.

- Feed back to NJOY User Group.

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LOCAL DEVELOPMENT

Step 1. Problem discovered - module known.

Step 2. Register in ANSWERS Database.

Step 3. "getnjoy" extracts source of module.
e.g. getnjoy njoy groupr

Step 4. Make a private sub-program
(maintain with SCCS).

Step 5. Investigate with "dbxtool"

Step 6. Find problem, make correction.

Step 7. Retest.

Step 8. "getnjoy njoy groupr" in new area.

Step 9. Compare corrected source (STEP 6) with original source (STEP 8).

Step 10. Make "UPDATE" instructions.

Step 11. Update status in ANSWERS data base.

Step 12. Arrange master version update and addition of a new test case with code custodian.

WIMSR DEVELOPMENT

- Source obtained informally from Canada via Switzerland
- Not released with NJOY89.62
- Released with NJOY91.91
- Include our developments in released version
- Feedback updates to R MacFarlane
(Need to be optional)

WIMSR IMPROVEMENTS AT WINFRITH

TRANSPORT

Thermal Energies

$$\sigma_{tr}(g) = \sigma_a(g) + \sigma_{s0}(g) - \sum_h \sigma_{s1}(g \rightarrow h)$$

Resonance Energies

$$\sigma_{tr}(g) = \sigma_a(g) + \sigma_{s0}(g) - \sum_h \sigma_{s1}(h \rightarrow g)\frac{J(h)}{J(g)}$$

Input J from AEEW-M 1782 (H, C, D)

NJOY B1 flux otherwise

(C(E) / (σ_T(E))^2)

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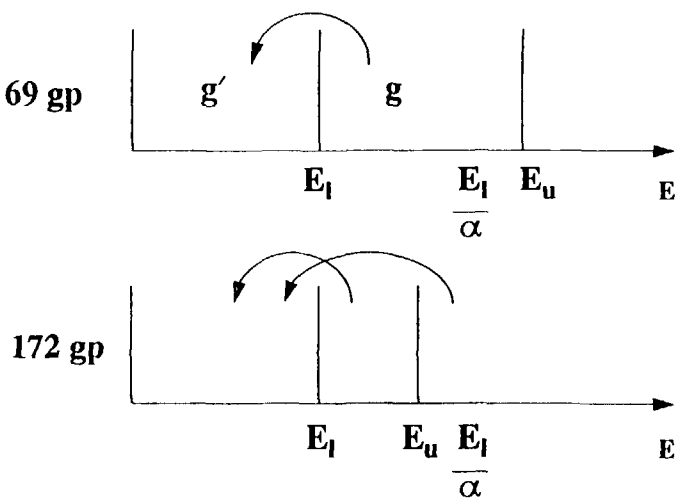
WIMSR IMPROVEMENTS AT WINFRITH

SCATTERING POWER / UNIT LETHARGY

$$\frac{\xi\sigma_s(g)}{\tau} = \sum_h \sigma_s(g \rightarrow h)$$



Note for A < 27 $\xi = 1 + \frac{\alpha}{1-\alpha} ln\alpha$

$$\alpha = \left(\frac{A-1}{A+1}\right)^2$$

Long term - improve GROUPR to calculate ξ correctly.

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WIMSR IMPROVEMENTS AT WINFRITH

RESONANCE INTEGRALS

$$I^a(\Sigma_p) = \frac{\Sigma_p \sigma^a}{\sigma_a + \Sigma_p}$$

$$I^f(\Sigma_p) = \frac{\Sigma_p \bar{\nu}\sigma^f}{\sigma_a + \Sigma_p}$$

$$\Sigma_p = \sigma_0 + \lambda\sigma_{pot}$$

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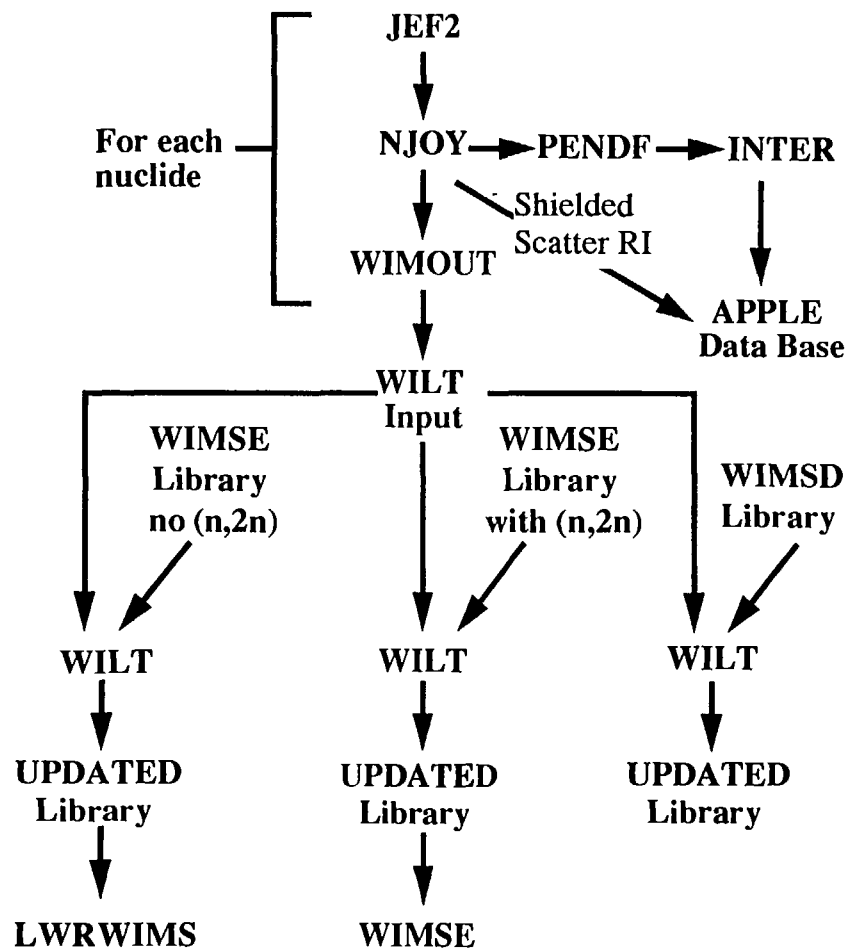
CREATION OF 172 GROUP "XMAS" LIBRARIES



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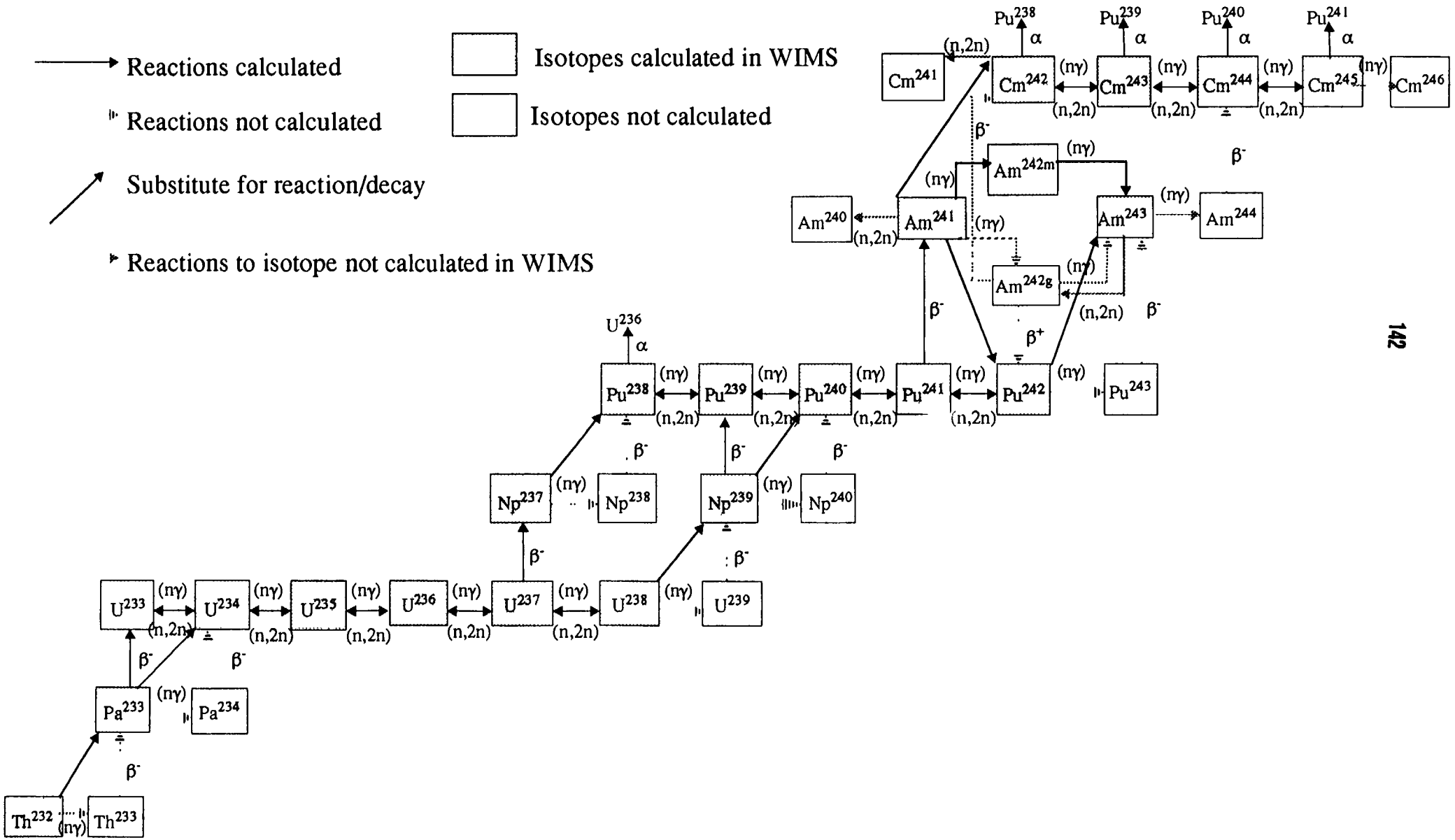
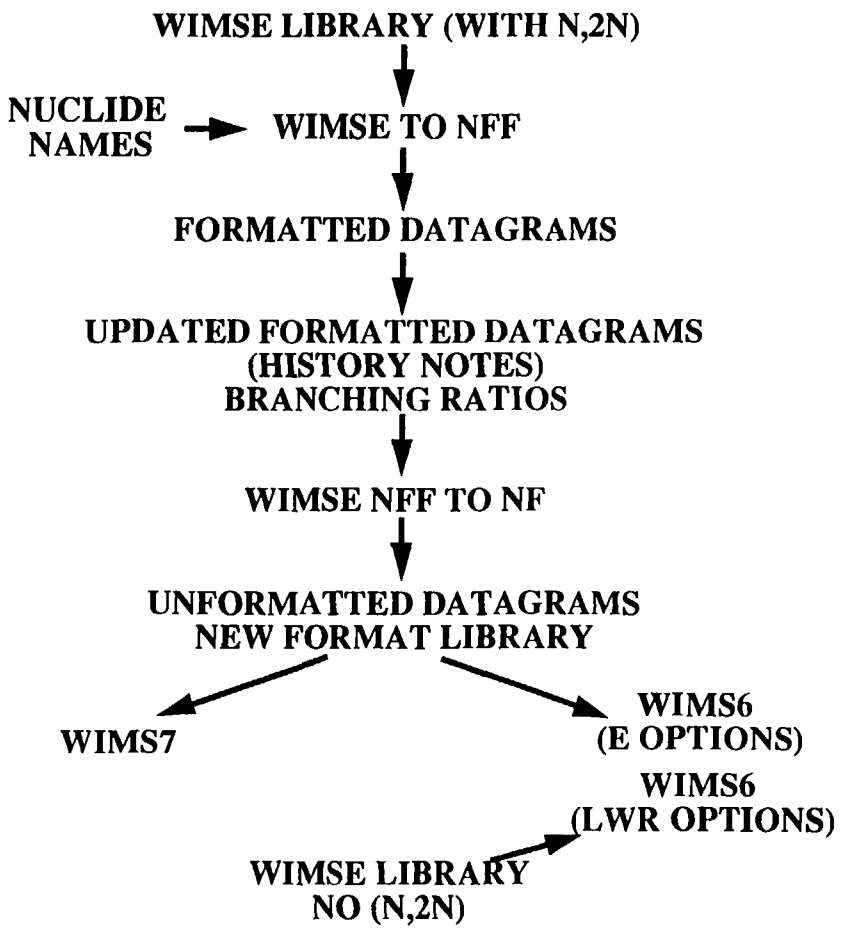
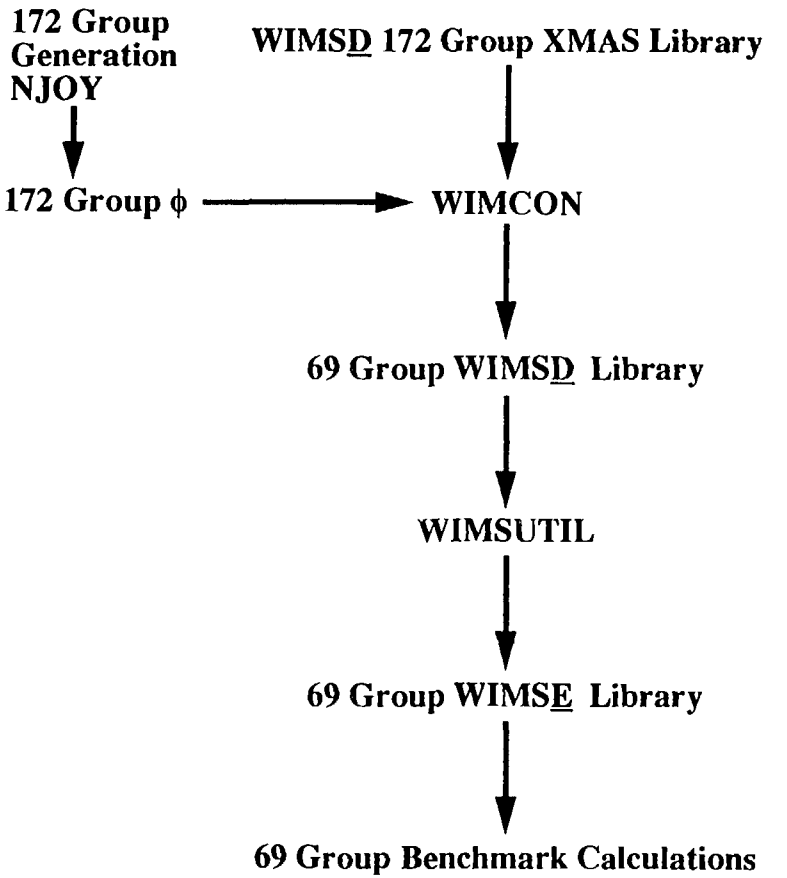
WIMS BURNUP ROUTE



NEW FORMAT LIBRARY



GENERATION OF 69 GROUP DATA



WILT CAPABILITIES
(WIMS Improved Library Translation)

(1) Create a WIMSD library.

(2) Add to an existing WIMSD library.

(3) Create a WIMSE library.

(4) Add to an existing WIMSE library.

(5) Have a flexible group structure.

(6) Produce P1 data for any isotope (WIMSE only).

(7) Delete isotopes from an existing library.

(8) Handle burnup data.

(9) Mix nuclides.

A NEW FORMAT FOR THE WIMS LIBRARY

- WIMS D and WIMS E library formats - sequential
- New format - random access, on disc or RAM
- Datagram - labelled block of data items, all of same type, e.g. real, integer, character *16.
- Datagram accessed by quoting its label. Gives access to type, size, address.
- Can then read all, or just a segment of the datagram.

QUALITY ASSURANCE

GENERAL

- Compare physics parameters from evaluation and library.

VALIDATION

- Resonance Shielding
- Optimise tabulation of resonance integrals (U238)
- Compare WIMS results with reference codes
  - homogeneous media.

VERIFICATION

- JEF2 Benchmarking.
- LWPC - UK - French methods and data.

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QUALITY ASSURANCE

PHYSICS CHECKS

- 2200 m/s Cross-section $\sigma_{0.0253eV}$
- Maxwellian Average Cross-section

$$\int_{1.1\times10^{-4}}^{4eV} \sigma_E \phi_E dE \;/\; \int_{1\,1\times10^{-4}}^{4eV} \phi_E dE$$

with $\phi_E = e^{-\frac{E}{Kt}} E$, (Kt = 0.02523eV)

- Resonance Integral (RI) $\int_{0.625eV}^{821KeV} \sigma_E \frac{1}{E} dE$
- Fission Spectrum Average Cross-section

$$\int_{914eV}^{19.64MeV} \sigma_E \phi_E dE \;/\; \int_{914eV}^{19.64eV} \phi_E dE$$

$E > 821KeV \quad \phi_E = e^{-E/Kt} E^{1/2} \quad (Kt = 1.13539MeV)$

$E < 821KeV \quad \phi_E = \frac{1}{E}$

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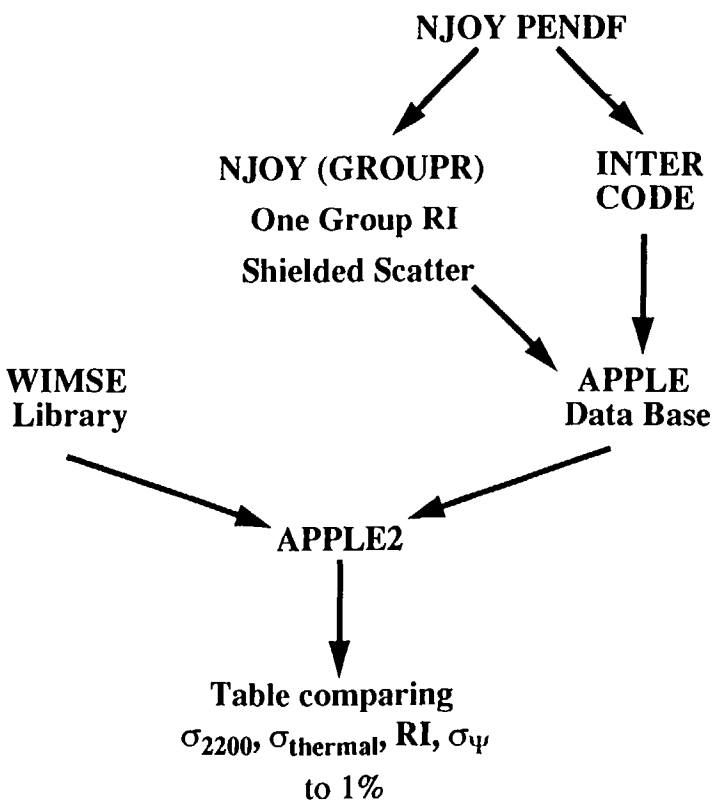
QUALITY ASSURANCE



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TABULATION OF RESONANCE INTEGRALS

Problem

- $RI = f(\sigma_p) = f(\text{composition} + \text{geometry})$.
- Tabulated (for each group) at fixed σ_p values.
- Interpolation linear in $\sqrt{\sigma_p}$.
- Limit to 10 σ_p values.
  (To reduce potential size problems in sub-group fitting and NJOY dimensions)
- Define σ_p values for tabular points.

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TABULATION OF RESONANCE INTEGRALS

Solution

- Work with U^{238}

- Form RI at many points (134).
  Requires many (14) NJOY runs as 10 σ_p limit.

- Establish region of interest.
  Run WIMS calculations and define "look-up" σ_p values.

- Write SigmaP Optimisation Code (SPOC)

- Adjust results for optimisation fit

Graph of NJOY Resonance Integral Values with WIMS Points Shown

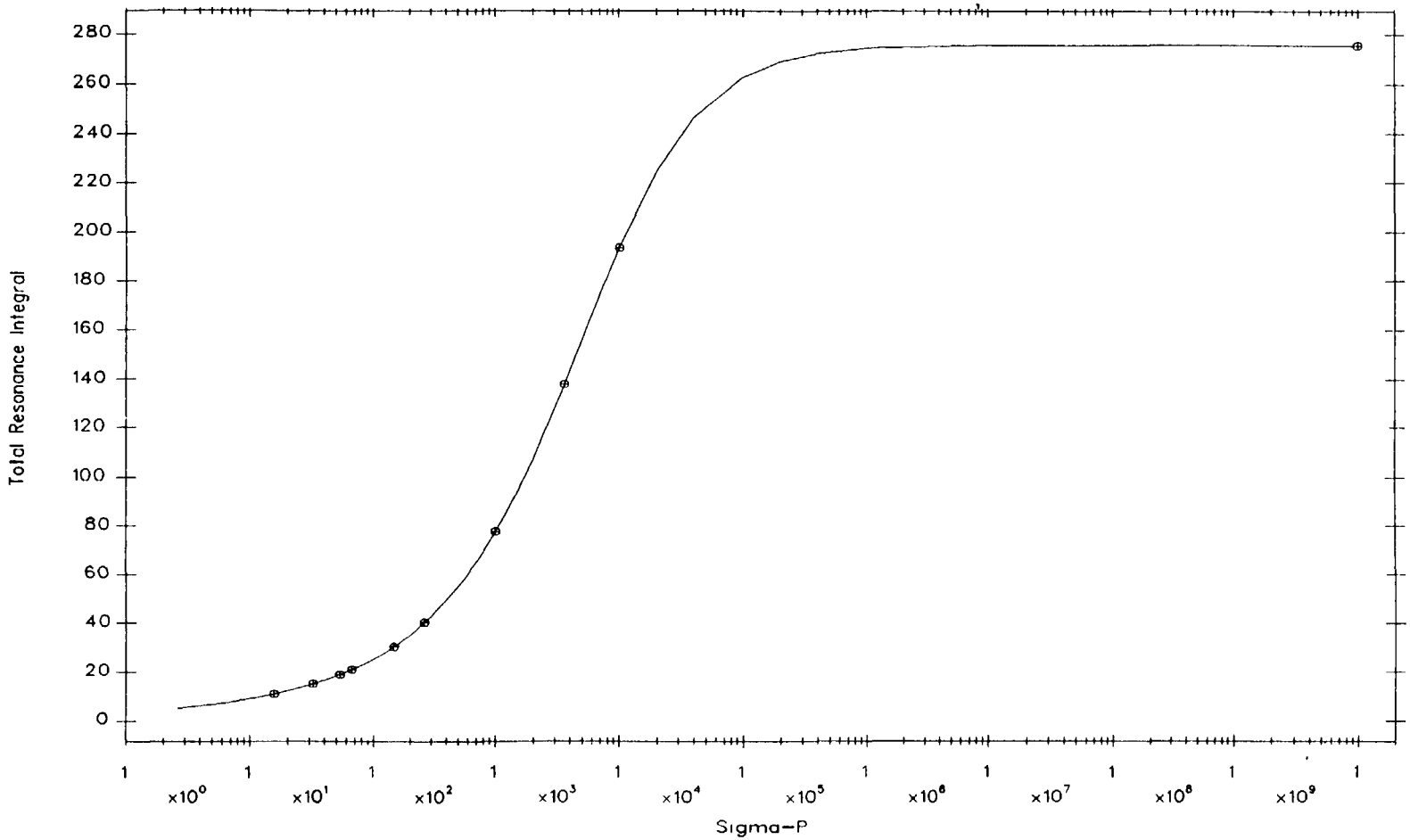

Figure 1. Graph of NJOY Values With WIMS Points Shown

Comparison of Errors for WIMS and Optimised Sigma-P Values

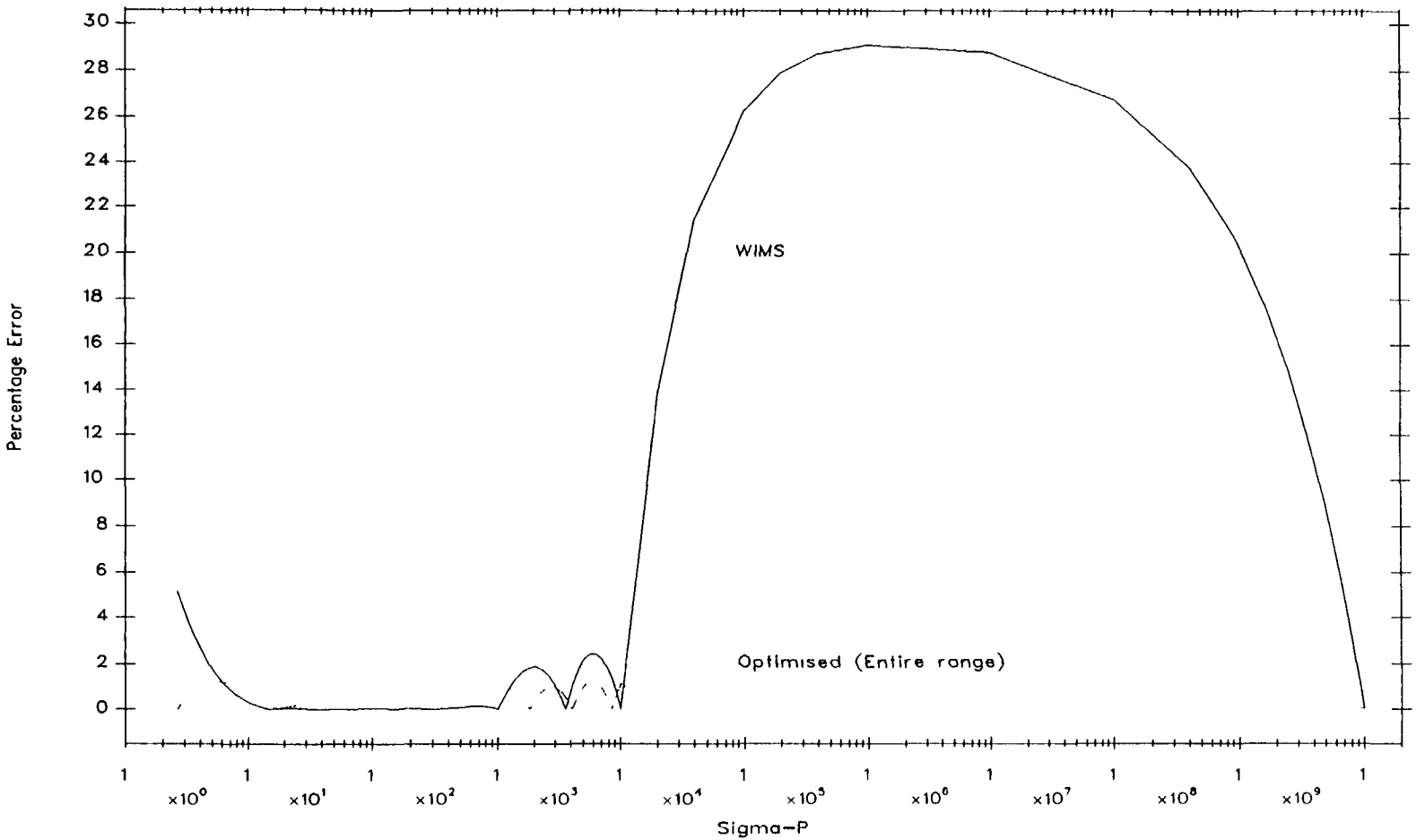

Figure 3. Comparison of Errors for WIMS and Optimised Sigma-P Values

Comparison of Errors for WIMS and NEW Sigma-P Values

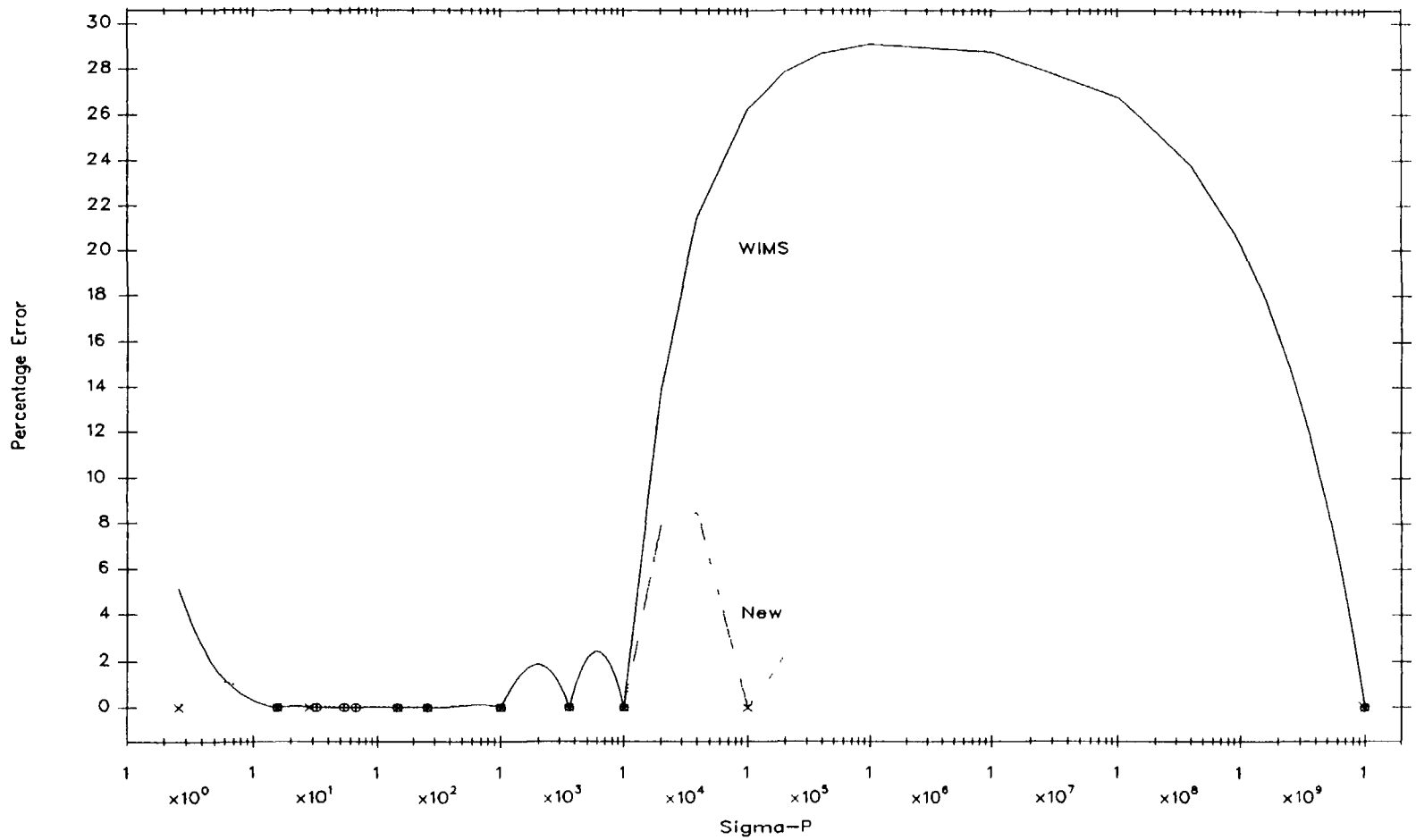

Figure 4. Comparison of Errors for WIMS and NEW Sigma-P Values

Region of Interest for Reactor Systems

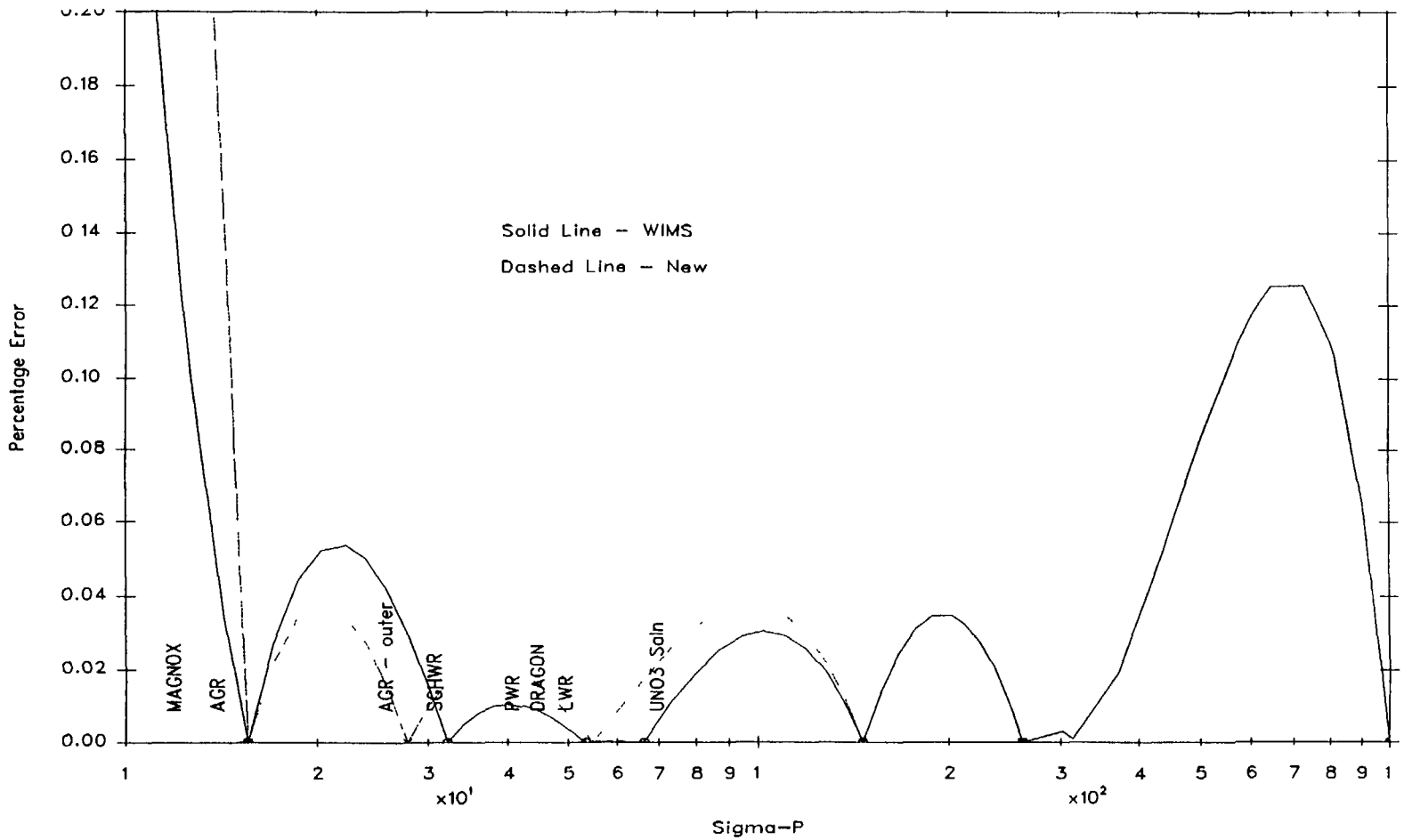

Figure 5. Region of Interest for Reactor Systems

Resonance Self Shielding

WIMS σ_p Values	New σ_p Values
	2.63421
15.5042	15.042
	27.8842
32.1642	
53.3742	
66.1742	
146.174	146.174
261.284	261.284
999.974	999.974
3599.97	3599.97
9999.97	9999.97
	1.00002E+05
	1.00000E+10
1.00000E+11	

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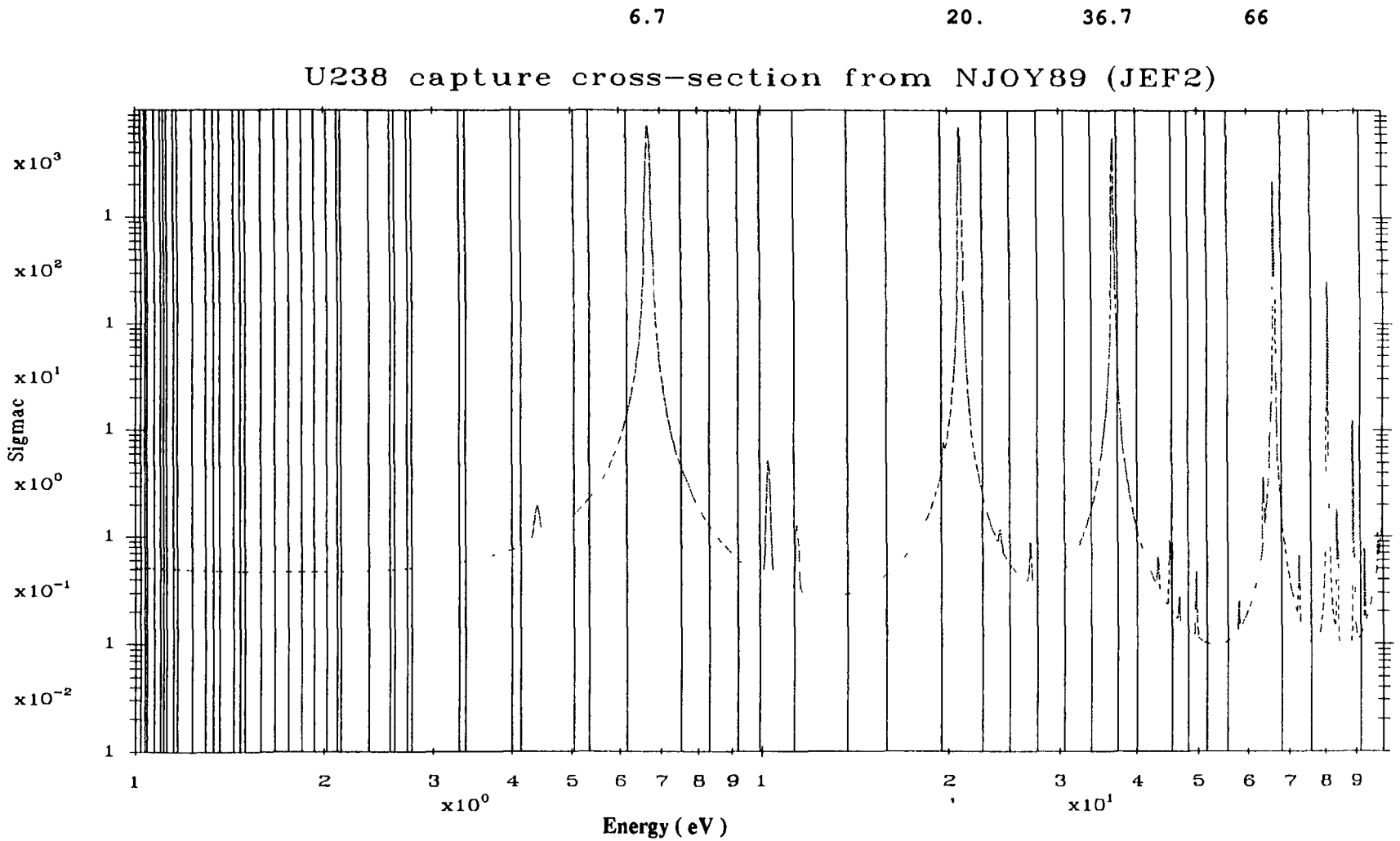
VALIDATION OF RESONANCE SHIELDING

U^{238} Absorption Reaction Rates (σ_p = 53.3742)					
Energy (eV)	WIMS	Diff %			St.Dev %
		MONK6/ WIMS	ECCO/ WIMS	ECCO/ MONK6	
75.67	1.98	3.9	1.2	-2.8	1.1
67.90	43.44	0.9	1.1	0.3	0.2
55.60	0.38	5.5	2.2	-3 4	2.6
51.58	0.47	-2.3	1.2	3.4	2.2
48.25	0.49	0 2	1.5	1.2	2 2
45.52	3.14	3.0	1.6	-1.5	0.9
40.17	19.06	7.6	7.4	-0.2	0.4
37.27	84.31	-3.4	-4.6	-1.2	0.2
33.72	4.01	2.4	0.9	-1.5	0.8
30.51	1.86	-0.3	1.2	1.5	1.1
27 61	2.14	3.4	1.4	-2.0	1.1
24.98	5.30	2.1	2.1	-0.0	0.7
22.60	125.08	0.9	0.5	-0.5	0.1
19.45	10.08	2.1	0.7	-1.4	0.5
15.93	2.30	2.6	1.5	-1.2	1 0
13.71	3.01	2.2	1.4	-0.8	0.9
11.22	4 62	2.0	1.5	-0.5	0.7
9.91	1 64	-1.0	1.2	2.2	1.2
9.19	3.76	0.9	2.6	1.7	0 8
8.32	10.13	1.0	2.4	1.4	0.5
7.52	203.32	-0.4	0.0	0 4	0.1
6.16	26.13	-1.2	0.9	2.0	0.3
5.35	4.22	0.6	1.9	1.3	0.8
5.04	9 25	-0.8	1.0	1 7	0.5
4.13	**0.97**	-2.2	1 7	3.8	1.5
4 00					

GROUP BOUNDS

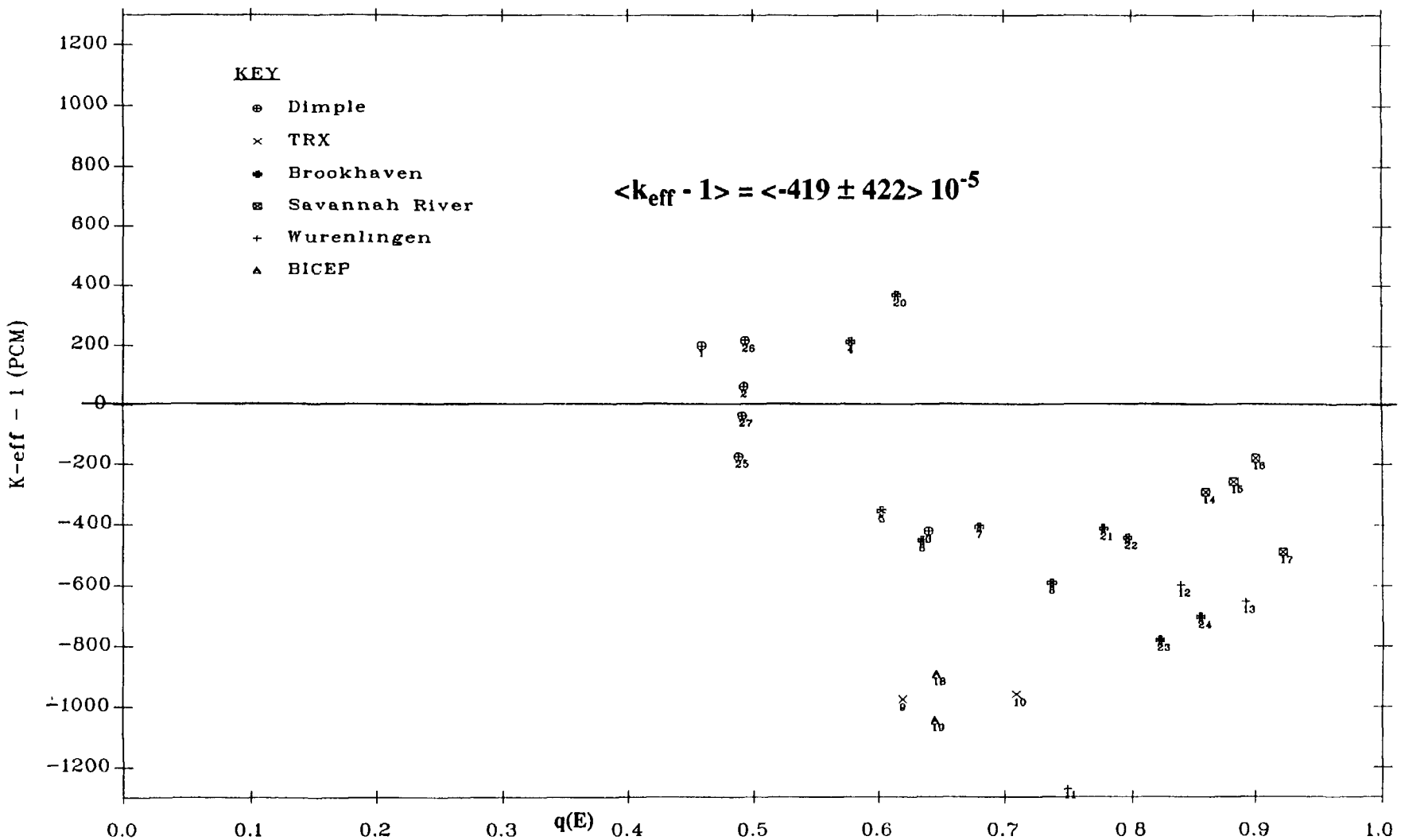


VALIDATION OF RESONANCE SHIELDING

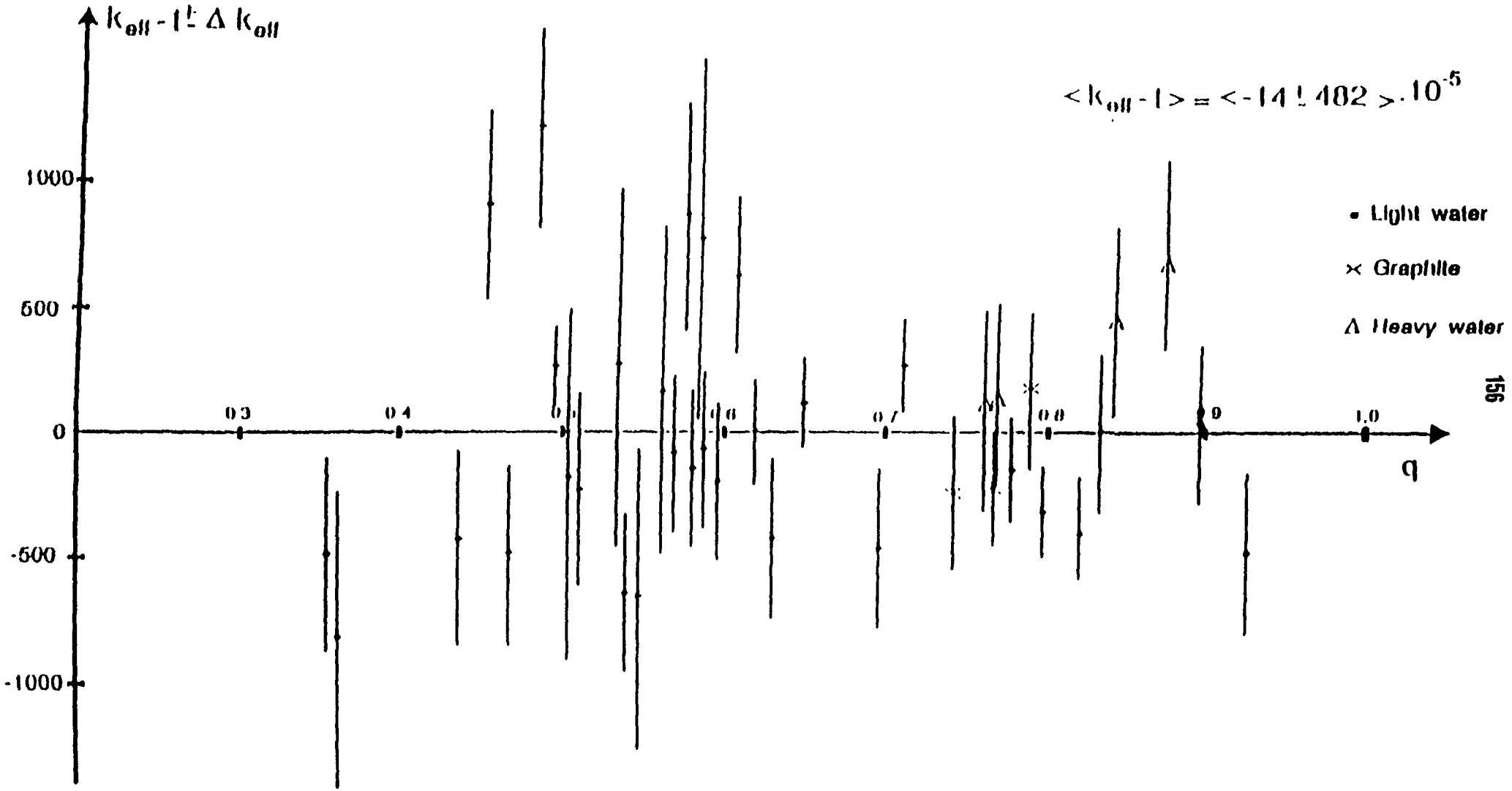
Computer Run Times for Generating Detailed Reaction Rates (U235)

CODE	Computer	Run Time
WIMSE	SUN 4	40s
ECCO	SUN 4	33m 59s
MONK	MEIKO	29h 49m

PLOT OF (K-EFF - 1) AGAINST q (2.6 eV)FOR JEF2.2 BENCHMARKS



APOLLO BENCHMARKING FOR SIMPLE GEOMETRIES



CONTENT OF THE 172 GROUP "XMAS" LIBRARY

- 140 nuclides.
- Isotopes rather than elements.
- Resonance self shielding for 34 substances.
- Multi-temperature self shielding for 5 substances.
- Multi-temperature thermal data for 4 substances.
- Bound data for 3 substances.

CONCLUSIONS

- 172 group library generated from JEF2 for WIMSD, WIMSE, LWRWIMS.
- Validated and used in benchmarking
- Being extended for Practical Reactor Calculations in WIMS6 (preliminary version). March 1994
- Further benchmarking JEF2.2 library for customer use. March 1995
- Possible adjustments to library resulting from user experience or JEF3 development. March 1996

Some Aspects of Preparation and Testing of Group Constants

Group Constant System ABBN-90

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Abstract

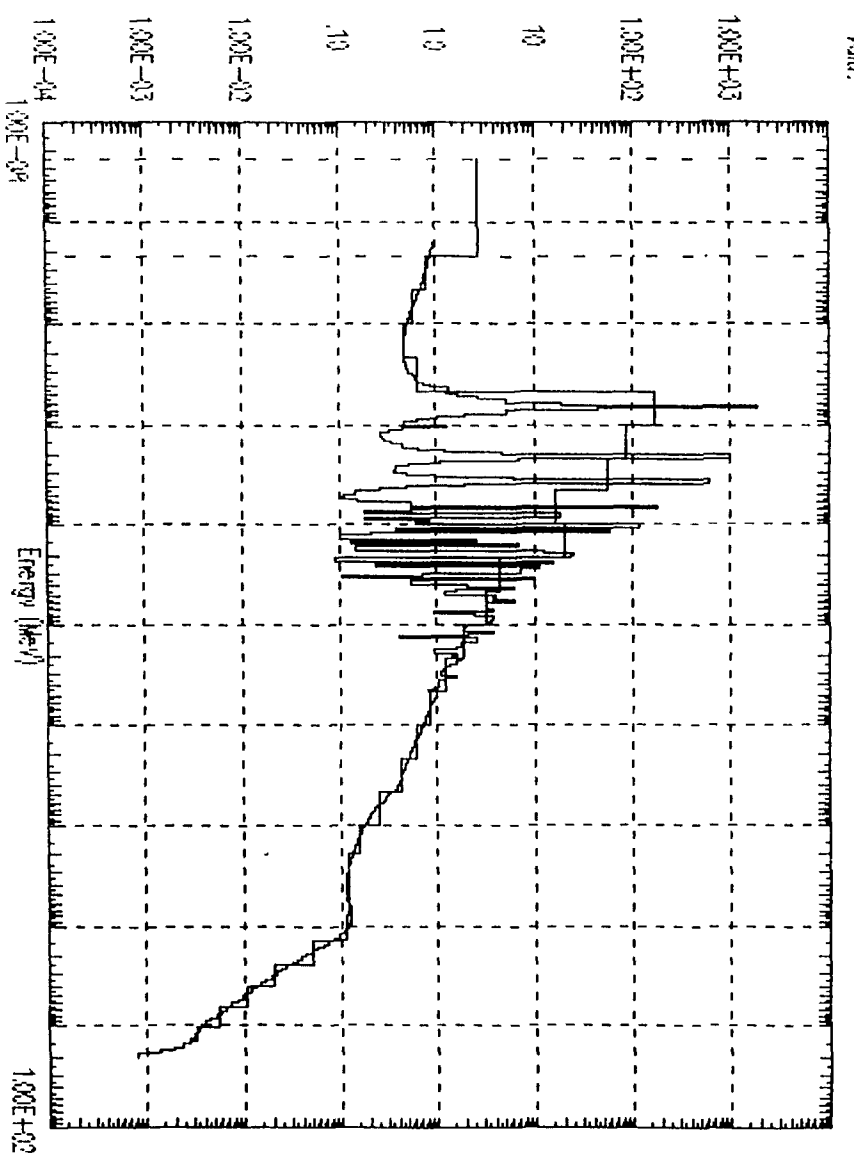
This paper presents an overview of activities performed to prepare and test the group constants ABBN-90. The ABBN-90 set is designed for application calculations of fast, intermediate and thermal nuclear reactors. The calculations of subgroup parameters are discussed. The processing code system GRUCON is mentioned in comparison to the NJOY code system. Proposals are made for future activities.

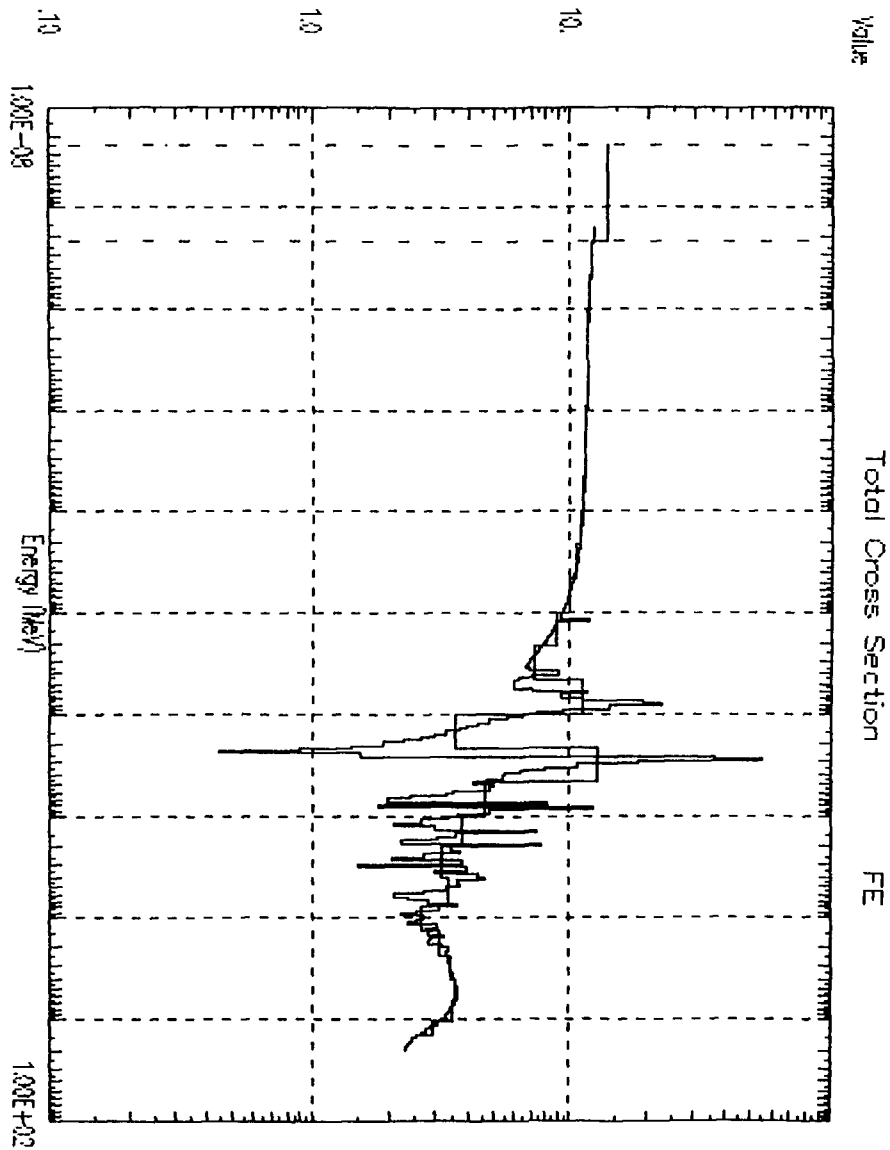
NUCLEAR DATA SET
ABBN-90

GROUP CONSTANT UNIVERSAL SYSTEM
FOR ENGINEERING CALCULATIONS OF:

- nuclear fast, intermediate and thermal reactors;
- nuclear safety;
- radioactive shielding;
- electro-nuclear and thermonuclear installations,
- radio-nuclide production.

Number of the neutron groups: 28 (0 0001 eV - 15 MeV)
300 (0.0001 eV - 20 MeV)
Number of the photon groups 15 (10 KeV - 11 MeV)





ABBN-90 CONSTANT SET STRUCTURE.

- text library of the obvious tables;
- main parameters for table:

NAM - name of nuclide;
MF - type of data.

MF	DESCRIPTION:
1	- main neutron cross-sections;
2	- inelastic scattering transition matrixes;
3	- elastic scattering transition matrixes;
4	- resonance self-shielding factors;
5	- it's Doppler increments;
6	- subgroup resonance structure parameters;
7	- Westkott's g-factors;
301-307	- multigroup neutron data;
8	- KERMA-factors;
9	- neutron cross-sections;
10	- photon production matrixes;
11	- photon interaction data;
61	- delayed neutron data;
90	- radio-nuclide decay data;
91	- radioactive decay photon spectra.

```
*
* revision: 1) 0 group cross sections was corrected
*              corresponding with U238V78 data.
*           2) Self Shielding Factor table (MF=4,5) was
*              obtained from subgroup parameters table.
*              In 17 group doppler increment was corrected
*              same as in BNAB78.
*           3) 12, 13, and 14 group constant set (MF=1 MT=0)
*              was taken from multygroup data.
*
NAM=U238      BIB=FOND  MF=  1  MT=  0  AWR=2.36006E+02
              LT =  28  LC= 10  LS= 10  LF = (I4,E7.0,2E9.0,E7.0,5E6.0)
*                          92-U -238
*                    MAIN NEUTRON CONSTANTS
* NG   total   capture   fission  elast  inel  mult    nu     mu    ksi
*
  -1    5.80    0.0027    1.1892   2.92 1.696 2.147 4.538 .8615 .0012
   0    5.77    0.0033    0.9992   2.87 1.906 1.797 4.087 .8277 .0015
*
   1    6.46    0.0056    0.9424   3.49 2.021 1.369 3.513 .7866 .0018
   2    7.55    0.0107    0.5733   4.33 2.638 1.001 3.115 .7673 .0020
   3    7.76    0.0206    0.5380   4.24 2.965 1.000 2.811 .7363 .0022
   4    7.12    0.0489    0.4651   3.84 2.769 1.000 2.642 .5510 .0038
*
   5    7.11    0.1099    0.0396   4.66 2.301 1.000 2.546 .4506 .0046
   6    8.14    0.1154    0.0011   6.33 1.697 1.000 2.493 .3577 .0054
   7    9.79    0.1185    0.0001   8.46 1.211 1.000 2.434 .2201 .0064
   8   11.35    0.1500    0.0000  10.44 0.756 1.000 2.397 .1209 .0074
*
   9   12.54    0.2424    0.0000  11.99 0.303 1.000 2.378 .0586 .0080
  10   13.63    0.4335    0.0000  13.19 0.000 1.000 2.369 .0248 .0083
  11   14.88    0.6107            14.27             2.364 .0100 .0084
*
  12   16.60     .8646            15.74             2.362 .0028 .0084
  13   19.86    1.2458            18.61             2.361 .0028 .0084
  14   21.84    1.8521            19.98             2.361 .0028 .0084
*
  15   22.28    3.3105            18.96             2.361 .0028 .0084
  16   21.88    4.5296            17.35             2.361 .0028 .0084
  17   89.09   20.2163            68.88             2.360 .0028 .0084
*
  18   41.00   16.5577            24.44             2.360 .0028 .0084
  19  143.40   54.1026            89.30             2.360 .0028 .0084
  20  126.47   83.8098            42.66             2.360 .0028 .0084
*
  21  189.36  169.7841            19.58             2.360 .0028 .0084
  22    8.87    0.6553             8.21             2.360 .0028 .0084
  23    9.14    0.4814             8.66             2.360 .0028 .0084
*
  24    9.39    0.5938             8.80             2.360 .0028 .0084
  25    9.67    0.8155             8.85             2.360 .0028 .0084
  26   11.61    2.7100             8.90             2.360 .0028 .0084
*
```

```
NAM=U238      BIB=FOND  MF=  2  MT=  0  AWR= 2.36006+ 2
              LT =  12  LC= 14  LS= 14  LF = (I4,13E5.0)
*                     INELASTIC TRANSITION MATRIX
*                     ZERO MOMENT (TRANSITIONS  G -> K)
* G/K  -1     0     1     2     3     4     5     6     7     8     9    10    11
*
  -1 .161 .047 .000 .010 .099 .519 .918 1.05 .549 .203 .065 .016 .004
   0      .216 .033 .008 .098 .527 .881 .926 .467 .179 .064 .018 .008
*
   1           .274 .035 .059 .304 .571 .731 .469 .214 .081 .022 .007
   2                .315 .092 .406 .670 .677 .321 .112 .036 .009 .003
   3                     .370 .253 .725 .893 .476 .175 .056 .013 .004
   4                          .635 .509 .924 .466 .164 .054 .013 .004
*
   5                               1.13 .513 .404 .192 .047 .011 .004
   6                                    1.34 .320 .007 .021 .007 .002
   7                                         .860 .346 .005
   8                                              .392 .363 .001 .000
*
   9                                                   .057 .168 .078
  10                                                              .000
*
NAM=U238      BIB=END6  MF=  2  MT=  1  AWR= 2.36006+ 2
              LT =  12  LC= 14  LS= 14  LF = (I4,13E5.0)
*                     INELASTIC TRANSITION MATRIX
*                     FIRST MOMENT (TRANSITIONS  G -> K)
* G/K  -1     0     1     2     3     4     5     6     7     8     9    10    11
*
  -1 .034 .068 .014 .025 .020 .022 .018 .010 .002 .000 .000 .000 .000
   0      .078 .051 .010 .017 .015 .016 .009 .003 .000 .000 .000 .000
*
   1           .098 .049 .010 .005 .011 .007 .003 .001 .000 .000 .000
   2                .105 .047 .028 .012 .005 .001 .000 .000 .000 .000
   3                     .084 .051 .029 .010 .003 .000 .000 .000 .000
   4                          .062 .031 .020 .002 .000 .000 .000 .000
*
   5                               .056 .011 .006 .000 .000 .000 .000
   6                                    .058 .003 .000 .000 .000 .000
   7                                         .028 .001 .000 .000 .000
   8                                              .008 .001 .000 .000
*
   9                                                   .002 .000 .000
  10                                                              .000
*
```

NAM=U238 BIB=FOND MF= 3 MT= 0 AWR=2.36006E+02
LT = 14 LC= 7 LS= 7 LF = (I4,6E11.0)
* ANGULAR MOMENTA OF ELASTIC TRASITIONS
* FROM GROUP g TO THE SAME GROUP

G/L	0	1	2	3	4	5
-1	0.9375	0.8141	0.6931	0.5782	0.4889	0.4205
0	0.9861	0.8252	0.6878	0.5850	0.4937	0.4030
1	0.9885	0.7849	0.6469	0.5521	0.4552	0.3502
2	0.9928	0.7664	0.6255	0.5181	0.4056	0.2937
3	0.9933	0.7363	0.5801	0.4494	0.3302	0.2098
4	0.9919	0.5516	0.3649	0.2935	0.1716	0.0552
5	0.9902	0.4516	0.2572	0.1529	0.0642	0.0019
6	0.9899	0.3593	0.1539	0.0490	0.0178	-.0033
7	0.9884	0.2228	0.0680	0.0064	0.0039	-.0038
8	0.9881	0.1242	0.0240	-.0024	0.0034	-.0023
9	0.9888	0.0621	0.0087	-.0015	0.0019	-.0003
10	0.9889	0.0284	0.0033	-.0006	0.0005	-.0000
11	0.9886	0.0137	0.0008	-.0002	0.0001	-.0000
12	0.9890	0.0065	0.0000	-.0000	-.0000	0.0000

NAM=U238 BIB=FOND MF= 3 MT= 1 AWR=2.36006E+02
LT = 14 LC= 7 LS= 7 LF = (I4,6E11.0)
* ANGULAR MOMENTA OF ELASTIC TRASITIONS
* FROM GROUP g TO THE NEXT GROUP

G/L	0	1	2	3	4	5
-1	0.0625	0.0474	0.0415	0.0358	0.0297	0.0255
0	0.0139	0.0025	0.0002	0.0008	0.0010	0.0003
1	0.0115	0.0017	0.0010	0.0016	0.0015	0.0013
2	0.0072	0.0009	0.0008	0.0008	0.0008	0.0007
3	0.0067	-.0000	0.0006	0.0004	0.0004	-.0000
4	0.0081	-.0006	-.0004	0.0006	-.0000	-.0005
5	0.0098	-.0010	-.0000	-.0003	0.0000	-.0003
6	0.0101	-.0016	-.0003	-.0004	0.0000	-.0001
7	0.0116	-.0027	-.0004	-.0003	0.0001	-.0001
8	0.0119	-.0033	-.0003	-.0001	0.0001	-.0000
9	0.0112	-.0035	-.0001	-.0000	0.0000	0.0000
10	0.0111	-.0036	-.0000	0.0000	0.0000	0.0000
11	0.0114	-.0037	-.0000	0.0000	0.0000	0.0000
12	0.0110	-.0037	-.0000	0.0000	-.0000	-.0000

NAM=U238 BIB=FOND MF= 4 MT= 1 AWR=2.36006E+02
LT = 14 LC= 18 LS= 18 LF = (I4,I2,1X,16I4)
* RESONANCE SELFSHIELDING FACTORS
* TRANSPORT

* G	N	0	.1	...	...	1	...	...	10	...	...	100	215	465	1000...*10		
8	0	955	955	956	956	958	960	964	970	976	981	985	986	987	988	988	988
9	0	917	917	917	918	920	924	932	942	955	967	975	980	982	983	984	984
10	0	880	881	881	882	885	890	900	916	938	961	978	988	994	997	998	999
11	0	797	797	798	799	802	807	818	838	871	911	946	971	985	993	996	998
12	0	704	705	705	706	707	711	718	734	766	816	876	927	962	981	991	995
13	0	574	575	575	576	578	581	588	600	622	661	722	801	876	931	965	983
14	0	205	224	243	277	325	372	411	445	485	535	597	672	761	847	913	955
15	0	349	353	357	365	376	391	407	424	448	482	530	599	691	792	877	935
16	0	480	480	481	484	487	494	502	513	527	546	575	619	684	766	849	915
17	1	72	81	89	99	110	121	133	148	175	223	301	416	565	717	839	916
18	1	203	208	212	218	225	234	244	260	285	319	368	452	584	730	847	922
19	1	60	62	64	67	72	78	86	95	108	127	155	197	273	410	590	757
20	1	63	63	63	64	65	66	68	74	86	105	134	185	289	459	650	804
21	1	56	57	58	59	61	65	72	82	96	118	156	224	350	529	709	842

*

NAM=U238 BIB=FOND MF= 4 MT=102 AWR=2.36006E+02
LT = 14 LC= 18 LS= 18 LF = (I4,I2,1X,16I4)
* CAPTURE

* G	N	0	.1	...	...	1	...	...	10	...	...	100	215	465	1000...*10		
8	0	984	984	984	985	985	986	988	991	994	996	998	999	999	999	999	999
9	0	963	963	964	964	965	968	972	978	985	991	995	997	998	999	999	999
10	0	929	929	930	931	932	936	943	954	967	980	989	994	997	998	999	999
11	0	853	853	854	855	858	864	876	894	920	947	969	984	992	996	998	999
12	0	730	730	731	733	736	743	756	781	820	870	918	954	976	988	994	997
13	0	569	569	570	571	574	579	590	611	648	707	782	857	918	956	978	989
14	0	267	270	274	280	290	306	332	374	437	520	619	726	825	900	947	974
15	0	170	171	172	174	178	186	203	234	286	364	469	596	729	838	913	956
16	0	126	126	127	128	131	136	147	168	205	267	357	476	616	750	855	924
17	1	42	45	48	52	60	74	98	136	196	283	402	547	697	820	904	953
18	1	47	48	49	52	59	71	94	131	188	267	377	520	677	810	898	949
19	1	33	34	35	36	39	45	54	70	95	135	195	287	422	586	742	858
20	1	18	18	19	20	23	29	39	57	87	134	206	318	474	646	792	890
21	1	27	28	29	32	37	45	59	81	115	171	256	381	542	704	832	913

*

NAM=U238 BIB=FOND MF= 4 MT= 2 AWR=2.36006E+02
LT = 14 LC= 18 LS= 18 LF = (I4,I2,1X,16I4)
* ELASTIC

* G	N	0	.1	...	...	1	...	...	10	...	...	100	215	465	1000...*10		
8	0	980	980	981	981	982	983	985	989	992	995	997	998	999	999	999	999
9	0	961	961	961	962	963	966	970	977	984	990	995	997	998	999	999	999
10	0	928	928	928	929	931	935	942	953	967	979	989	994	997	998	999	999
11	0	862	862	863	864	867	873	883	901	925	951	972	985	993	996	998	999
12	0	770	770	771	772	775	781	793	814	848	891	931	962	980	990	995	997
13	0	638	638	639	640	643	648	659	678	711	761	822	883	933	964	982	991
14	0	457	462	467	475	487	503	524	554	594	647	712	785	856	915	954	977
15	0	487	488	489	492	497	504	517	536	566	609	666	738	816	886	937	967
16	0	634	635	635	636	638	642	649	659	674	697	729	773	827	883	930	962
17	1	135	141	146	154	165	180	203	238	294	376	484	610	739	845	916	958
18	1	356	361	364	371	380	392	409	434	468	517	586	680	784	872	932	966
19	1	103	106	108	113	120	130	143	160	186	225	282	365	486	632	771	874
20	1	167	167	168	169	172	177	185	201	227	268	330	425	556	701	824	907
21	1	435	436	438	440	444	451	462	477	498	530	579	649	740	832	904	950

*

```
NAM=U238      BIB=FOND  MF=  5  MT=  1  AWR=2.35955E+02
              LT =  28  LC= 18  LS= 18  LF = (I4,I2,1X,1614)
*                     DOPPLER INCREMENTS
*                     F(900K)-F(300K) and F(2100K)-F(900K)
*                     TRANSPORT
*  G N    0   .1  ...  ...   1  ...  ...   10  ...  ... 100 215 465 1000...*10
*
   8 0   13  13  13  13  12  11  10   7   5   3   1   0   0   0   0   0
          8   8   7   7   7   6   5   4   3   1   0   0   0   0   0   0
   9 0   25  25  25  24  24  22  20  16  11   6   3   1   0   0   0   0
         15  15  15  15  15  14  12   9   6   4   2   1   0   0   0   0
  10 0   39  39  39  38  38  36  34  29  22  14   8   4   2   0   0   0
         27  26  26  26  25  24  22  19  14   9   5   2   1   0   0   0
  11 0   49  49  49  49  49  48  47  43  36  26  16   9   4   2   1   0
         36  36  36  36  36  35  34  31  25  17  10   5   2   1   0   0
  12 0   51  51  51  51  51  51  51  50  47  40  29  18   9   5   2   1
         39  39  39  39  39  39  38  37  35  29  20  12   6   3   1   0
  13 0   43  43  43  43  43  43  43  44  44  44  40  32  21  12   6   3
         33  33  33  33  33  33  33  34  34  33  30  23  15   8   4   2
  14 0   96  88  81  68  52  40  34  33  35  40  48  52  47  35  21  11
         81  76  70  61  49  38  32  30  31  35  41  44  39  27  16   7
  15 0   45  43  41  37  31  26  22  20  22  30  46  62  66  54  36  20
         41  41  40  38  35  32  29  28  28  31  40  50  51  40  26  14
  16 0   24  23  22  21  18  15  11   9  10  16  28  43  54  54  42  27
         17  17  17  17  17  17  16  16  17  19  26  36  43  40  29  17
  17 1    2   2   2   2   1   2   3   4  16  28  41  52  63  74  66  57
          5   5   4   4   3   4   6   7  22  37  52  54  56  58  52  45

  18 1    0   0   0   0   0   0   0   1   7  26  55  78  80  62  39  20
          3   4   4   5   6   7   8   8   8  17  41  68  73  56  34  18
  19 1    0   0   0   0   0   0   0   0   0   0  10  38  69  84  74  50
          0   0   0   0   0   0   0   0   1   1   5  28  64  87  77  51
  20 1    0   0   0   0   0   0   0   0   0   4  27  60  85  87  67  41
          0   0   0   0   0   0   0   0   0   0  14  51  91 100  76  45
  21 1    0   0   0   0   0   0   0   0   0   0  12  43  76  80  55  25
          0   0   0   0   0   0   0   0   0   5  15  32  44  33   5   0
*
*
NAM=U238      BIB=FOND  MF=  5  MT=102  AWR=2.35955E+02
              LT =  28  LC= 18  LS= 18  LF = (I4,I2,1X,1614)
*                     CAPTURE
*  G N    0   .1  ...  ...   1  ...  ...   10  ...  ... 100 215 465 1000...*10
*
   8 0    7   7   7   7   7   6   5   4   2   1   0   0   0   0   0   0
          4   4   3   3   3   3   2   2   1   0   0   0   0   0   0   0
   9 0   16  16  16  16  15  14  12  10   7   4   2   1   0   0   0   0
          9   9   9   8   8   7   6   5   3   2   1   0   0   0   0   0
  10 0   32  32  32  31  31  29  26  21  15   9   5   2   1   0   0   0
         18  18  18  18  17  16  14  12   8   5   2   1   0   0   0   0
  11 0   55  55  55  54  53  51  47  41  31  21  12   6   3   1   0   0
         36  36  36  35  34  33  30  26  19  13   7   3   1   0   0   0
  12 0   78  78  78  78  77  75  72  66  56  41  27  15   8   3   1   0
         56  56  55  55  55  53  51  46  38  28  17   9   5   2   1   0
  13 0   87  87  87  87  87  86  85  82  76  66  51  35  20  11   5   2
         65  65  65  65  65  64  63  60  56  47  36  24  14   7   3   1
  14 0   72  71  70  69  69  69  71  76  83  88  85  71  50  30  16   7
         70  70  70  69  69  69  69  70  72  71  65  52  35  19   9   3
  15 0   40  41  41  42  43  45  49  57  70  87  99  96  77  51  30  16
         54  54  54  54  55  56  58  62  69  78  84  78  60  39  23  13
  16 0   25  25  25  26  27  29  32  39  51  68  85  94  88  67  43  23
         35  35  35  36  36  37  40  44  51  62  73  77  69  50  30  15
  17 1    4   3   3   3   3   4   7  14  28  47  62  63  51  34  18   8
          0   3   5   7  11  16  23  32  44  59  71  71  58  39  23  13
```

```
NAM=U238      BIB=FOND  MF=  6  MT=  0  AWR=2.36006E+02
              LT =  47  LC=  5  LS=  5  LF = (I4,E5.0,9E7.0)
*                    SUBGROUP PARAMETERS
* G share    capt   elast    fis
*
   9.3511 1.1732 1.2425 0.0
    .6489 0.9063 0.8688 0.0
  10.3567 1.3065 1.3758 0.0
    .6433 0.8301 0.7916 0.0
  11.1058 2.3591 2.4065 0.0
    .8607 0.8695 0.8518 0.0
    .0335 0.0600 0.3667 0.0
  12.0384 4.7587 5.1930 0.0
    .3874 1.5140 1.1184 0.0
    .5696 0.4021 0.6440 0.0
    .0046 0.3615 0.1105 0.0
  13.0247 8.9924 10.124 0.0
    .1361 3.3327 1.8650 0.0
    .8241 0.3889 0.6002 0.0
    .0151 0.2540 0.0961 0.0
  14.0198 15.058 16.941 0.0
    .0650 7.6782 2.8316 0.0
    .8920 0.2241 0.5374 0.0
    .0232 0.1254 0.0503 0.0
  15.0196 24.915 19.069 0.0
    .0581 7.0192 2.8172 0.0
    .8801 0.1134 0.5192 0.0
    .0422 0.0970 0.1334 0.0
  16.0116 49.981 25.209 0.0
    .0276 12.268 2.7856 0.0
    .7857 0.0933 0.7149 0.0
    .1751 0.0477 0.3940 0.0
  17.0213 37.891 34.171 0.0
    .0701 2.3471 1.8306 0.0
    .8161 0.0298 0.1732 0.0
    .0925 0.0443 0.0267 0.0
  18.0093 86.755 59.505 0.0
    .0191 7.9674 3.3303 0.0
    .7437 0.0484 0.4555 0.0
    .2279 0.0220 0.1943 0.0
  19.0142 62.687 56.127 0.0
    .0472 1.7148 1.7974 0.0
    .5564 0.0375 0.1737 0.0
    .3822 0.0210 0.0563 0.0
  20.0143 59.091 50.288 0.0
    .0182 6.8134 5.8890 0.0
    .0553 0.3659 0.4617 0.0
    .9122 0.0118 0.1624 0.0
  21.0272 30.739 17.956 0.0
    .0392 3.1657 2.0815 0.0
    .1900 0.1627 0.6167 0.0
    .7436 0.0120 0.4207 0.0
*
```

NAM=U238 BIB=BNAB MF= 10 MT= 1 AWR=0.00000E+00
LT = 28 LC= 18 LS= 9 LF = (I4,8F8.0)
* PHOTON PRODUCTION MATRIX
* ZERO MOMENT

* N/G	1	2	3	4	5	6	7	8
-1	0.0011	0.0027	0.0126	0.0415	0.1414	0.3674	0.6933	0.8096
0	0.0007	0.0023	0.0105	0.0339	0.1157	0.3216	0.6513	0.7951
1	0.0000	0.0007	0.0071	0.0291	0.1295	0.3315	0.6018	0.6868
2		0.0002	0.0029	0.0116	0.0497	0.2293	0.5011	0.5986
3		0.0002	0.0025	0.0090	0.0347	0.1065	0.2685	0.4141
4		0.0001	0.0021	0.0076	0.0310	0.0821	0.1825	0.2566
5		0.0000	0.0002	0.0010	0.0105	0.0246	0.0554	0.0529
6		0.0000	0.0000	0.0002	0.0089	0.0196	0.0454	0.0350
7		0.0000	0.0000	0.0002	0.0101	0.0222	0.0515	0.0393
8				0.0004	0.0166	0.0366	0.0847	0.0646
9				0.0012	0.0530	0.1164	0.2702	0.2060
10				0.0029	0.1319	0.2897	0.6724	0.5125
11				0.0029	0.1320	0.2890	0.6700	0.5120
12				0.0029	0.1310	0.2890	0.6690	0.5110
13				0.0029	0.1310	0.2880	0.6690	0.5100
14				0.0029	0.1310	0.2880	0.6690	0.5100
15				0.0029	0.1310	0.2880	0.6680	0.5100
16				0.0029	0.1310	0.2880	0.6680	0.5100
17				0.0029	0.1310	0.2880	0.6680	0.5100
18				0.0029	0.1310	0.2880	0.6680	0.5100
19				0.0029	0.1310	0.2880	0.6680	0.5100
20				0.0029	0.1310	0.2880	0.6680	0.5100
21				0.0029	0.1310	0.2880	0.6680	0.5100
22				0.0029	0.1310	0.2880	0.6680	0.5100
23				0.0029	0.1310	0.2880	0.6680	0.5100
24				0.0029	0.1310	0.2880	0.6680	0.5100
25				0.0029	0.1310	0.2880	0.6680	0.5100
26				0.0029	0.1310	0.2880	0.6680	0.5100

* N/G	9	10	11	12	13	14	15	EGAM
-1	1.6519	1.8877	1.1975	0.5834	0.1533	0.1201	0.0416	7.7447
0	1.5971	1.8040	0.9940	0.4200	0.0845	0.0871	0.0322	7.1596
1	1.3498	1.5730	0.9418	0.4610	0.1932	0.1495	0.0558	6.5370
2	1.0816	1.1631	0.6025	0.2560	0.1059	0.1006	0.0367	4.8384
3	0.9436	1.2404	0.7325	0.2924	0.1414	0.1282	0.0513	3.5675
4	0.7196	1.0988	0.7515	0.3247	0.1768	0.1393	0.0589	2.7599
5	0.2441	0.3677	0.0846	0.3486	0.6520	0.0049	0.0025	0.8665
6	0.0407	0.0817	0.0223	0.1782	0.7141	0.0023	0.0016	0.3993
7	0.0447	0.0648	0.0228	0.0964	0.7069	0.0025	0.0018	0.4160
8	0.0732	0.1064	0.0373	0.0916	0.6796	0.0041	0.0030	0.6475

NAM=U238 BIB= SAI MF= 11 MT= 0 AWR=0.00000E+00
LT = 15 LC= 9 LS= 9 LF = (I4,F7.3,2F8.3,5F9.2)

PHOTON INTERACTION CROSS SECTIONS

GR	SKN	SBD	SN	SPH	STT	ST	SA	SAE
1	4.746	4.746	15.145	0.33	20.19	20.19	18.66	12.95
2	5.583	5.583	12.880	0.44	18.93	18.93	17.02	12.12
3	6.627	6.627	10.791	0.61	18.04	18.04	15.68	11.46
4	7.673	7.673	9.046	0.81	17.59	17.52	14.69	10.90
5	8.918	8.918	7.263	1.11	17.37	17.25	13.78	10.41
6	10.741	10.726	5.064	1.71	17.67	17.49	12.89	10.12
7	13.221	13.123	2.698	2.92	19.00	18.78	12.63	10.63
8	15.973	15.855	0.866	5.34	22.56	22.09	14.06	12.66
9	19.647	19.504	0.095	11.58	32.27	31.18	20.24	18.58
10	25.899	25.459	0.000	46.73	76.08	71.99	55.64	47.17
11	35.143	33.229	0.000	357.78	410.89	390.78	366.08	218.77
12	44.007	37.705	0.000	970.59	1086.52	1009.64	978.52	590.21
13	50.296	36.670	0.000	3106.72	3387.35	3141.30	3108.69	2764.82
14	55.020	30.756	0.000	19578.06	20283.80	19578.06	19578.06	15159.77
15	57.793	22.249	0.000	34839.41	36456.70	34839.41	34839.41	33594.88

NAM=U238 BIB=END6 MF= 61 MT= 0 AWR= 2.36006+ 2
LT = 13 LC= 8 LS= 8 LF = (I4,7E8.0)

DELAYED NEUTRON DATA

NG	1	2	3	4	5	6	vd
	.0139	.1128	.1310	.3851	.2539	.1031	.0450
	.0136	.0313	.1233	.3237	.9059	3.0487	.0270
1	.0000	.0000	.0000	.0000	.0000	.0000	.0000
2	.0000	.0000	.0000	.0000	.0000	.0000	.0000
3	.0000	.0000	.0001	.0021	.0053	.0097	.0032
4	.0073	.0096	.0132	.0576	.0499	.0639	.0444
5	.1586	.1885	.0992	.1737	.1276	.1406	.1502
6	.2964	.3782	.3390	.3069	.2980	.2858	.3145
7	.2679	.2363	.2750	.2369	.2582	.2515	.2491
8	.1363	.0843	.1556	.1291	.1481	.1317	.1327
9	.0795	.0637	.0652	.0518	.0702	.0697	.0618
10	.0318	.0211	.0304	.0239	.0238	.0243	.0245
11	.0223	.0183	.0223	.0180	.0189	.0229	.0194

MT= 0 Z = 28.
LT = 32 LC= 25 LS= 7 LF =(I4,7E9.0)

*

NEUTRON CROSS - SECTIONS

*

	260540.	260560.	260570.	260580.	260000.	260540.	260540.
	1.90	1.90	1.90	1.90	1.90	1.90	1.90
	102.	102.	102.	102.	102.	16.	28.
	260550.	260570.	260580.	260590.	260000.	260530.	250530.
*							
-1	.001	.001	.001	.001	.001	.009	.251
0	.001	.000	.000	.001	.000		.055
1	.002	.000	.000	.001	.000		
2	.003	.001	.000	.001	.001		
3	.004	.002	.001	.002	.002		
4	.005	.002	.001	.002	.002		
5	.007	.003	.001	.003	.003		
6	.007	.005	.001	.005	.006		
7	.010	.006	.003	.005	.006		
8	.017	.007	.010	.008	.007		
9	.020	.011	.016	.012	.011		
10	.023	.016	.038	.024	.014		
11	.058	.001	.062	.072	.005		
12	.177	.001	.191	.044	.014		
13	.017	.003	.074	.004	.005		
14	.012	.239	.117	.006	.221		
15	.015	.012	.014	.008	.011		
16	.020	.021	.021	1.127	.022		
17	.029	.033	.032	.018	.031		
18	.042	.050	.047	.025	.047		
19	.062	.075	.070	.036	.071		
20	.090	.111	.102	.053	.106		
21	.132	.163	.150	.078	.156		
22	.194	.240	.221	.115	.230		
23	.285	.353	.325	.169	.338		
24	.417	.517	.476	.247	.497		
25	.613	.760	.699	.364	.729		
26	2.140	2.655	2.441	1.270	2.561		
*							
*							
	260540.	260540.	260560.	260560.	260560.	260560.	260570.
	1.90	1.90	1.90	1.90	1.90	1.90	1.90
	103.	107.	16.	28.	103.	107.	16.
	250540.	240510.	260550.	250550.	250560.	240530.	260560.
*							
-1	.370	.082	.479	.073	.111	.040	.625
0	.550	.059	.052	.011	.095	.028	.422
1	.554	.019			.039	.013	.033
2	.389	.002			.004	.002	
3	.163						
4	.024						
*							
*							
	260570.	260570.	260580.	260580.	260580.	260000.	260000.
	1.90	1.90	1.90	1.90	1.90	1.90	1.90
	103.	107.	16.	103.	107.	16.	22.
	250570.	240540.	260570.	250580.	240550.	260000.	260000.
*							
-1	.056	.023	.787	.016	.021	.445	.002
0	.057	.019	.272	.004	.007	.061	
1	.031	.021				.001	
2	.001	.006					
3		.002					
*							
*							

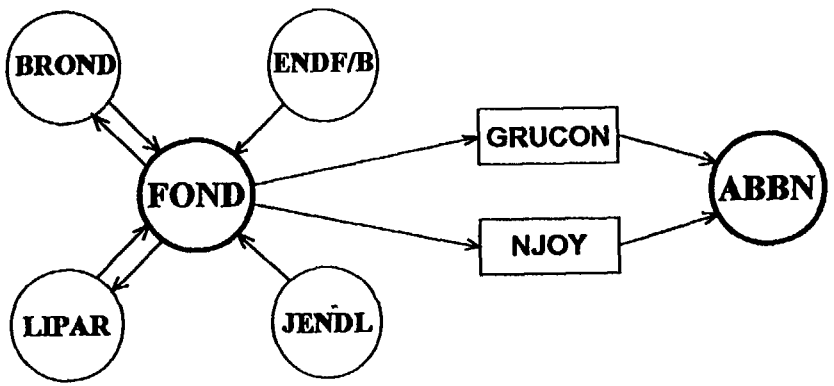
PROCEEDING

FOND≠BROND

- *FOND* contains more materials than *BROND*
- some materials from *BROND* are not included in *FOND*
- *all files* from *FOND* were proceeded by *GRUCON* and some by *NJOY* also

LIPAR- resonance parameter library below 1 Kev

- the basic isotopes were included in *ABBN*

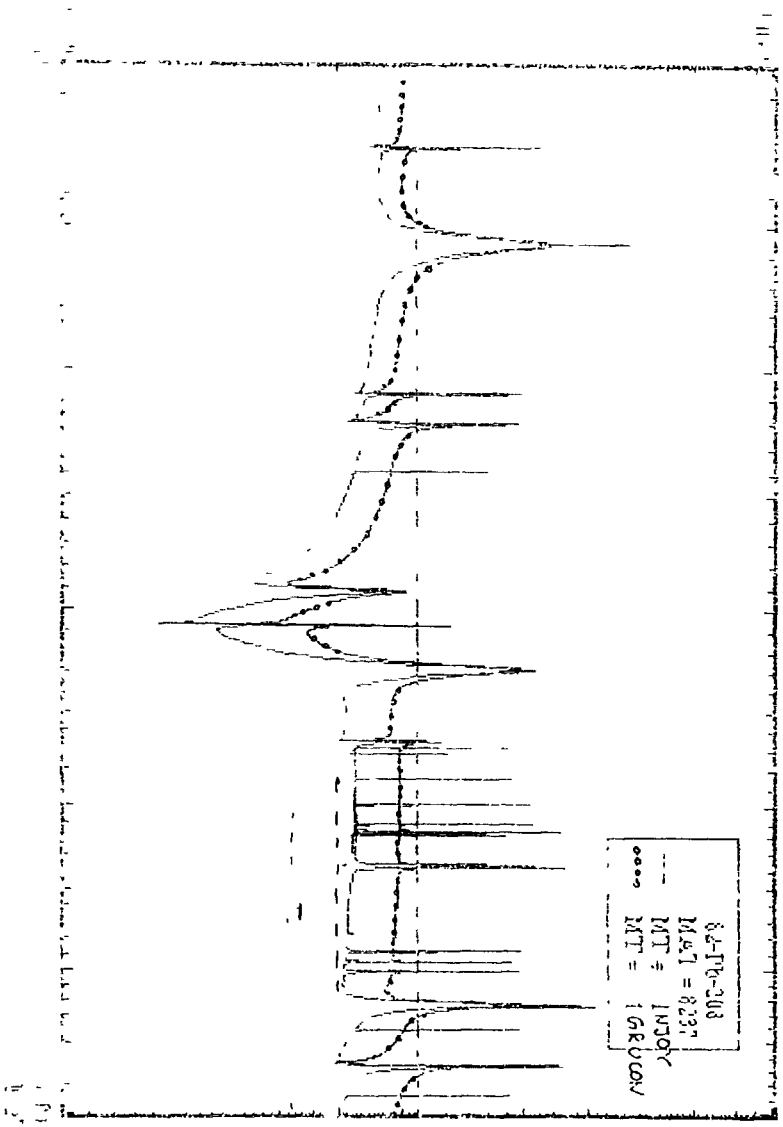


GRUCON

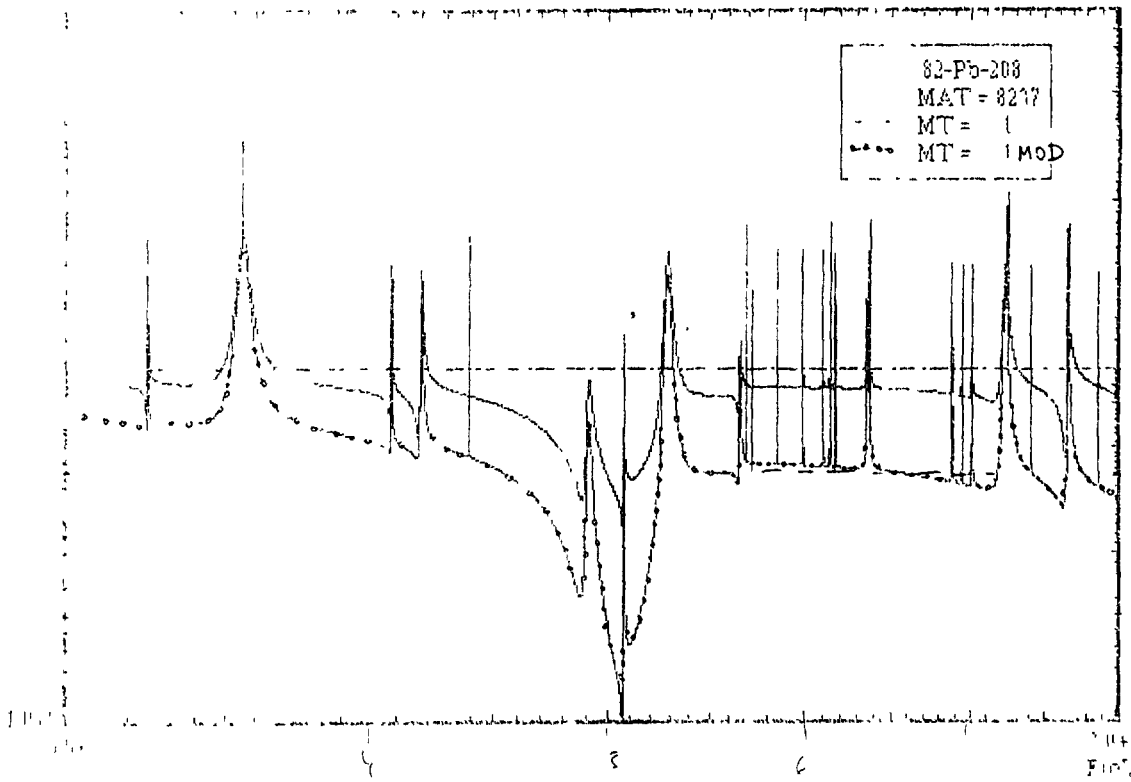
- own development (Sinitsa V.);
- possibility of local energy problem solution;
- various method of subgroup calculation;
- analysis of transmission experiment results;
- file editing.

NJOY

- reliability (many users)
- completeness (not only neutron cross-section)

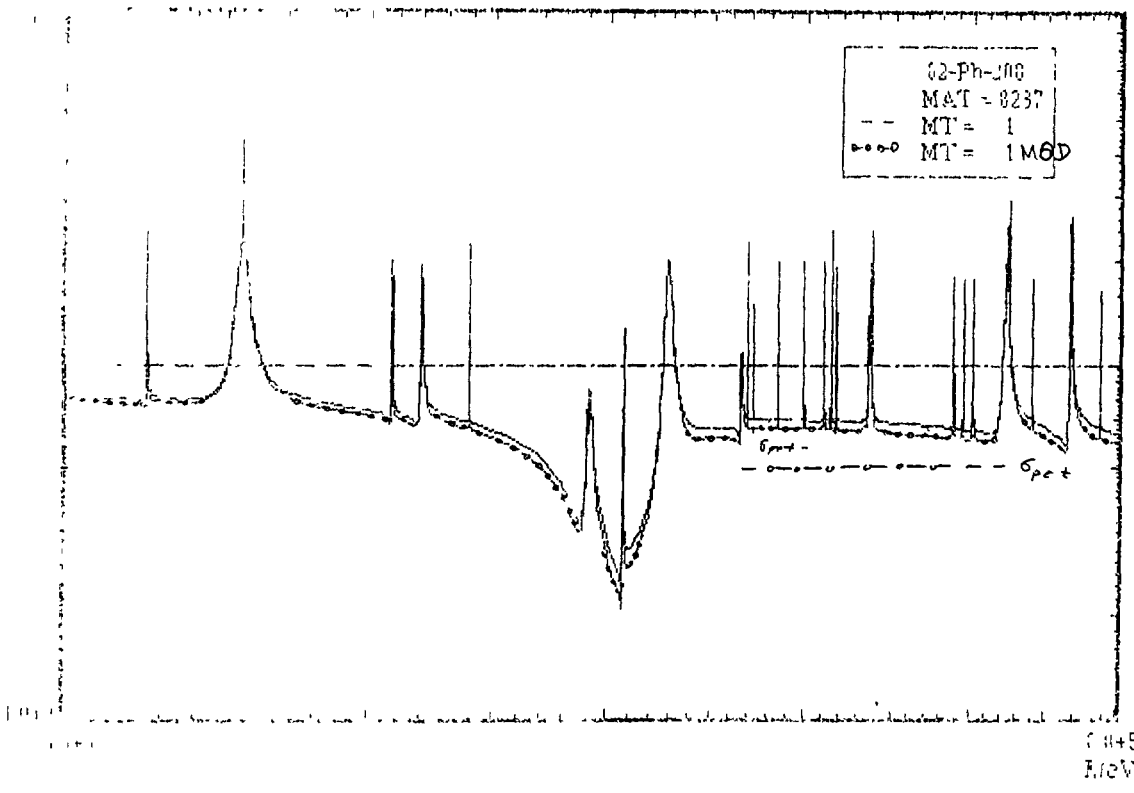


Cross-Sections after GROCON



a - a 123 A^1/3 + 0.08

Cross-Sections after NJOY



a = AP
a - 0.123 A^1/3 + 0.08

SUBGROUP PARAMETER CALCULATION.

a) Moment method:

a_s, $\sigma_{r,s}$ are determinated from equation

$$\sum_{s=1}^{NS} \frac{a_s \sigma_{r,s}}{(\sigma_{t,s}+\sigma_0)^n} = \left\langle \frac{\sigma_r}{(\sigma_t+\sigma_0)^n} \right\rangle$$

b) Method of least squares:

a_s, $\sigma_{r,s}$ are determinated from condition

$$\min \left\{ \sum_k \left[\sum_{s=1}^{NS} \frac{a_s \sigma_{r,s}}{\sigma_t+\sigma_{0,k}} - \left\langle \frac{\sigma_r}{\sigma_t+\sigma_{0,k}} \right\rangle \right] \right\}$$

c) Combinations of previous methods:

In all cases:

1) NS≤5,

2) a_s does not depend from temperature,

3) the balance conditions are strongly fulfilled:

$$\sum a_s = 1, \quad \sum a_s \sigma_{r,s} = \langle \sigma_r \rangle .$$

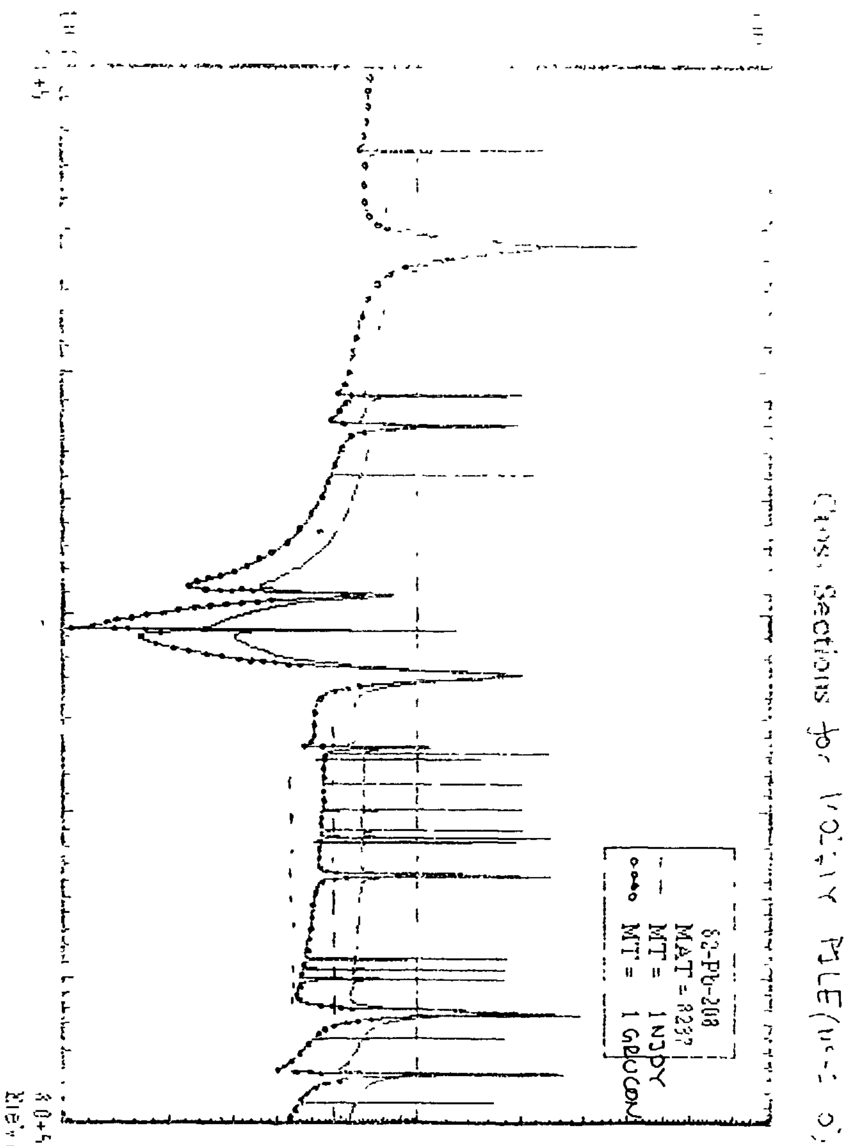
SUBGROUP PARAMETERS IN ABBN:

AT 300K - for all materials.

AT 300,900,2100K - for U-238 and Pu-239 only.

SUBGROUP PARAMETERS CALCULATION PROCESS IN GRUCON IS NOT FULLY AUTOMATIZED TILL NOW!

SOLVING THIS PROBLEM IN THE FRAME ON NJOY APPEARS AS VERY USEFUL.



UNRESOLVED RESONANCE REGION

IMPORTANT REGION FOR FAST REACTORS AND ELECTRO-NUCLEAR ACTINIDE TRANSMUTATORS ESPECIALLY FOR TH-232, U-233,235,238, PU-239,240,241, NP-237, AM-241,243

THE NEXT IS DONE

1. NJOY and GRUCON results are compared using the same input data files.
2. Indefiniteness of f and f(T) caused by different evaluation are estimated and the more certain value are adopted.
3. Energy dependent subgroup parameters with temperature independent subgroup shares are constructed for U-238 and Pu-239.
4. Data for Th-237,U-235,U-238 and Pu-239 are verified in the deep transmission experiments on the IBR reactor in Dubna.
5. Recent calculations based on ENDF/B-6 and JENDL-3 agreed with ABBN subgroup data
6. Unresolved resonance structure subgroup paramiterisation used in detail energy CM Monte-Carlo calculations.

IT WOULD BE WANTED

1. To introduce the subgroups in the ENDF URR format.
2. To verify the Np-237 and U-233 data by the deep transmission experiments
3. Data revision for U-238,U-235 and Pu-239 only if some contradictions with new experimental data would be found

UNRESOLVED RESONANCE REGION

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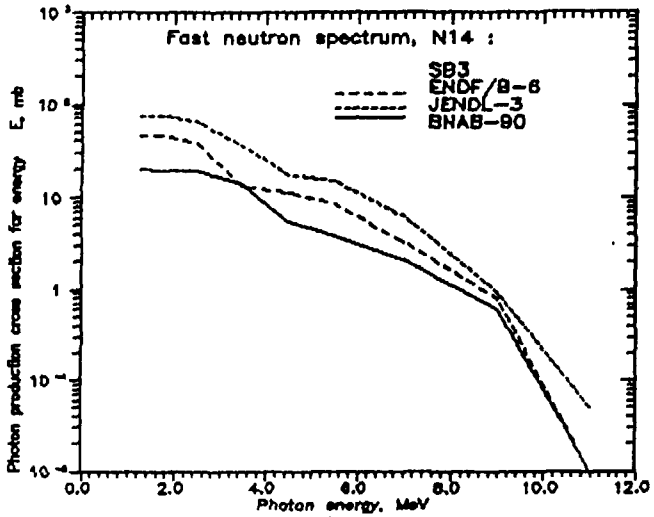
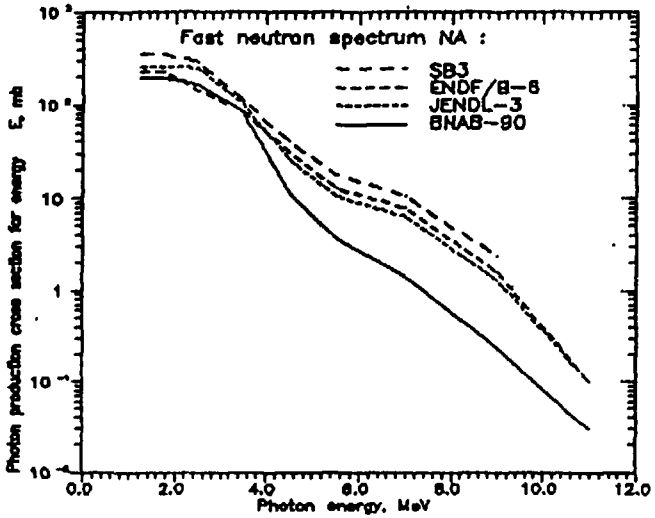
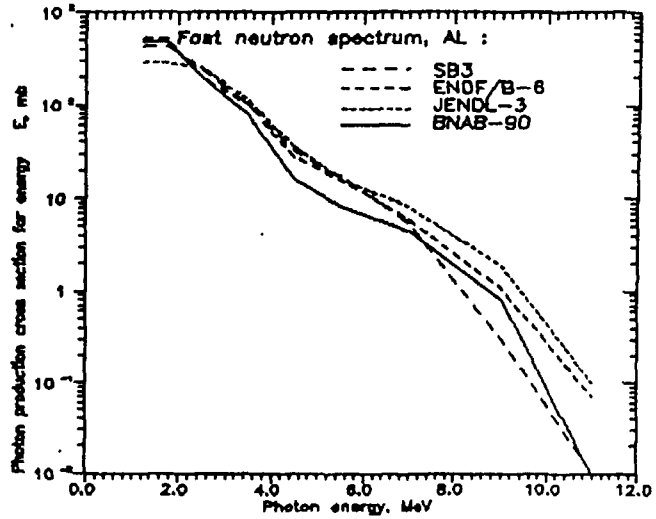
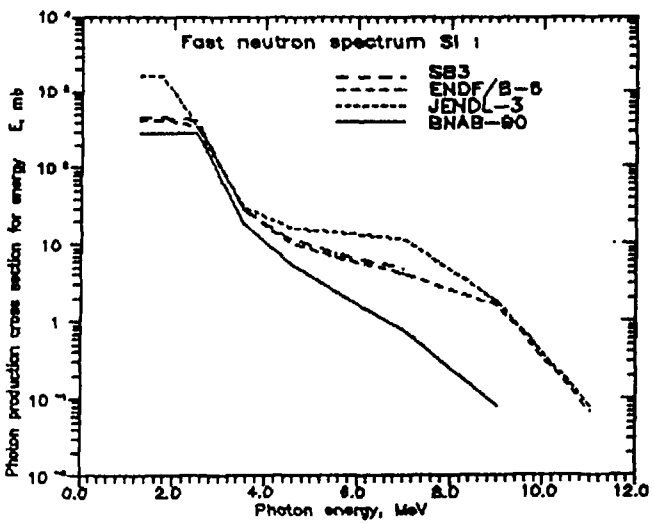
IT WOULD BE WANTED

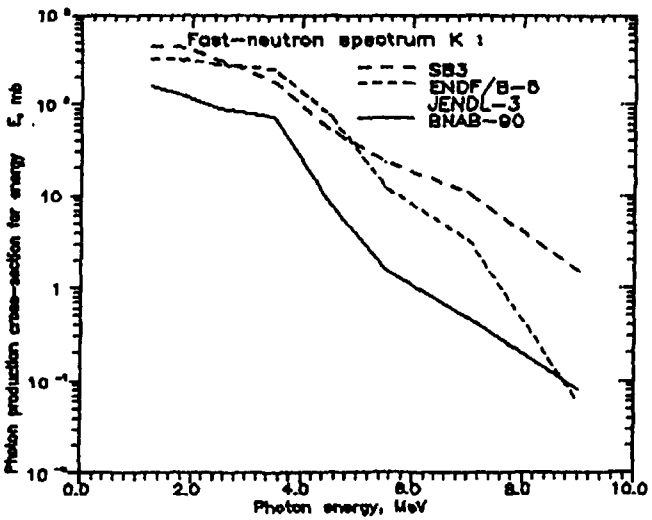
1. To introduce the subgroups in the ENDF URR format.
2. To verify the Np-237 and U-233 data by the deep transmission experiments
3. Data revision for U-238,U-235 and Pu-239 only if some contradictions with new experimental data would be found.

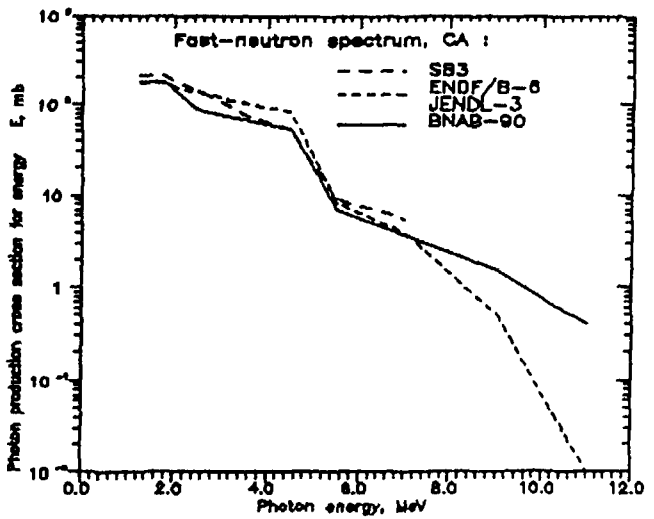

Fast neutron spectrum, N14 :
SB3
ENDF/B-6
JENDL-3
BNAB-90
Photon production cross section for energy E, mb
Photon energy, MeV


Fast neutron spectrum NA :
SB3
ENDF/B-6
JENDL-3
BNAB-90
Photon production cross section for energy E, mb
Photon energy, MeV

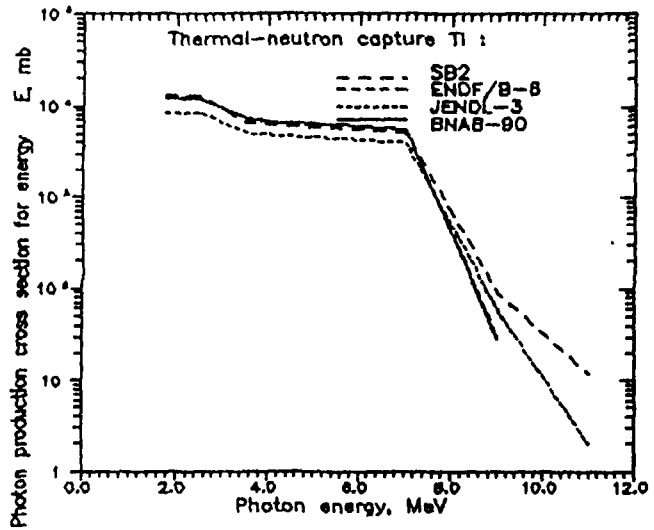

Fast neutron spectrum, AL :
SB3
ENDF/B-6
JENDL-3
BNAB-90
Photon production cross section for energy E, mb
Photon energy, MeV

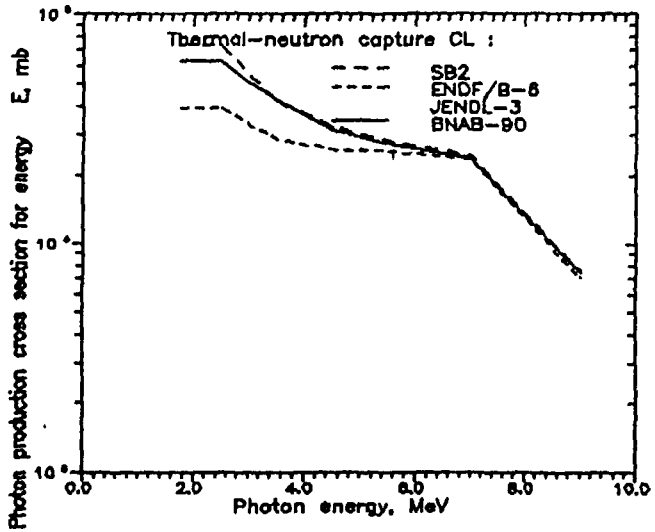

Fast neutron spectrum SI :
SB3
ENDF/B-6
JENDL-3
BNAB-90
Photon production cross section for energy E, mb
Photon energy, MeV

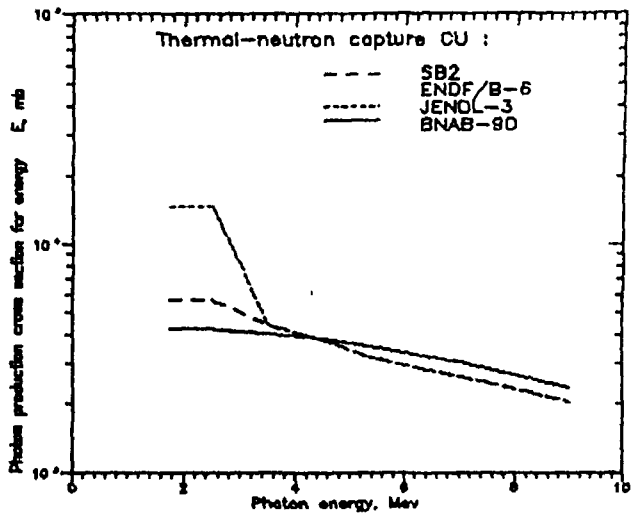

Fast-neutron spectrum K :
SB3
ENDF/B-6
JENDL-3
BNAB-90
Photon production cross-section for energy E, mb
Photon energy, MeV

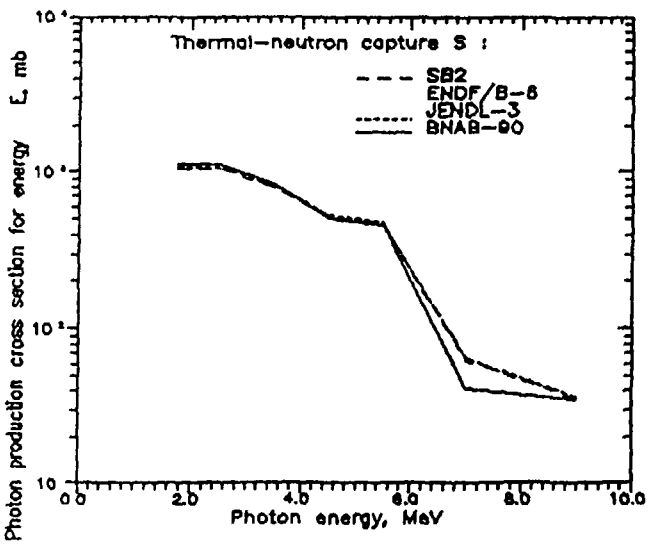

Fast-neutron spectrum, CA :
SB3
ENDF/B-6
JENDL-3
BNAB-90
Photon production cross section for energy E, mb
Photon energy, MeV

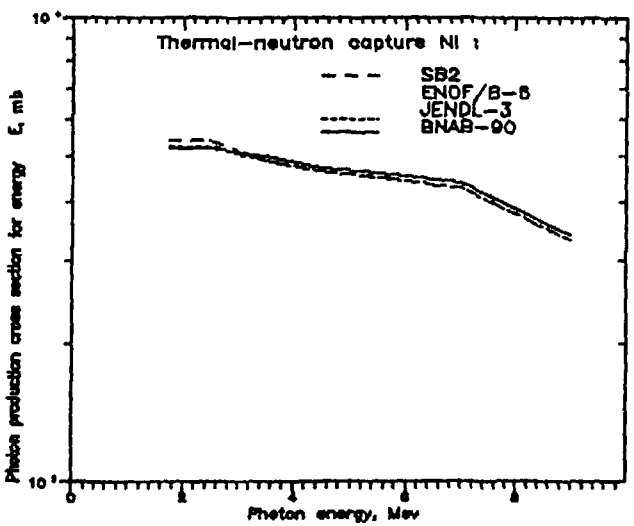
3.4.3

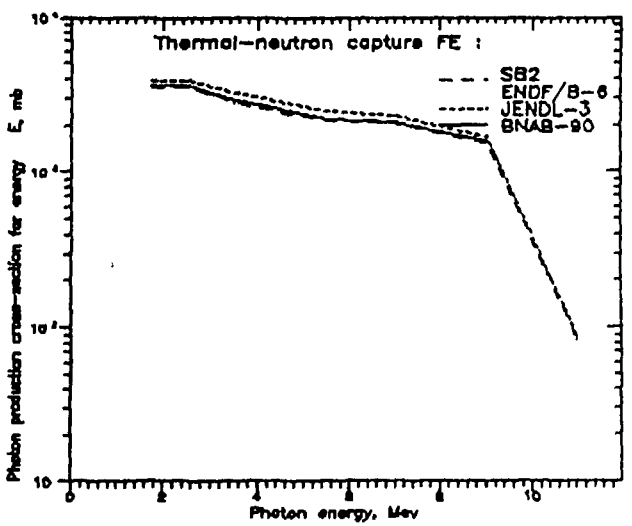

Thermal-neutron capture Ti :
SB2
ENDF/B-6
JENDL-3
BNAB-90
Photon production cross section for energy E, mb
Photon energy, MeV

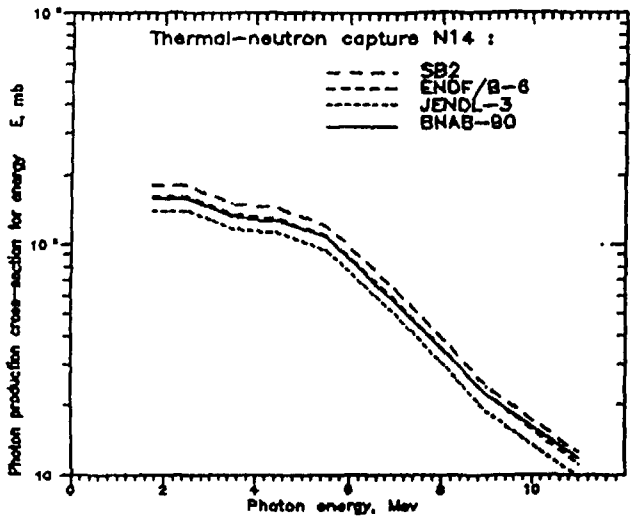

Thermal-neutron capture CL :
SB2
ENDF/B-6
JENDL-3
BNAB-90
Photon production cross section for energy E, mb
Photon energy, MeV

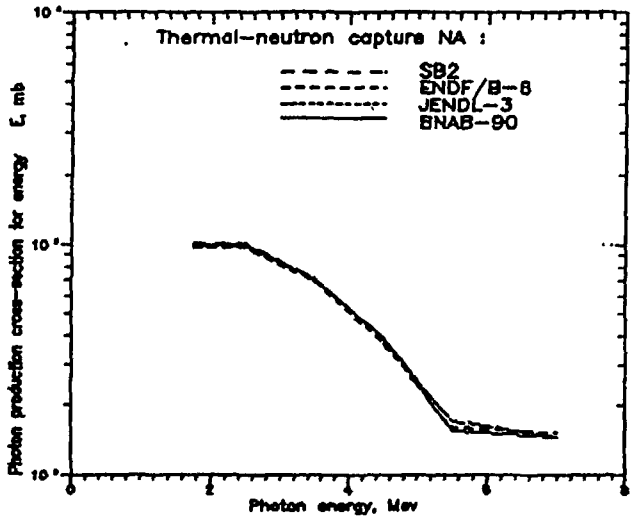

Thermal-neutron capture CU :
SB2
ENDF/B-6
JENDL-3
BNAB-90
Photon production cross section for energy E, mb
Photon energy, Mev

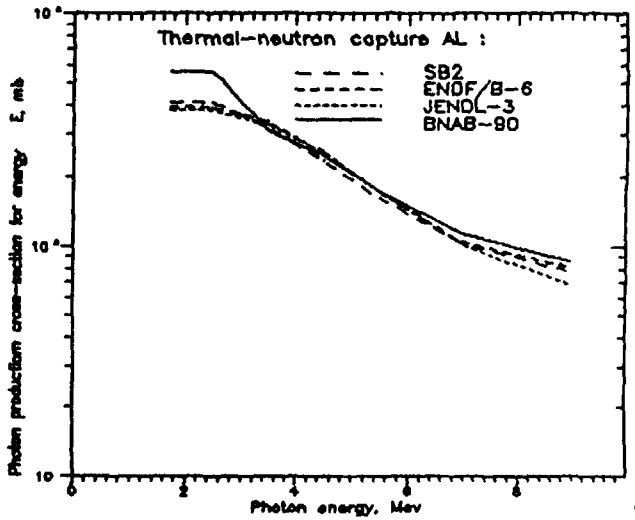

Thermal-neutron capture S :
SB2
ENDF/B-6
JENDL-3
BNAB-90
Photon production cross section for energy E, mb
Photon energy, MeV

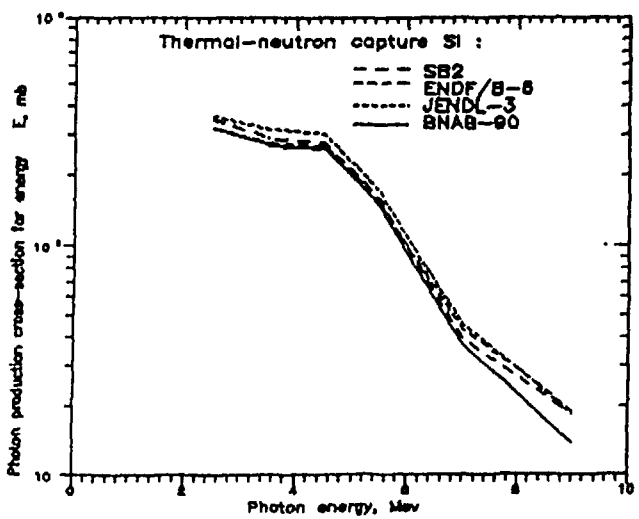

Thermal-neutron capture NI :
SB2
ENDF/B-6
JENDL-3
BNAB-90
Photon production cross section for energy E, mb
Photon energy, Mev

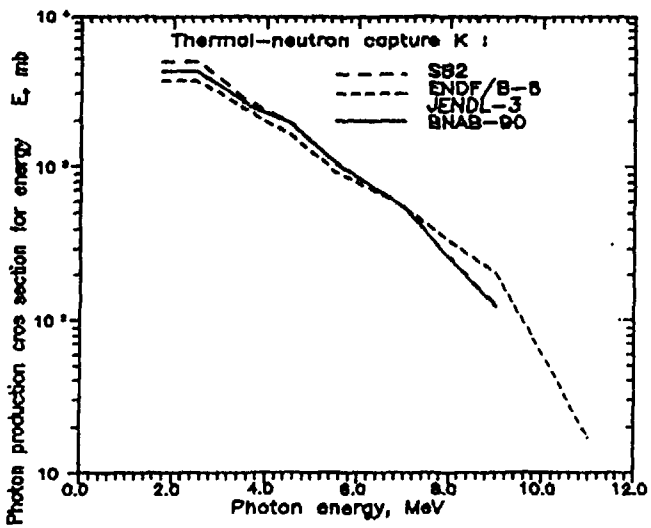

Thermal-neutron capture FE :
SB2
ENDF/B-6
JENDL-3
BNAB-90
Photon production cross-section for energy E, mb
Photon energy, Mev

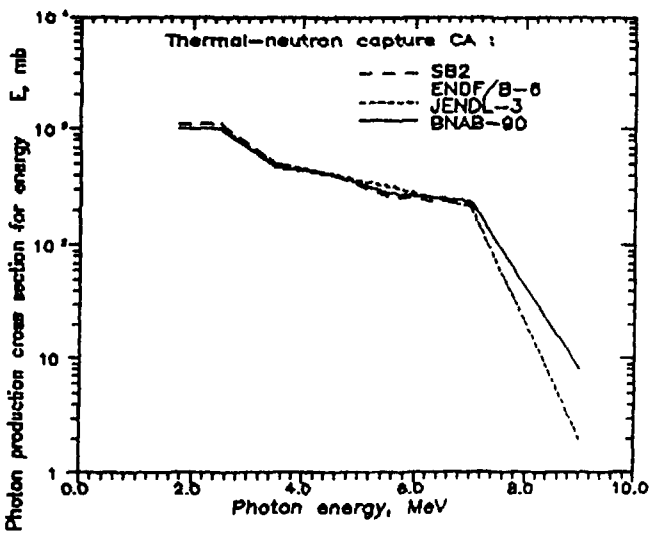

Thermal–neutron capture N14 :
SB2
ENDF/B–6
JENDL–3
BNAB–90
Photon production cross–section for energy E, mb
Photon energy, Mev


Thermal–neutron capture NA :
SB2
ENDF/B–6
JENDL–3
BNAB–90
Photon production cross–section for energy E, mb
Photon energy, Mev

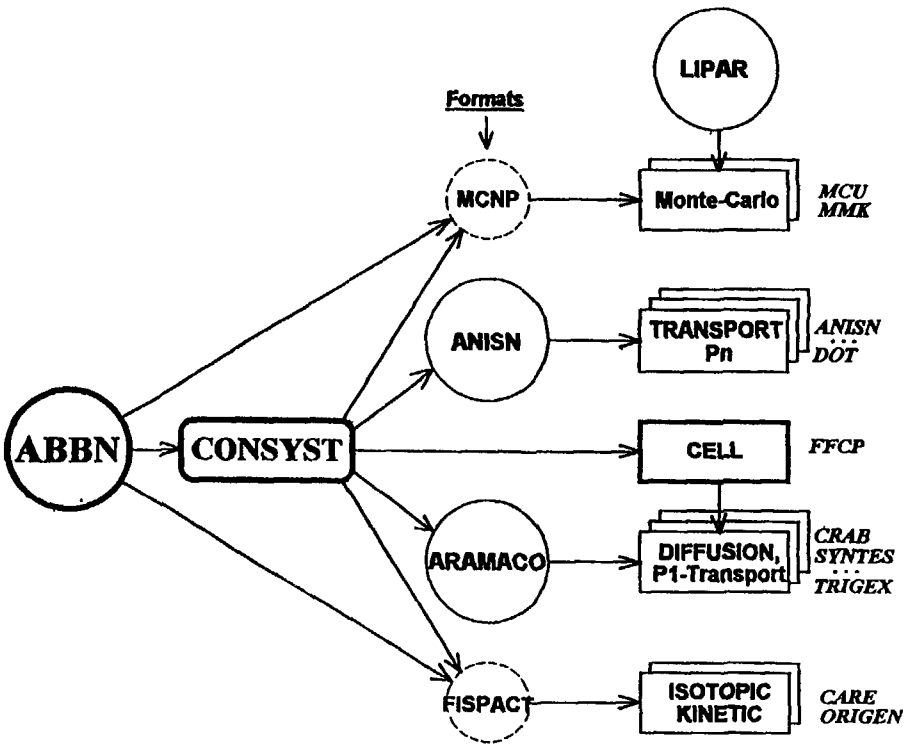

Thermal–neutron capture AL :
SB2
ENDF/B–6
JENDL–3
BNAB–90
Photon production cross–section for energy E, mb
Photon energy, Mev


Thermal–neutron capture SI :
SB2
ENDF/B–6
JENDL–3
BNAB–90
Photon production cross–section for energy E, mb
Photon energy, Mev


Thermal–neutron capture K :
SB2
ENDF/B–6
JENDL–3
BNAB–90
Photon production cros section for energy E, mb
Photon energy, MeV


Thermal–neutron capture CA :
SB2
ENDF/B–6
JENDL–3
BNAB–90
Photon production cross section for energy E, mb
Photon energy, MeV

CONstant SYSTem
for group constant preparation.



ARAMACO - NGN=26 group
ANISN - NGN=1 - 300 , NGG=15 .group

TESTING

Integral experiments (well known spectrum):
- -average cross-section;
- -resonance integrals;
- -transmissions;
- -cross-section of removing under fission threshold.

Direct file testing.

Macroscopic experiments:
- compact critical assembles (GODIVA type);
- solution experiments (U,Pu - fuel);
- fast critical assembles;
- regular lattices.

We test: files, reprocessing codes, calculational codes.

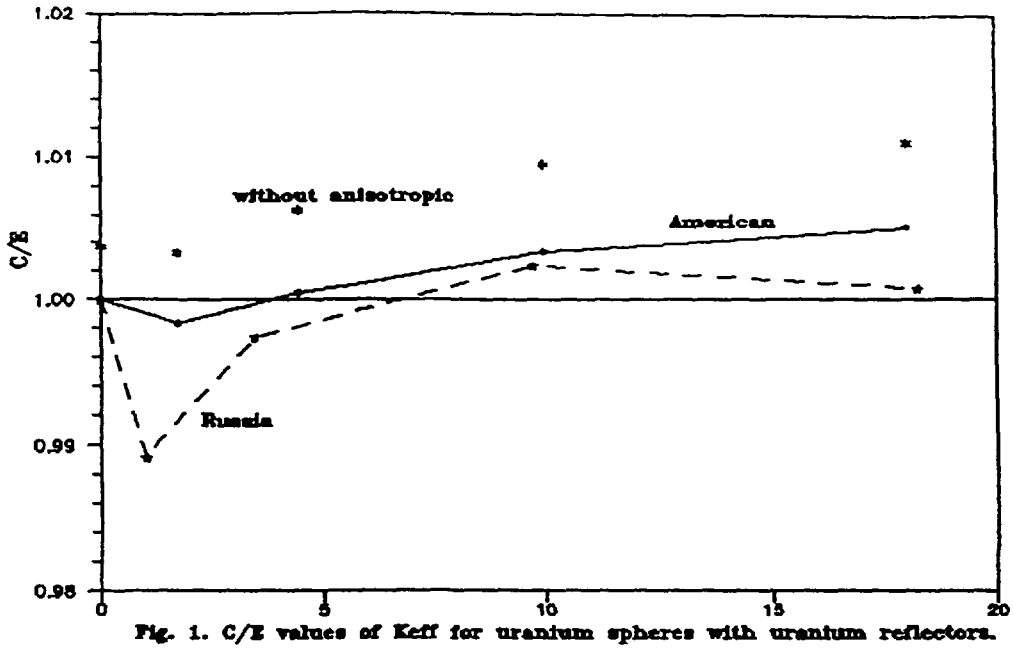


Fig. 1. C/E values of Keff for uranium spheres with uranium reflectors.

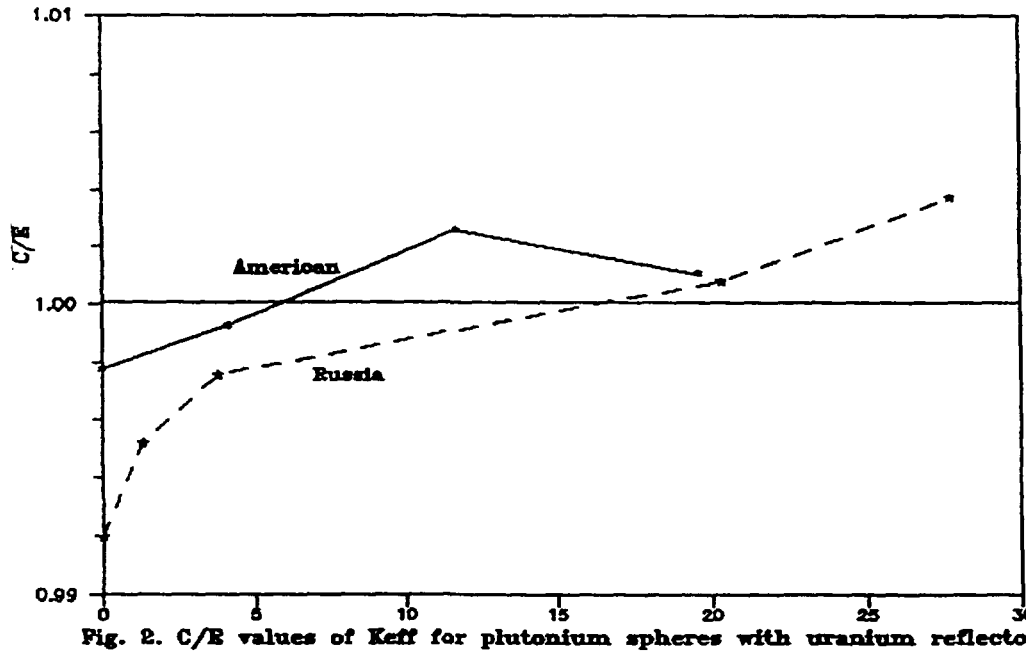


Fig. 2. C/E values of Keff for plutonium spheres with uranium reflectors.

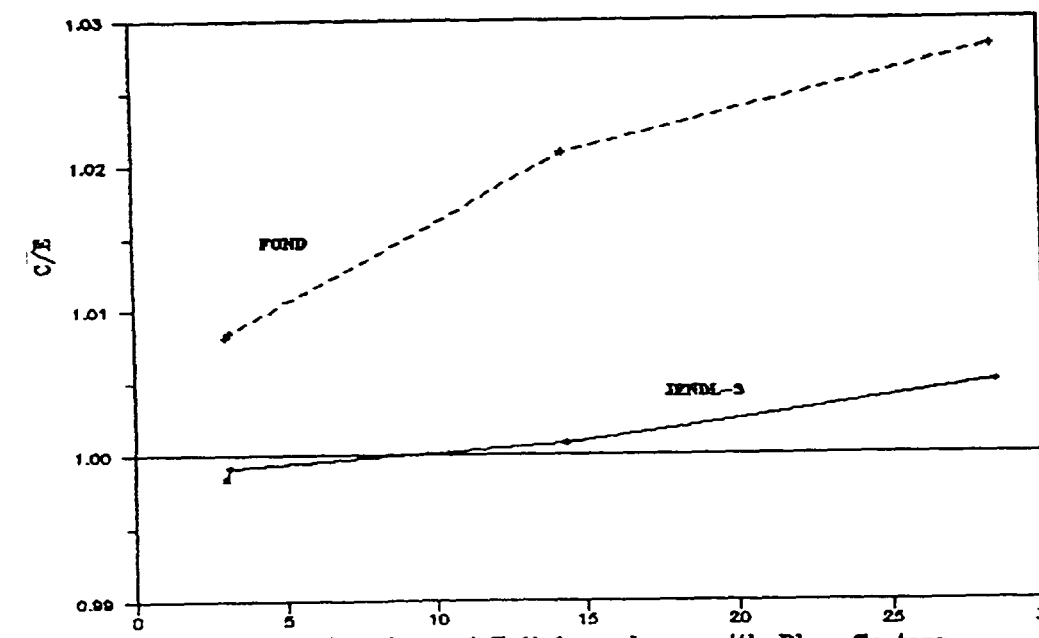


Fig. 3. C/E values of Keff for spheres with Pb reflectors.

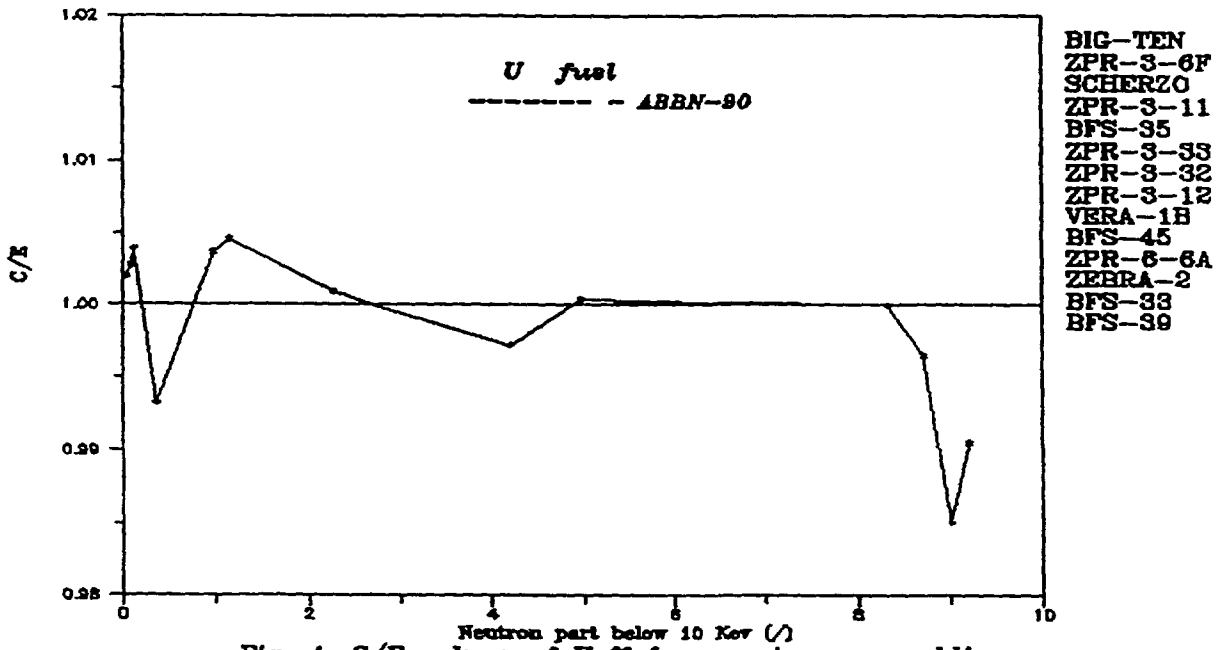


Fig. 4. C/E values of Keff for uranium assemblies.

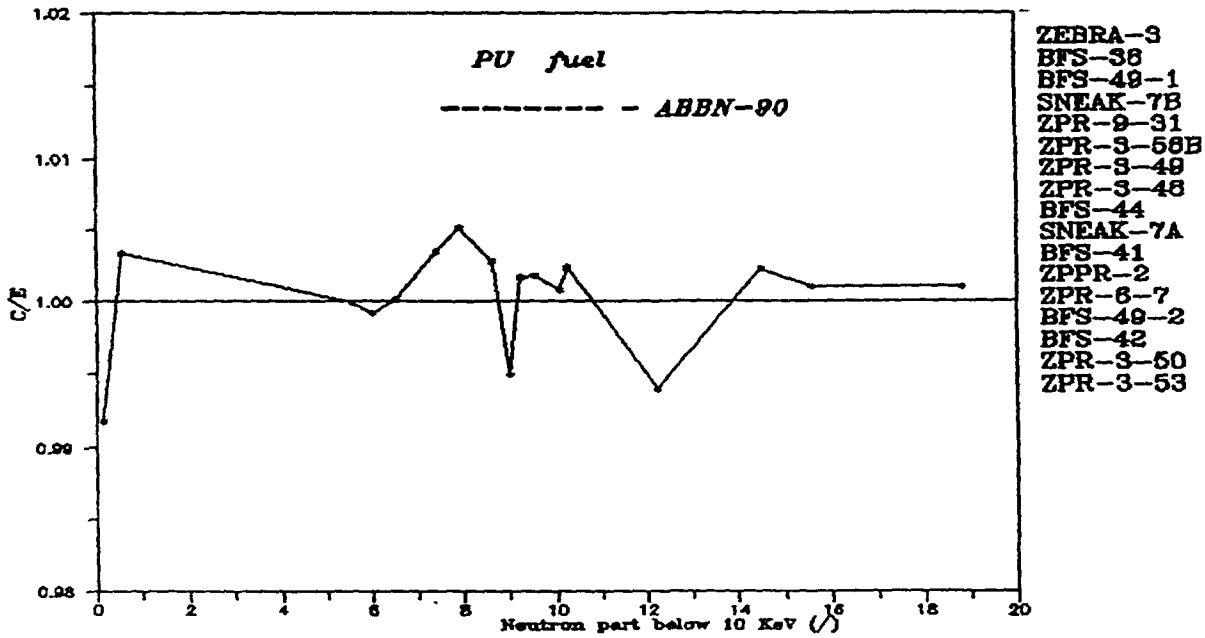
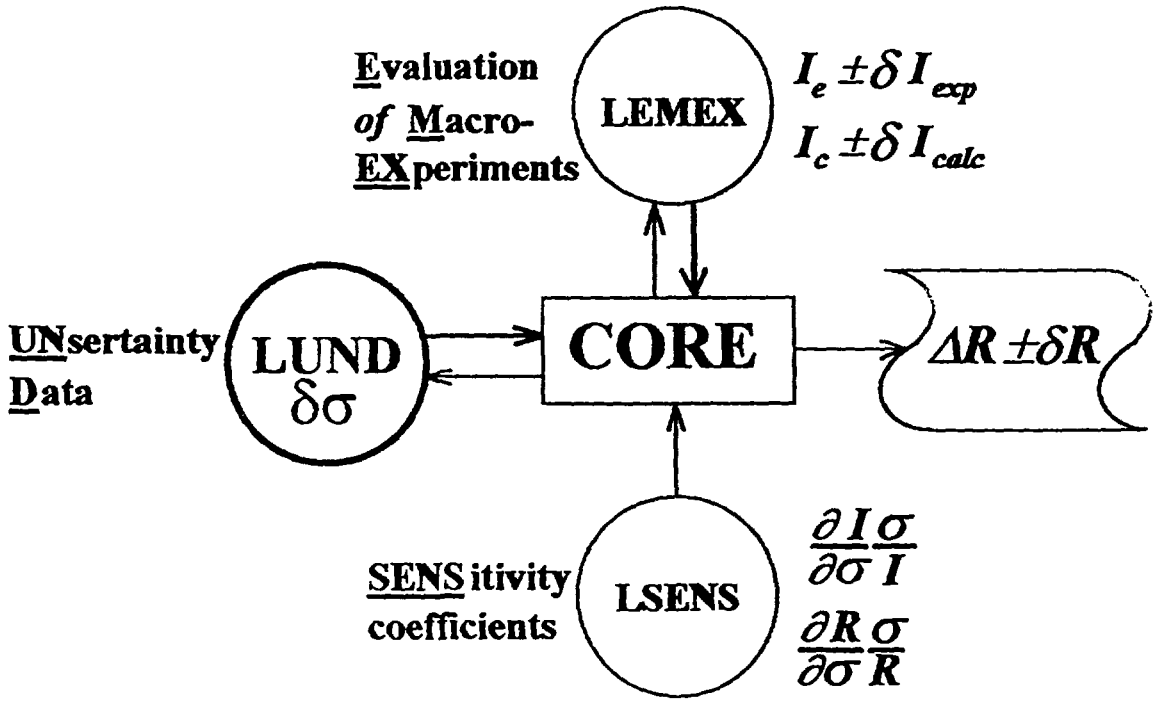


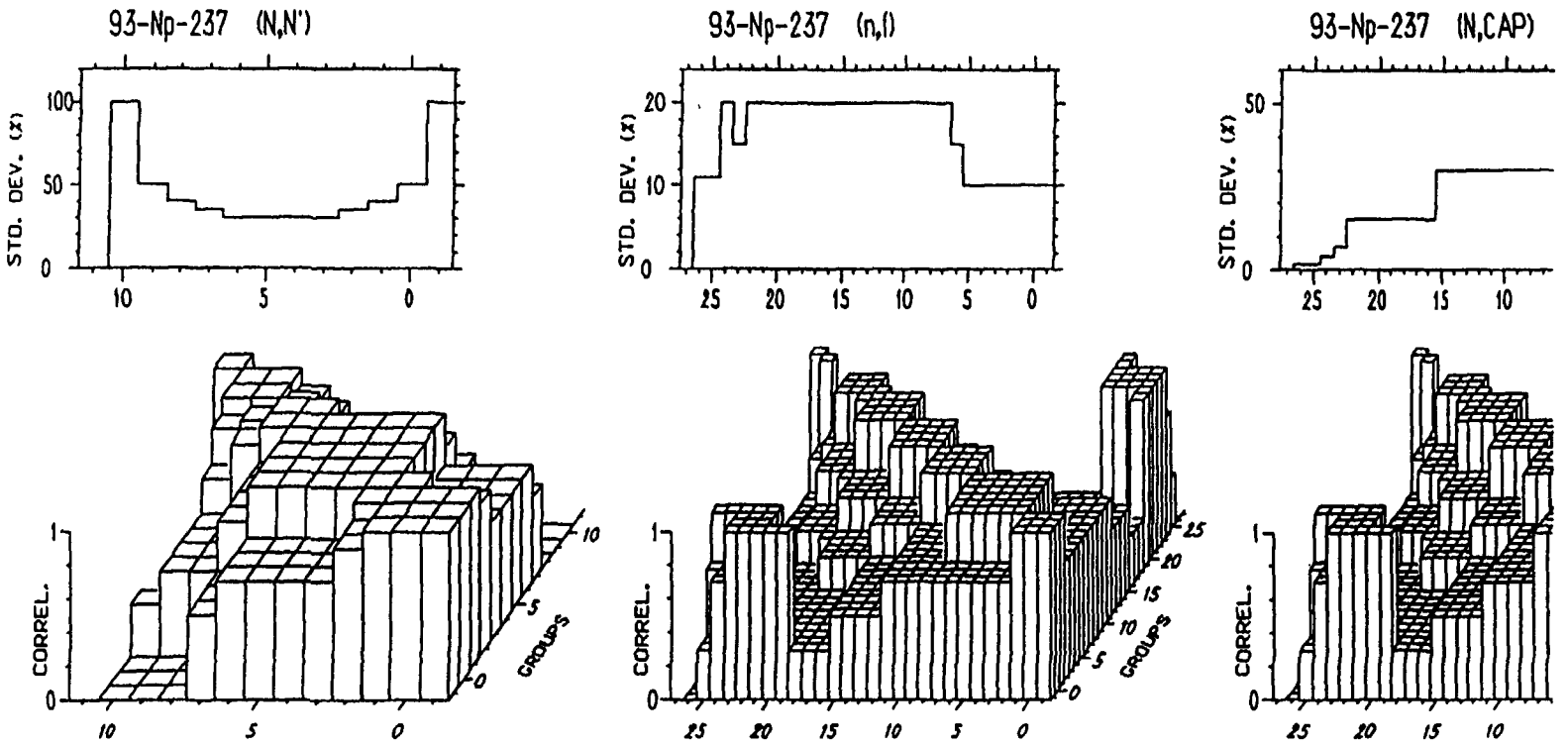
Fig. 5. C/E values of Keff for plutonium assemblies.

COVARIANCE MATRIX

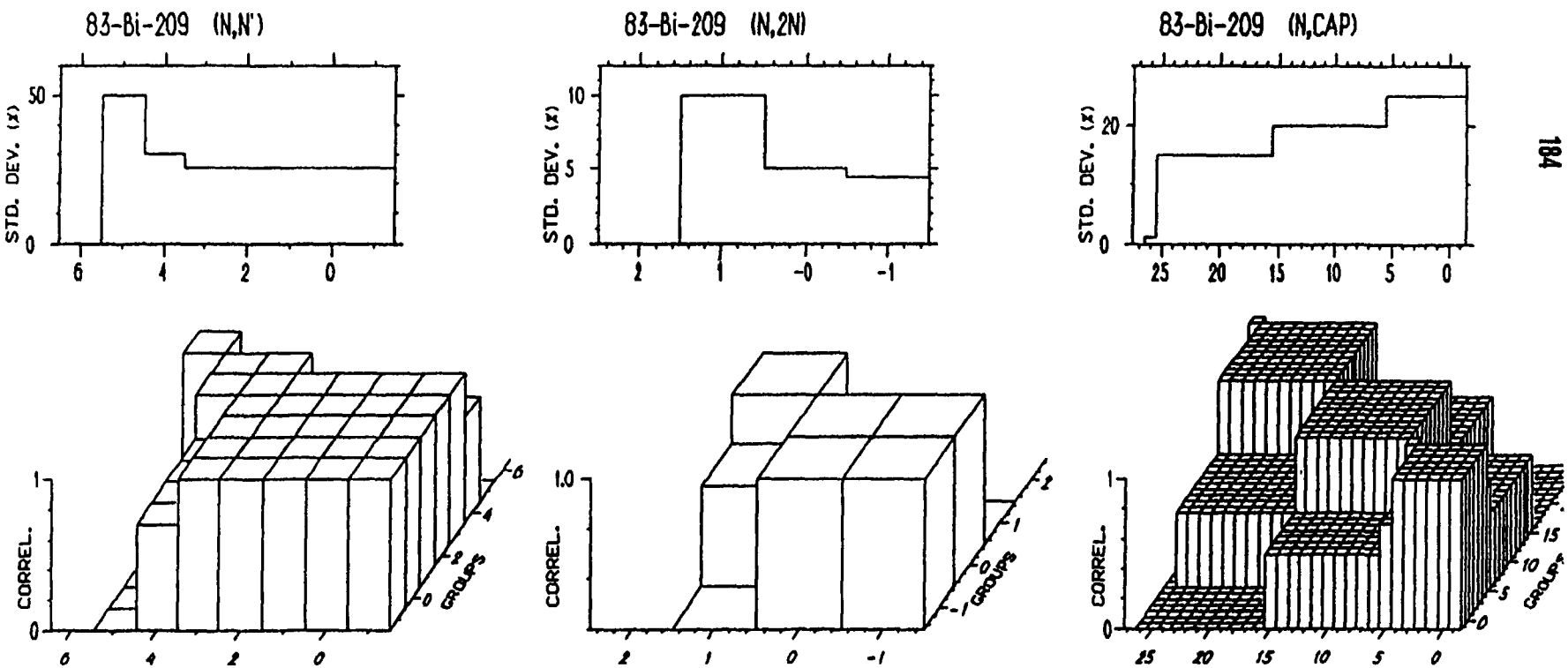
WAS - ABBN-78 12 groups, 60 reactions
NOW - ABBN-90 28 groups 500 reactions

- is obtained not from ENDF/B file but from consideration of experimental conditions and theoretical parameterization;
- is widely used in uncertainty analysis of reactor physics calculations;
- 28 groups are quite sufficient;
- all ENDF/B-6 MF=3 data are processed by NJOY but are not analyzed;
- results obtained on the basis of covariance matrix should be interpreted with the care taking into account the common sense.





КОВАРИАЦИОННЫЕ МАТРИЦЫ ДЛЯ НЕПТУНИЯ-237


83-Bi-209 (N,N')
STD. DEV. (%)
50
0
6
4
2
0
CORREL.
1
0
GROUPS
83-Bi-209 (N,2N)
STD. DEV. (%)
10
5
0
2
1
-0
-1
CORREL.
1.0
GROUPS
83-Bi-209 (N,CAP)
STD. DEV. (%)
20
0
25
20
15
10
5
0
CORREL.
1
0
GROUPS

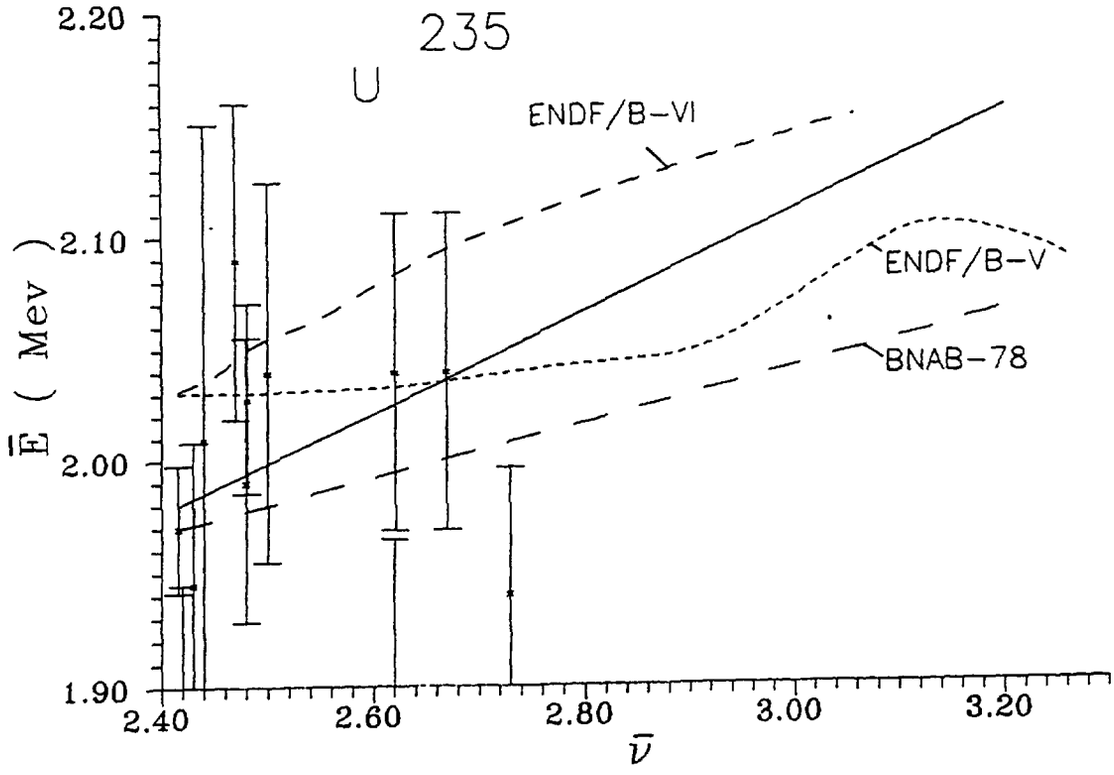


Рис. 6

FISSION SPECTRUM

1. It was considered that average energy (E) of U-235 fission spectrum induced by thermal neutrons is equal 1.97 Mev with the accuracy which is better than 1%. Such accuracy meets requirements of fast and thermal reactors.
2. ENDF/B-5 increases E up to 2.03 Mev (3%). Macroscopic experiments ask the same value of E.
3. The results of automatic adjustment will mainly depend on evaluated accuracy of average energy (E) of fission spectrum. What is a real accuracy of E for the basic nuclei (U-235, Pu-239) ?

HYDROGEN TOTAL CROSS-SECTION

1. Evaluated accuracy of hydrogen scattering cross-section is equal 0.2% .
2. The difference of current ENDF/B-6 data and ABBN-64 data is about 1.5% (in average).
3. Calculation of homogeneous critical assembles with U-235 (90%)and water shows :
   - MCNP and MCU give the same results;
   - calculation with H(ENDF/B-6) overestimates the Keff value critical assemblies;
   - calculation with ABBN-64 data gives good results.
4. Can we believe to 0.2% accuracy of hydrogen scattering cross-section ?

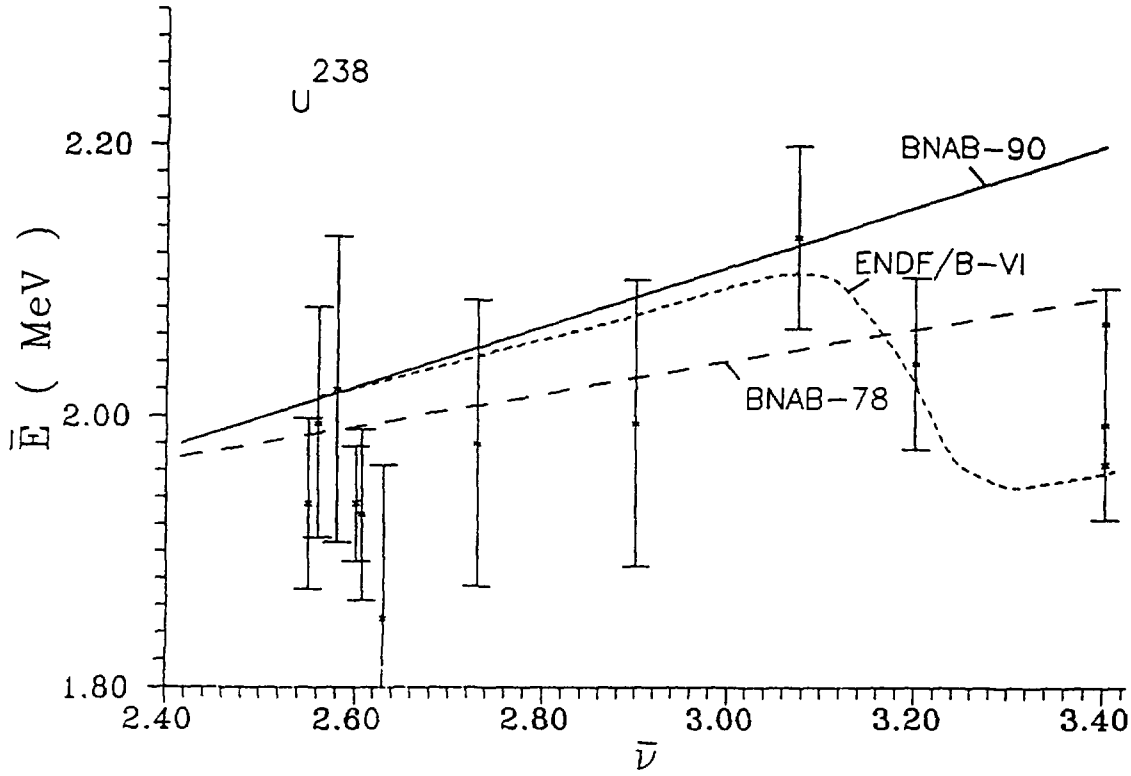

U 238
BNAB-90
ENDF/B-VI
BNAB-78
Ē (MeV)
2.20
2.00
1.80
2.40
2.60
2.80
3.00
3.20
3.40
ν̄

Рис. 7

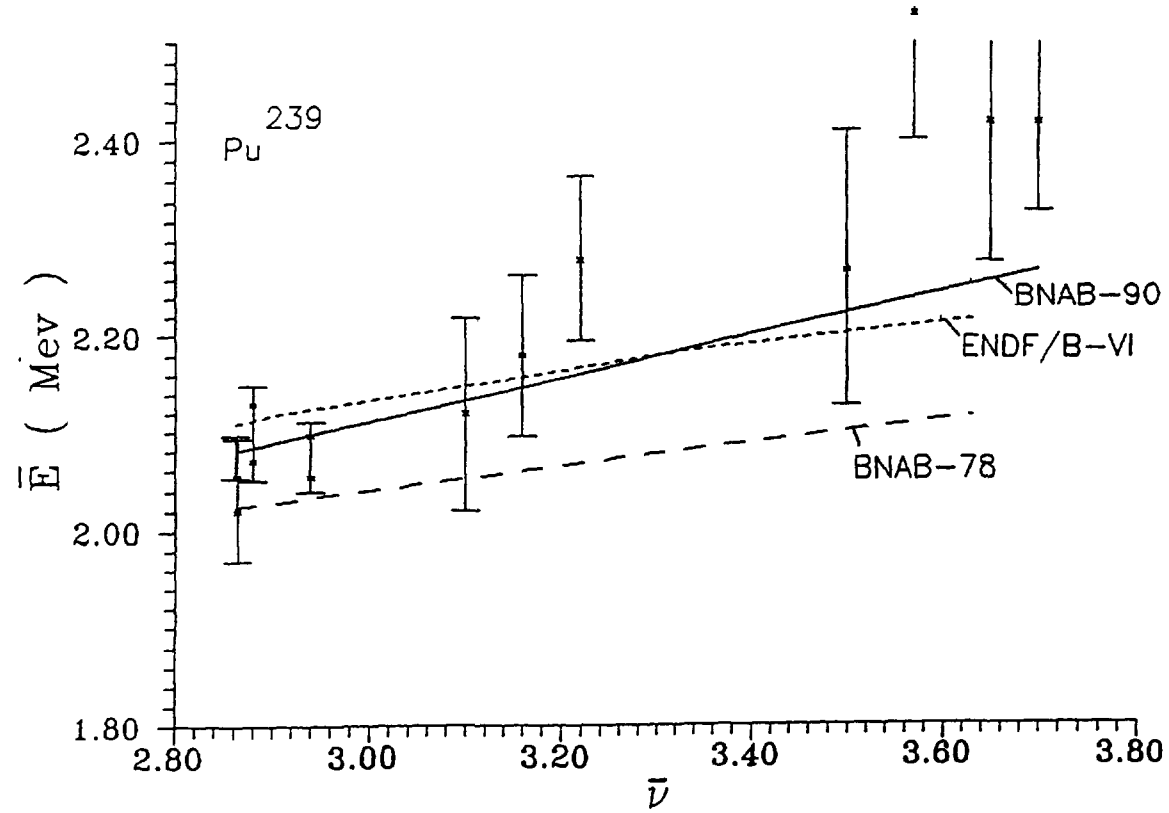

Pu 239
BNAB-90
ENDF/B-VI
BNAB-78
Ē (Mev)
2.40
2.20
2.00
1.80
2.80
3.00
3.20
3.40
3.60
3.80
ν̄

Рис. 8

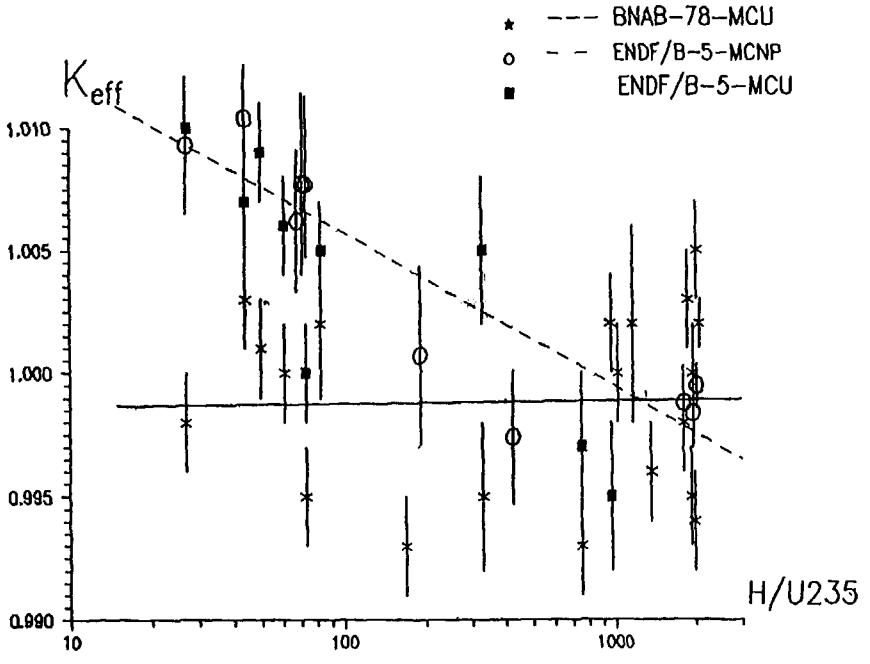
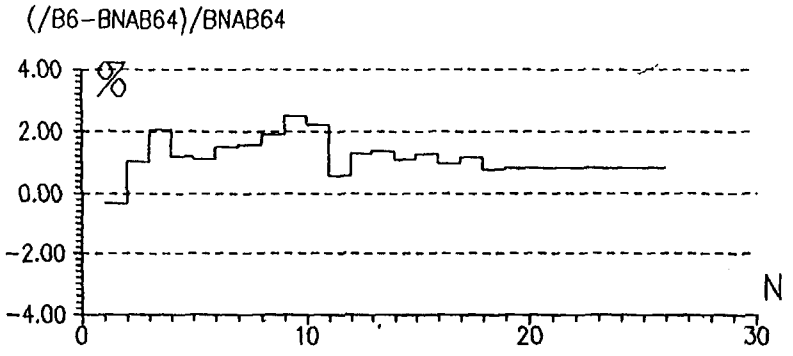

BNAB-78-MCU
ENDF/B-5-MCNP
ENDF/B-5-MCU
K_{eff}
1.010
1.005
1.000
0.995
0.990
10
100
1000
H/U235

Рис 1


(/B6-BNAB64)/BNAB64
%
4.00
2.00
0.00
-2.00
-4.00
0
10
20
30
N

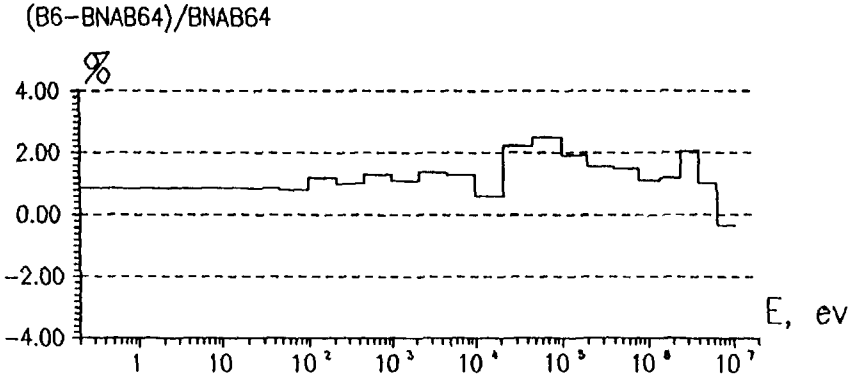

(B6-BNAB64)/BNAB64
%
4.00
2.00
0.00
-2.00
-4.00
1
10
10^2
10^3
10^4
10^5
10^6
10^7
E, ev

Рис 2

PROPOSALS

1. To do additional NJOY testing concerning Reich-Moore, General R-Matrix and Hybrid R-Function formalism treatment especially in the regions where p-wave scattering is important.
2. To include the subgroup calculation module into NJOY.
3. To include subgroup approximation of cross-section resonance structure in ENDF format for URR (with LSSF=1) (for direct using in Monte-Carlo codes).
4. To create the international library of evaluated integral in macroscopic experiments (under the NDS IAEA aegis) for validation of neutron constants used in technology.
5. To publish ABBN-90 group constant set and supplement documentation by IAEA.

UPDATING OF THE WIMS GROUP CONSTANTS.

What is done

1. Subroutine for calculation of detailed neutron spectrum in homogeneous resonance media is included in GRUCON.
2. Code for the WIMS library edition is written:
   - inclusion or deletion of material from the library;
   - infinite dilution cross-section may be changed;
   - inclusion or changing of resonance tables;
   - graphical comparison of cross-sections.
3. Data for new materials are included into the WIMS library: W, Mo, Sn, Hf, Ta, Am-241, Np-237.
4. Data for fission spectrum and for next materials are changed: U-235, U238, Pu-239, Pu-240, Zr .
5. H20 thermalization matrices calculated by NJOY and TERMAC were compared and good agreement was found.
6. Reaction cross-sections for fission products and actinides for WIMS energy grid are completed in the ABBN format. Collapsed one-group cross-sections may be calculated using WIMS neutron spectra for ORIGEN and other isotopic codes.

What has to be done

1. Resonance self-shielding has to be taken in to account below 4 eV (for Pu-242, Hf).

Experiences with NJOY and ENDF Pre-Processing System on PC-486 and Benchmarks of WIMS Library Generated from ENDF/B-VI

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Yonin-Kun 449-701, Korea

and

Jung-Do Kim
Korea Atomic Energy Research Institute
Daejon 305-606, Korea

ABSTRACT

The objective of the present study is to develop WIMS-CITATION computation system using PC-486 for nuclear analysis of advanced Pressrurized Water Reactors, whose burnups are desinged to achieve more than 60,000MWD/MTU, and CANDU reactors. In order to achieve the objective, it is essential to generate the proper WIMS library. The possibility of employing NJOY code and ENDF Pre-Processing System are under investigation.

In order to verify the acceptability of the WIMS library generated from ENDF/B-VI on PC-486, new libraries are generated and compared with the libraries generated using CYBER and SUN Work Station for testing benchmark results of five light water and five heavy water lattices. The result shows that PC-486-based WIMS library is acceptable even though there exist some minor problems.

1. Introduction

The capability of personal computers has been improved remarkably, especially in hardwares, in the last few years so that almost all the codes for main frame computers can be adapted on PC-486 with proper FORTRAN compiler. Therefore, it becomes possible to use NJOY91.38 code, which is originally designed to be used on main frame computers and recently on work station computers, on PC-486. Personal computers are widely used by students in universties. It is the reason why PC-486-based nuclear design system for nuclear reactors draws attention.

WIMS code[1] developed by Askew et. al. and distributed by NEA Data Bank has been widely used for nuclear analyses of various thermal reactors. However, almost all of the libraries supplied with different versions of

WIMS code are based on the evaluated data in the sixties. Therefore, the libraries are out of date and have limited applicabilities. The Nuclear Data Section of IAEA initiated the WIMS Library Update Project[2,3] through international cooperations starting from 1991. However, this project may aim, in mind, at producing the library that can be used for analyzing present thermal power reactors which typical burnups are usually up to around 35,000MWD/MTU.

In recent years, the most updated evaluated nuclear data file ENDF/B-VI is released so that a series of activities to generate 69-group constant library for WIMS code that can be used for the extended burnup more than 60,000MWD/MTU, which is the target for the advanced PWR as an utility requirement specified by the Elecric Power Research Institue in the U. S. A., is initiated at Kyung Hee University utilizing NJOY code and ENDF/B-VI file. In order to justify PC-486 Version NJOY a number of light water lattices and heavy water lattices for benchmark experiments recommended by the Cross Section Evaluation Working Group(CSEWG)[4] except heavy water moderated lattices are analyzed using WIMS library generated from ENDF/B-VI. Heavy water option is chosen because the Korea nuclear power program includes a number of CANDU power reactors.

2. Benchmark Experiments

The TRX benchmarks are light water moderated lattices of 1.3w/o-enriched aluminum clad uranium rods arranged in a triangular pattern.[4] The BAPL benchmarks consist of light water moderated critical lattices of 1.311w/o enriched aluminum clad uranium oxide rods in a triangular arrangement.[4] The ZEEP experiments[5] are heavy water moderated and fully reflected lattices operated at room temperature. The fuel rods are of natural uranium metal clad in aluminum. The SRL experiments[6] are single-rod natural uranium/heavy-water clad in aluminum. The dimensions and moderator/fuel volume ratios for the lattice experiments mentioned above are shown in Table 1.

In the benchmark experiments, the following integral parameters were measured at the center of each lattices:

ρ^{28}: the ratio of the epithermal to thermal capture reaction rates in ^{238}U
δ^{25}: the ratio of the epithermal to thermal fission reaction rates in ^{235}U
δ^{28}: the ratio of the total fission reaction rates in ^{238}U and ^{235}U
C^*: the ratio of the capture reaction rates in ^{238}U and fission reaction rates in ^{235}U, which is measured in the TRX lattice only.

The thermal cutoff energy for the parameters defined above is 0.625eV.

For the heavy water benchmark experiments, only one value of δ^{28} for ZEEP-1 was measured. Therefore, for the rest of lattices the ENDF/B-V benchmark test results[9] for ZEEP experiments are comprared with present computations.

3. Processed Data

Processed nuclides in ENDF/B-VI for the present study are as indicated in Table 2. The total cross section of ^{241}Pu is chosen to verify computational results using various computers. All other nuclides are used for the calculations of benchmark problems.

In the group averaging processes, a combined of the Maxwellian, 1/E and fission spectrum is used for weighting Doppler broadened pointwise data. The thermal portion is Maxwellian with a temperature of 0.0235eV which joins 1/E at 0.1eV. The fission spectrum joins 1/E at 674.0keV and has a characteristic temperature of 1.27MeV. In NJOY code[7], a weighing flux of the actinide nuclides is produced by a pointwise solution of the slowing down equations for the heavy absorber mixed with light moderator in order to take account of broad and intermediate resonance effects in the epithermal region accurately. The fine-group weighting fluxes calculated by the above method are used for weighting actinides; ^{235}U and ^{238}U.

The thermal scattering kernels for most nuclides are computed with the free gas model with some exceptions which will be pointed out later in this report. The special thermal scattering kernels for hydrogen bound in light water and deuterium bound in heavy water are computed using the scattering law data in ENDF/B-VI format.[8] The transport cross sections for all nuclides, except hydrogen and deuterium, are computed using the row sum correction from P_1 scattering matrix data. For the above two, a weighted column sum correction is applied to group above 4eV. For most nuclides P_1 data are only stored implicitly in WIMS library in the form of transport corrections. But for the principal moderators, explicit P_1 matrix data are attached to the end of WIMS library. Only three P_1 matrices are processed. These are for hydrogen, deuterium and oxygen. The Goldstein-Cohen parameters are taken from the reported data[1], which vary smoothly with atomic weights and are assumed to have energy independent values, without detailed evaluations. The potential scattering cross sections are processed from the source data files. The (n,2n) cross sections are subtracted from absorption cross section to conserve total cross sections which is equal to the sum of the rest of cross sections.

Fission spectra of are generated from Watt spectrum data in ENDF/B-VI file. ^{235}U and ^{238}U have the self-shielded resonance integral data computed for one temperature and seven backgroud cross sections. Brief descriptions of NJOY processing information for each nuclides are described below. In order to verify the acceptability of the WIMS library generated using PC-486-based NJOY, all inputs used in the present study are exactly identical to data used by Kim, et. al.[10]

Uranium-238: The data are tabulated at the temperature of 300°K. The self-shielded cross sections are calculated at the Bondarenko σ_0 values of 0.1, 1, 10, 50, 10^2, 10^3 and 10^{10} barns. Only the absorption resonance integrals are tabulated. To calculate the self-shielded cross sections

above 367.26eV the Narrow Resonance approximation is used. Between 0.1eV - 367.26eV, the flux calculator in NJOY code is applied which can treat the slowing down across resonances much more accurately. For fully-shielded scattering cross sections the Bondarenko σ_0 of 10 barns is used

Uranium-235: The data are tabulated at the temperature of 300°K. The self-shielded cross sections are calculated at the Bondarenko σ_0 values of 1, 10, 10^3, 10^4, 10^5 and 10^{10} barns. To calculate the self-shielded cross sections above 48.052eV the Narrow Resonance approximation is used. Between 0.1eV - 48.052eV, the flux calculator in NJOY code is applied.

Aluminum-27: The group constants are calculated at the temperature of 300°K. All cross sections are assumed self-shielded at the Bondarenko σ_0 value of 10^{10} barns.

Oxygen-16: The group constants are calculated at the temperature of 300°K. All cross sections are assumed to be self-shielded at the Bondarenko σ_0 value of 10^{10} barns. The P_1 scattering matrix at 300°K is inserted in WIMS library.

Hydrogen bound in light water and Deuterium bound in heavy water: The group constants are calculated at two temperatures of 296°K and 350°K. The temperature grid is dictated by the available temperature mesh in the thermal scattering law library in ENDF/B-VI format. P_1 scattering matrices at 296°K and 350°K are inserted in WIMS library, but no transport correction to the self-scattering terms of scattering matrices are made for both nuclides.

4. Computations

For the calculations of benchmark problems PC-486 Version WIMS compiled using Lahey F77L/EM32 Version 5.20, which is also used for NJOY compiliation, is used. The inputs are identical to the recommendations of the WIMS Library Update Projects[2] for consistency. The comparison of computed results of PC-486 Version WIMS and CYBER Version WIMS for typical inputs with same library shows that there are no significant differences for k_{eff} and k_∞ as well as other parameters but last digit in 7-digit numbers at most. Therefore, PC-486 Version WIMS is considered to be identical to main frame version.

A set of the total cross section of ^{241}Pu is generated from BROADR module in NJOY using CYBER-960-13/VE, Work Station SUN Sparc 1$^+$ and PC-486 and by PC-486 Version of ENDF Pre-Processing System [LINEAR/RECENT/SIGMA1][11] for a series of checks of the processed results assuming that the total cross sections of ^{241}Pu is a typical and upon obtaining acceptable result, then there are no problems in group constants for other nuclides as well.

Two set of WIMS library are generated from ENDF/B-VI on PC-486; one with no transport corrections in order to compare the results reported by Kim, et. al.[10] where no corrected scattering data are used and one with transport corrections for ^{16}O, ^{27}Al, ^{235}U and ^{238}U. The computation scheme for the library preparation is depicted in Fig.1. In Fig.1, WIMSKR program is written for getting proper group constants for WIMS library using the output file generated by GROUPR module in NJOY code.

5. Results and Discussion

5.1. Comparison of Total Cross Section of Plutonium-241

Figs.2~6 show the machine- and the processing system-dependancies of the Doppler broadened total cross sections at 300°K for ^{241}Pu. Close observations of the maximum difference in Fig.2 indicates that it happens because of truncated energy grid number. Therefore, it can be judged that the most of differences in the resonance region may caused by similar problems. However, it seems that the result from ENDF Pre-Processing System is more close to CYBER Version NJOY result compared with PC Version NJOY result. It can also be pointed out that, with same error criteria, ENDF Pre-Processing System run on PC-486 is much faster than NJOY run on PC-486.

5.2. Benchmarks for Light Water Lattices

Table 3 shows that the computed results of both k_∞ and k_{eff} values obtained using CYBER- and PC-486-based libraries along with the measured k_{eff} values with measurement errors for TRX and BAPL experiments. It shows most of computed k_{eff} values are ouside of experimental error bands except a few cases. However, the computed results from both machines are within 0.02% at maximum difference. It can be concluded that PC-486 NJOY results are well acceptable.

Table 4 shows the computed results by NJOY runs on CYBER and PC-486 for various experimentally available parameters. Both computed results of ρ^{28} value are very close together, but are somewhat better predicted with uncorrected library. For other parameters, the computed results are ouside of the experimental errors. Figs.7~11 show the 69-group constant differences generated by CYBER-NJOY and PC-486-NJOY for σ_a, σ_s and σ_{tr} of ^1H and σ_a and $\nu\sigma_f$ of ^{238}U. The rest of group constants including other nuclides used are either identical or less than ± 1.0001 in ratio for a few groups so that above mentioned differences may come from differences in group constants of ^1H and ^{238}U. The differences in scattering and transport cross sections of hydrogen in thermal group may come from the different version of NJOY. Kim, et. al.[10] used NJOY87 while the present study adapted NJOY91.38. Since Hydrogen is moderator in this case, the differences in k_{eff} may come from these deviations in addition to $\nu\sigma_f$ differences of ^{238}U as shown in Fig.11. Larger group constant of absorption cross section in thermal group of ^{238}U may

contribute the differences in other parameters. The reason why certain group constants and nuclides are different while other group constants are identical has not been searched yet. However, PC-486 results seems better compared with CYBER results. It can be concluded that PC-486 Version NJOY is well acceptable.

Since the transport corrected library generated using CYBER Version NJOY is not available, the transport corrected libraries generated from ENDF/B-VI using SUN Sparc 1$^+$ and PC-486 are compared with each other. Since the present study is aimed for building PC-486-based ENDF processing utilizing NJOY code, such a comparison would be required. The values of k_{eff} for light water lattices computed by SUN and PC-486 have relative errors of 0.14% to 0.23% as shown in Table 5. Since group constants of all other nuclides are almost identical except ^{1}H and ^{238}U, again the group constants of ^{238}U may cause the differences. The values for ρ^{28}, δ^{25}, δ^{28} and C^* are more deviated from the experimental results as shown in Table 6, but well within experimental errors. Therefore, both results are judged acceptable.

5.3. Benchmarks for Heavy Water Lattices

The values of k_{eff} for heavy water lattices computed by SUN and PC-486 are very close together as shown in Table 7. However, the computed values for k_{eff} are not within the experimental uncertainties given for ZEEP lattices. The deviations between SUN and PC-486 results may come from the relatively large differences in deuterium group constants in high energy groups as shown in Fig.12~14. As before, Figs.12~16 show the compared results for those differences which are beyond $\pm$1.0001 in ratio. All other group constants are either identical or the differences are negligible so that these are not presented here.

For other parameters, the experimental results are not available except δ^{28} for ZEEP-1. For this value, both computed results are well within experimental errors. For the rest of parameters, the ENDF/B-V test results[9] are given in order to compare with the present computations. As shown in Table 8, based on the computed results with ENDF/B-VI it can be said that ENDF/B-VI-based libraries are better for both machines.

6. Conclusion

There exist some problems to the group constants for Hydrogen, Deuterium and Uranium-238 generated from ENDF/B-VI using PC-486 as discussed in previous sections. However, the difference may be not too much deviated for general cases. As summarized in Table 9, the geometrical mean square errors for all light water lattices from experimental values for both no-shielded and fully-shielded ^{238}U scattering cross sections are approximately less than 1%. Overall performance of PC-486-based WIMS library is either comparable or better than CYBER- or SUN-based libraries even though the differences are

minor.

However, it may be needed to investigate the reasons why the group constants for Hydrogen, Deuterium and Uranium-238 are different from each other. When the problems are resolved, PC-486 Version NJOY can be utilized for generating WIMS library for advanced PWRs as well as CANDU reactors as intended in the present study. At the same time, ENDF Pre-Processing System on PC-486 can be combined with NJOY when the required computation time is matter.

If the runing time on computer is important, it may be pointed out that the runing times of NJOY code for ^{235}U and ^{238}U on CYBER 960-13/VE are 30,105 seconds and 83,130 seconds, respectively while these are 47,590 seconds and 116,310 seconds on PC-486(CPU clock rate at 33MHz). Therefore, PC-486 run takes only about 1.5 times more compared with CYBER.

References

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9) Craig, D.S., Testing ENDF/B-V Data For Thermal Reactors, AECL7690(Rev.1), June 1984
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Table 1. Physical Details of Benchmark Experiments

Lattice	Outer Radius(cm)				Moderator/Fuel Volume Ratio
	Fuel	Void	Clad	Latice Spacing	
TRX-1	0.49150	0.50420	0.5753	1.8060	2.35
TRX-2	0.49150	0.50420	0.5753	2.1740	4.02
BAPL-1	0.48640	0.50420	0.5753	1.5578	1.43
BAPL-2	0.48640	0.50420	0.5753	1.6523	1.78
BAPL-3	0.48640	0.50420	0.5753	1.8057	2.40
ZEEP-1	1.62850	1.64700	1.7490	10.5008	40.43
ZEEP-2	1.62850	1.64700	1.7490	7.3348	19.13
ZEEP-3	1.62850	1.64700	1.7490	6.3320	13.97
SRL 1-7-I	1.26745	1.30302	1.3843	9.3352	53.13
SRL 1-8-I	1.26745	1.30302	1.3843	10.7755	71.13
SRL 1-9-II	1.26745	1.30302	1.3843	12.4424	95.23

Table 2. Nuclides and Identification Number Used in This Study

Nuclide	Identification Number	
	ENDF/B-VI	WIMS LIBRARY
1-H - 1	125	1001[1]
1-H - 2	128	1002[1]
8-O - 16	825	8016
13-Al- 27	1325	13027
92-U -235	9228	92235
92-U -238	9237	92238
		92338[2]
94-Pu-241	9443	---

1) Scattering Law Data(H in H_2O & D in D_2O) in ENDF/B-VI
2) Fully-Shielded Scattering Cross Sections

Table 3. Comparison of Multiplication Factors Computed by CYBER and PC-486

[No-Shileded U-238 Scattering]

Lattice	k-infinite			k-effective			
	CYBER	PC-486	(CY-PC)/CY (%)	Experiment	CYBER	PC-486	(CY-PC)/CY (%)
TRX-1	1.1807	1.1803	0.03387	1.0±0.00260	1.0006	1.0003	0.02998
TRX-2	1.1646	1.1643	0.02575	1.0±0.00091	0.9984	0.9981	0.03004
BAPL-1	1.1391	1.1388	0.02633	1.0±0.00060	1.0012	1.0010	0.01997
BAPL-2	1.1432	1.1429	0.02624	1.0±0.00060	1.0005	1.0003	0.01999
BAPL-3	1.1298	1.1295	0.02655	1.0±0.00040	1.0000	0.9998	0.02000

[Fully-Shileded U-238 Scattering]

Lattice	k-infinite			k-effective			
	CYBER	PC-486	(CY-PC)/CY (%)	Experiment	CYBER	PC-486	(CY-PC)/CY (%)
TRX-1	1.1748	1.1767	-0.16172	1.0±0.00260	0.9908	0.9909	-0.01009
TRX-2	1.1616	1.1621	-0.04304	1.0±0.00091	0.9924	0.9923	0.01007
BAPL-1	1.1402	1.1373	0.25434	1.0±0.00060	0.9952	0.9952	0.0
BAPL-2	1.1445	1.1417	0.24464	1.0±0.00060	0.9951	0.9951	0.0
BAPL-3	1.1310	1.1286	0.21220	1.0±0.00040	0.9960	0.9959	0.01004

Table 4. Comparison of Calculated and C/E Values by CYBER and PC-486

[Table 4-1. Ratio of Epithermal to Thermal U-238 Capture Rate(ρ^{28})]

Lattice	Experiment	No-Shielded U-238 Scattering				Fully-Shielded	
		CYBER		PC-486		CYBER	PC-486
		CALCULATED	C/E	CALCULATED	C/E	C/E	C/E
TRX-1	1.320±0.016	1.3226E+0	1.0020	1.3246E+0	1.0035	1.0224	1.0241
TRX-2	0.837±0.010	8.3098E-1	0.9928	8.3223E-1	0.9943	1.0111	1.0128
BAPL-1	1.390±0.007	1.4075E+0	1.0126	1.4089E+0	1.0136	1.0209	1.0221
BAPL-2	1.120±0.009	1.1742E+0	1.0484	1.1736E+0	1.0479	1.0551	1.0563
BAPL-3	0.906±0.011	9.2189E-1	1.0175	9.2283E-1	1.0186	1.0251	1.0263

[Table 4-2. Ratio of Epithermal to Thermal U-235 Fission Rate(δ^{25})]

Lattice	Experiment	No-Shielded U-238 Scattering				Fully-Shielded	
		CYBER		PC-486		CYBER	PC-486
		CALCULATED	C/E	CALCULATED	C/E	C/E	C/E
TRX-1	0.0987±0.0100	9.4549E-2	0.9579	9.4805E-2	0.9605	0.9736	0.9761
TRX-2	0.0614±0.0130	5.8149E-2	0.9471	5.8308E-2	0.9496	0.9607	0.9633
BAPL-1	0.0840±0.0240	8.0229E-2	0.9551	8.0431E-2	0.9575	0.9613	0.9637
BAPL-2	0.0680±0.0150	6.5511E-2	0.9634	6.5677E-2	0.9658	0.9694	0.9719
BAPL-3	0.0520±0.0190	5.0419E-2	0.9700	5.0548E-2	0.9720	0.9751	0.9776

[Table 4-3. Ratio of U-238 to U-235 Fission Rate(δ^{28})]

Lattice	Experiment	No-Shielded U-238 Scattering				Fully-Shielded	
		CYBER		PC-486		CYBER	PC-486
		CALCULATED	C/E	CALCULATED	C/E	C/E	C/E
TRX-1	0.0946±0.043	9.6099E-2	1.0158	9.6239E-2	1.0173	1.0281	1.0280
TRX-2	0.0693±0.051	6.8430E-2	0.9874	6.8455E-2	0.9878	0.9940	0.9940
BAPL-1	0.0780±0.051	7.4574E-2	0.9561	7.4597E-2	0.9564	0.9624	0.9625
BAPL-2	0.0700±0.057	6.4120E-2	0.9160	6.4138E-2	0.9163	0.9213	0.9214
BAPL-3	0.0570±0.053	5.2564E-2	0.9222	5.2579E-2	0.9224	0.9261	0.9262

[Table 4-4. Ratio of U-238 Capture to U-235 Fission Rate(C^*)]

Lattice	Experiment	No-Shielded U-238 Scattering				Fully-Shielded	
		CYBER		PC-486		CYBER	PC-486
		CALCULATED	C/E	CALCULATED	C/E	C/E	C/E
TRX-1	0.797±0.010	7.8708E-1	0.9876	7.8773E-1	0.9884	0.9976	0.9986
TRX-2	0.647±0.009	6.3687E-1	0.9843	6.3731E-1	0.9850	0.9918	0.9926
BAPL-1	-	8.0898E-1	-	8.0946E-1	-	-	-
BAPL-2	-	7.3747E-1	-	7.3788E-1	-	-	-
BAPL-3	-	6.5931E-1	-	6.5965E-1	-	-	-

Table 5. Comparison of Multiplication Factors Computed by SUN WS and PC-486

[No-Shileded U-238 Scattering]

Lattice	k-infinite			k-effective			
	SUN	PC-486	(SU-PC)/SU (%)	Experiment	SUN	PC-486	(SU-PC)/SU (%)
TRX-1	1.1829	1.1803	0.220	1.0±0.00260	0.9978	0.9956	0.221
TRX-2	1.1659	1.1642	0.146	1.0±0.00091	0.9943	0.9929	0.141
BAPL-1	1.1415	1.1387	0.246	1.0±0.00060	1.0002	0.9979	0.230
BAPL-2	1.1452	1.1429	0.201	1.0±0.00060	0.9987	0.9967	0.201
BAPL-3	1.1312	1.1295	0.151	1.0±0.00040	0.9975	0.9961	0.141

[Fully-Shileded U-238 Scattering]

Lattice	k-infinite			k-effective			
	SUN	PC-486	(SU-PC)/SU (%)	Experiment	SUN	PC-486	(SU-PC)/SU (%)
TRX-1	1.1794	1.1767	0.229	1.0±0.00260	0.9884	0.9863	0.213
TRX-2	1.1637	1.1620	0.146	1.0±0.00091	0.9884	0.9871	0.132
BAPL-1	1.1400	1.1373	0.237	1.0±0.00060	0.9943	0.9921	0.222
BAPL-2	1.1440	1.1417	0,201	1.0±0.00060	0.9934	0.9915	0.192
BAPL-3	1.1303	1.1285	0.160	1.0±0.00040	0.9937	0.9922	0.151

Table 6. Comparison of Calculated and C/E Values by SUN WS and PC-486

[Table 6-1. Ratio of Epithermal to Thermal U-238 Capture Rate(ρ^{28})]

Lattice	Experiment	No-Shielded U-238 Scattering				Fully-Shielded	
		SUN SPARC 1+		PC-486		SUN	PC-486
		CALCULATED	C/E	CALCULATED	C/E	C/E	C/E
TRX-1	1.320±0.016	1.3088E+0	0.9915	1.3248E+0	1.0036	1.0119	1.0243
TRX-2	0.837±0.010	8.2362E-1	0.9840	8.3237E-1	0.9945	1.0014	1.0130
BAPL-1	1.390±0.007	1.3919E+0	1.0014	1.4091E+0	1.0137	1.0098	1.0223
BAPL-2	1.120±0.009	1.1597E+0	1.0355	1.1737E+0	1.0480	1.0439	1.0565
BAPL-3	0.906±0.011	9.1226E-1	1.0069	9.2293E-1	1.0187	1.0145	1.0264

[Table 6-2. Ratio of Epithermal to Thermal U-235 Fission Rate(δ^{25})]

Lattice	Experiment	No-Shielded U-238 Scattering				Fully-Shielded	
		SUN SPARC 1+		PC-486		SUN	PC-486
		CALCULATED	C/E	CALCULATED	C/E	C/E	C/E
TRX-1	0.0987±0.0100	9.4815E-2	0.9606	9.4817E-2	0.9607	0.9762	0.9763
TRX-2	0.0614±0.0130	5.6986E-2	0.9281	5.8316E-2	0.9498	0.9635	0.9634
BAPL-1	0.0840±0.0240	8.0418E-2	0.9574	8.0438E-2	0.9576	0.9636	0.9638
BAPL-2	0.0680±0.0150	6.5672E-2	0.9658	6.5682E-2	0.9659	0.9718	0.9719
BAPL-3	0.0520±0.0190	5.0549E-2	0.9721	5.0551E-2	0.9721	0.9776	0.9777

[Table 6-3. Ratio of U-238 to U-235 Fission Rate(δ^{28})]

Lattice	Experiment	No-Shielded U-238 Scattering				Fully-Shielded	
		SUN SPARC 1+		PC-486		SUN	PC-486
		CALCULATED	C/E	CALCULATED	C/E	C/E	C/E
TRX-1	0.0946±0.043	9.6015E-2	1.0150	9.6244E-2	1.0174	1.0257	1.0282
TRX-2	0.0693±0.051	6.8357E-2	0.9864	6.8435E-2	0.9875	0.9923	0.9938
BAPL-1	0.0780±0.051	7.4389E-2	0.9537	7.4576E-2	0.9561	0.9599	0.9622
BAPL-2	0.0700±0.057	6.3979E-2	0.9140	6.4113E-2	0.9159	0.9192	0.9211
BAPL-3	0.0570±0.053	5.2468E-2	0.9205	5.2552E-2	0.9220	0.9244	0.9258

[Table 6-4. Ratio of U-238 Capture to U-235 Fission Rate(C^*)]

Lattice	Experiment	No-Shielded U-238 Scattering				Fully-Shielded	
		SUN SPARC 1+		PC-486		SUN	PC-486
		CALCULATED	C/E	CALCULATED	C/E	CALC	CALC
TRX-1	0.797±0.010	0.7824	0.9816	0.7878	0.9885	0.7904	0.7959
TRX-2	0.647±0.009	0.6316	0.9761	0.6374	0.9851	0.6389	0.6422
BAPL-1	-	0.8037	-	0.8095	-	0.8073	0.8131
BAPL-2	-	0.7332	-	0.7379	-	0.7361	0.7409
BAPL-3	-	0.6560	-	0.6597	-	0.6582	0.6619

Table 7. Comparison of Multiplication Factors Computed by SUN WS and PC-486 for ZEEP and SRL Lattices

	k-infinite				k-effective				
	no-shielded		fully-shielded		Experiment	no-shielded		fully-shielded	
	SUN	PC486	SUN	PC486		SUN	PC486	SUN	PC486
ZEEP-1	1.2363	1.2354	1.2316	1.2307	1.0±0.0020	0.9998	0.9992	0.9963	0.9957
ZEEP-2	1.1814	1.1798	1.1726	1.1708	1.0±0.0006	1.0071	1.0059	0.9993	0.9980
ZEEP-3	1.1405	1.1384	1.1293	1.1271	1.0±0.0047	1.0120	1.0103	1.0014	0.9997
SRL(1-7-I)	1.2333	1.2325	1.2300	1.2292	1.0	0.9980	0.9974	0.9954	0.9948
SRL(1-8-I)	1.2352	1.2346	1.2327	1.2321	1.0	0.9974	0.9970	0.9956	0.9951
SRL(1-9-II)	1.2276	1.2271	1.2258	1.2253	1.0	0.9979	0.9976	0.9966	0.9963

Table 8. Comparison of Experiment, ENDF/B-V Test and Computed Values for ρ^{28}, δ^{25}, δ^{28} and C^* for ZEEP and SRL Lattices

			ZEEP-1	ZEEP-2	ZEEP-3	SRL(1-7-I)	SRL(1-8-I)	SRL(1-9-II)
ρ28	ENDF/B-V Result		2.8200E-1	5.1600E-1	6.8800E-1			
	No-Shielded	SUN	2.5503E-1	4.6150E-1	6.1149E-1	2.1341E-1	1.7320E-1	1.4523E-1
		PC	2.5735E-1	4.6638E-1	6.1828E-1	2.1559E-1	1.7486E-1	1.4648E-1
	Fully-Shielded	SUN	2.6878E-1	4.9117E-1	6.5323E-1	2.2270E-1	1.8019E-1	1.5049E-1
		PC	2.7123E-1	4.9647E-1	6.6068E-1	2.2499E-1	1.8190E-1	1.5183E-1
δ25	ENDF/B-V Result		2.6300E-2	5.0200E-2	6.7400E-2			
	No-Shielded	SUN	2.4285E-2	4.5970E-2	6.1384E-2	1.9079E-2	1.5105E-2	1.2230E-2
		PC	2.4248E-2	4.5907E-2	6.1306E-2	1.9055E-2	1.5086E-2	1.2210E-2
	Fully-Shielded	SUN	2.5477E-2	4.8557E-2	6.5037E-2	1.9785E-2	1.5635E-2	1.2627E-2
		PC	2.5439E-2	4.8505E-2	6.4975E-2	1.9765E-2	1.5615E-2	1.2611E-2
δ28	Experimental Result		6.7500E-2 ±0.0014					
	ENDF/B-V Result		6.8200E-2	7.2500E-2	7.6400E-2			
	No-Shielded	SUN	6.7228E-2	7.0679E-2	7.4043E-2	5.4086E-2	5.3401E-2	5.2660E-2
		PC	6.7272E-2	7.0775E-2	7.4178E-2	5.4119E-2	5.3427E-2	5.2683E-2
	Fully-Shielded	SUN	6.7512E-2	7.1299E-2	7.4914E-2	5.4239E-2	5.3516E-2	5.2758E-2
		PC	6.7548E-2	7.1400E-2	7.5059E-2	5.4274E-2	5.3539E-2	5.2773E-2
C*	No-Shielded	SUN	8.2184E-1	9.4296E-1	1.0288E+0	7.9662E-1	7.7230E-1	7.5536E-1
		PC	8.2339E-1	9.4617E-1	1.0332E+0	7.9806E-1	7.7340E-1	7.5619E-1
	Fully-Shielded	SUN	8.2987E-1	9.5973E-1	1.0519E+0	8.0216E-1	7.7649E-1	7.5852E-1
		PC	8.3150E-1	9.6319E-1	1.0566E+0	8.0368E-1	7.7763E-1	7.5942E-1

Tablb 9. Geometrical Mean Square Errors in Percent of Computed Values Compared with Light Water Lattice Experimental Results

Using Uncorrected Library	No-Shielded		Fully-Shielded	
	CYBER	PC-486	CYBER	PC-486
	0.976	0.957	0.911	0.902
Using Corrected Library	No-Shielded		Fully-Shielded	
	SUN	PC-486	SUN	PC-486
	1.007	0.959	0.866	0.906

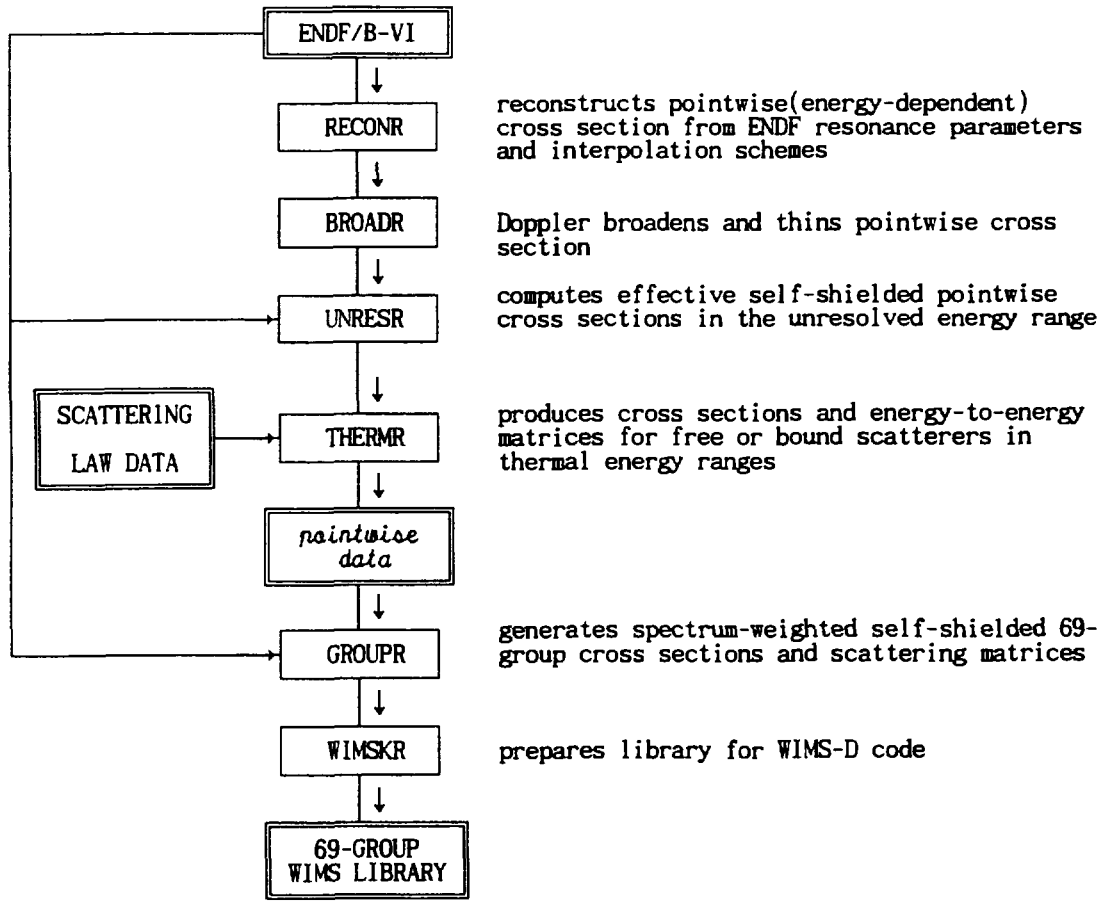

Fig.1. Flow Diagram for Generating 69-group Library for WIMS Code

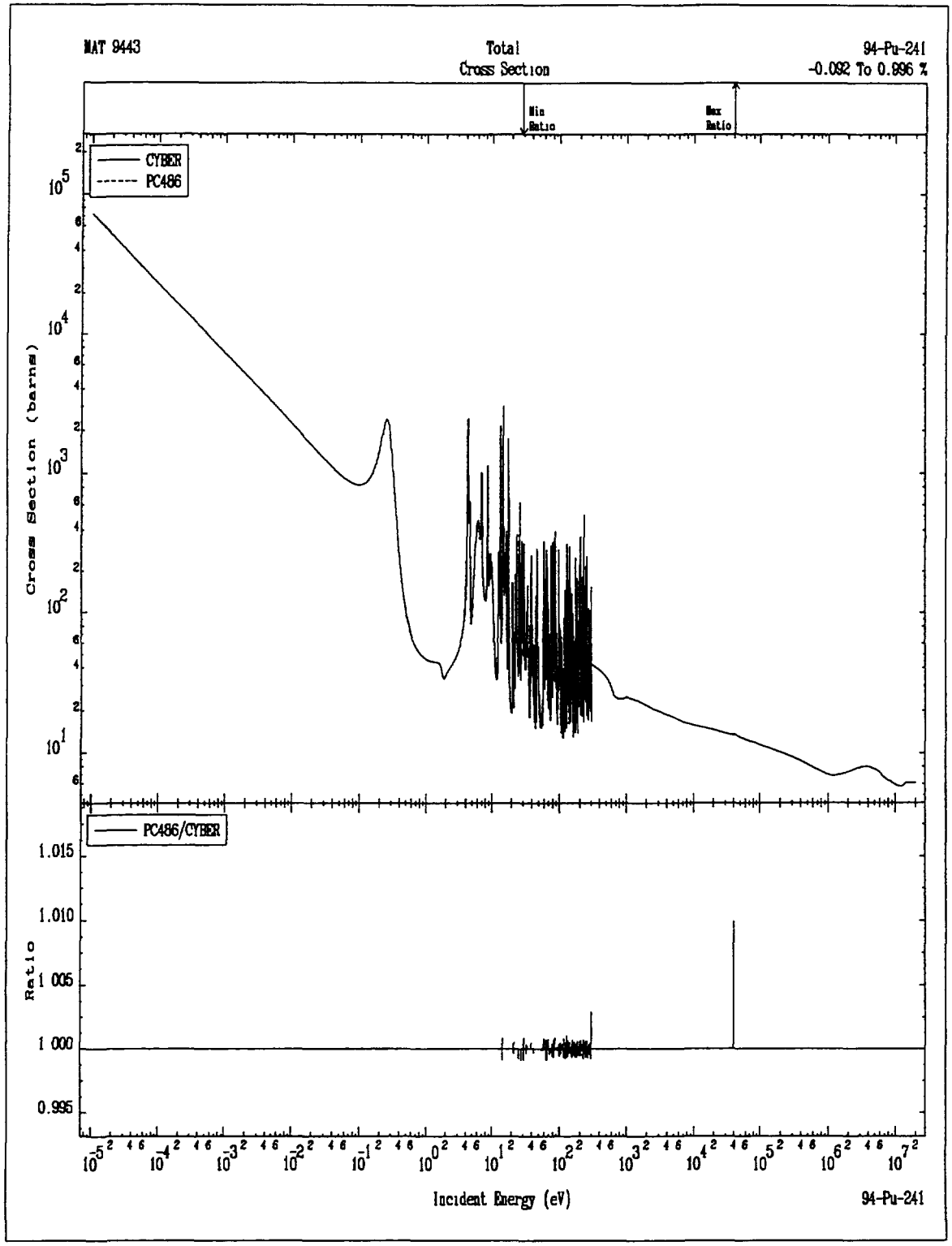

Fig. 2. Comparison of Total Cross Secion of ^{241}Pu Computed by CYBER and PC-486

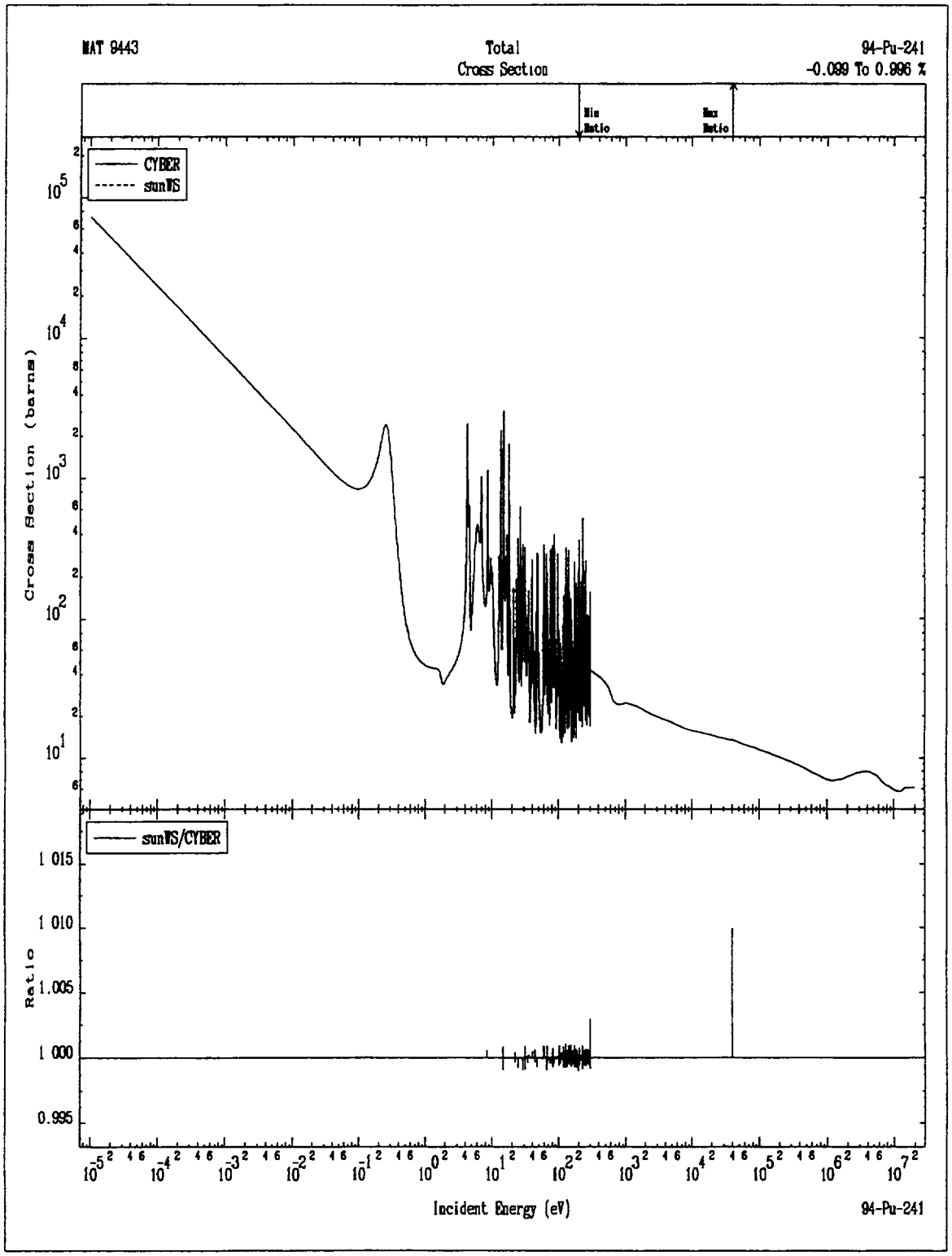

Fig. 3. Comparison of Total Cross Secion of ^{241}Pu Computed on CYBER and SUN Work Station

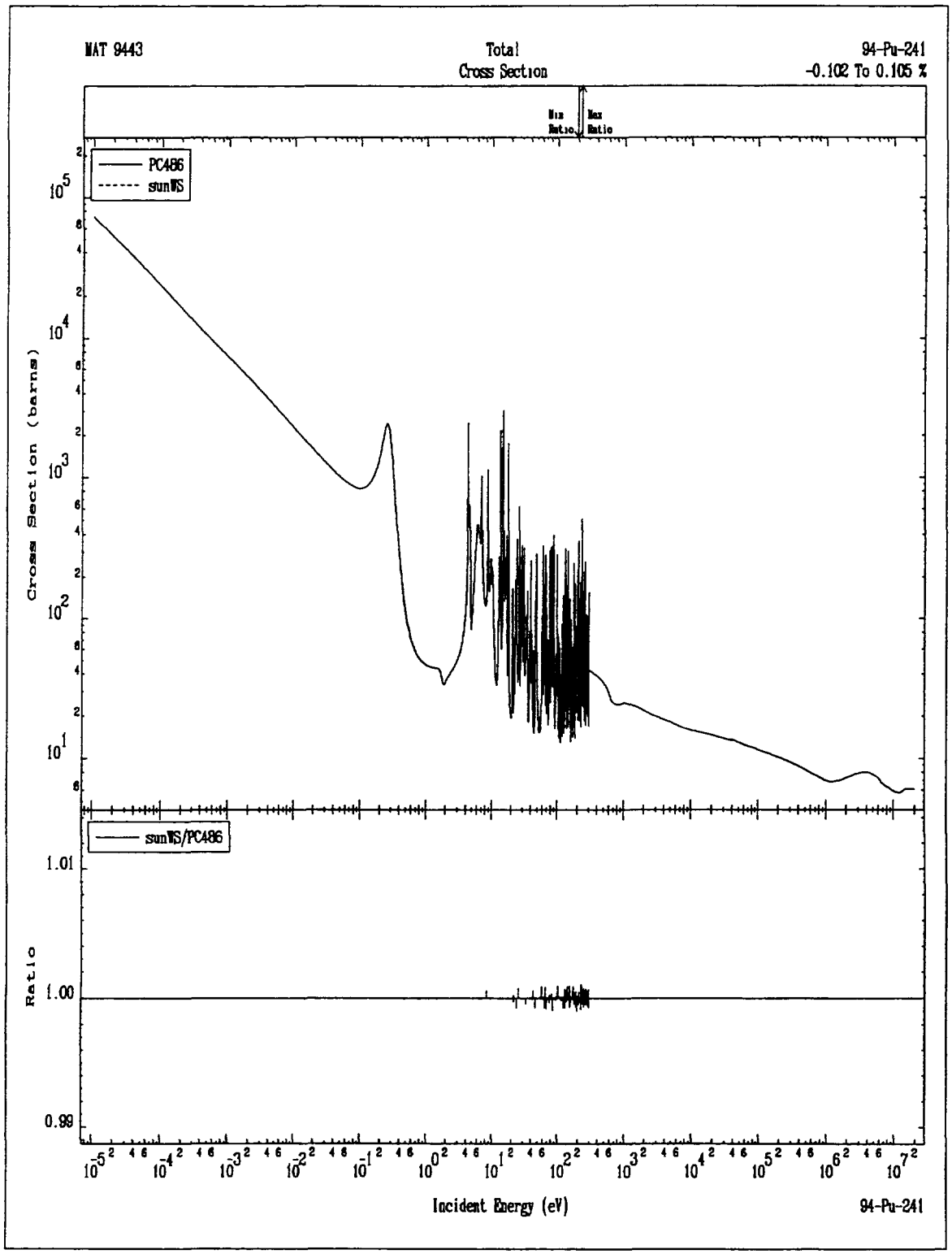

Fig. 4. Comparison of Total Cross Secion of ^{241}Pu Computed on PC-486 and SUN Work Station

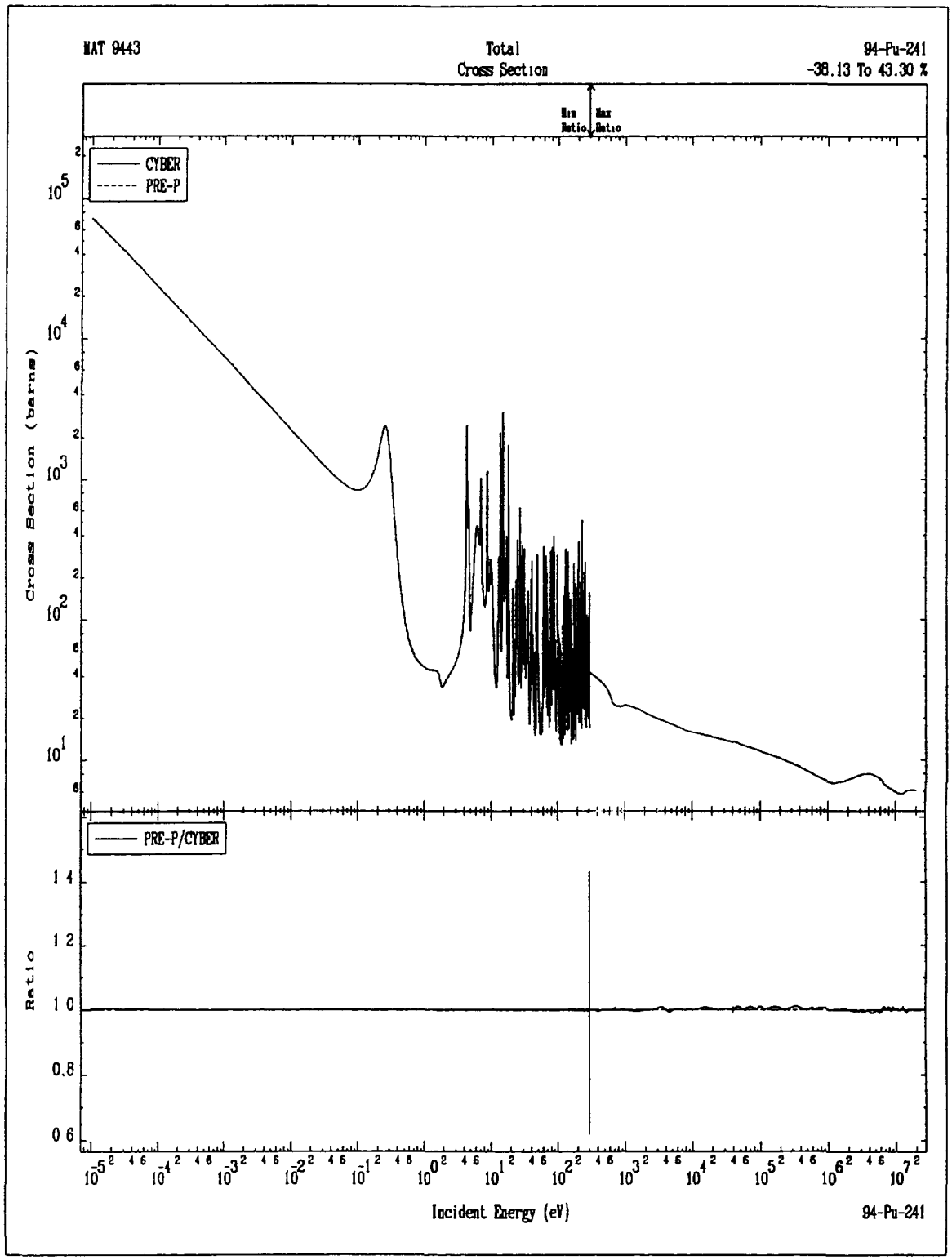

Fig. 5. Comparison of Total Cross Secion of ^{241}Pu Computed on CYBER-NJOY and PC-486-ENDF Pre-Processing System

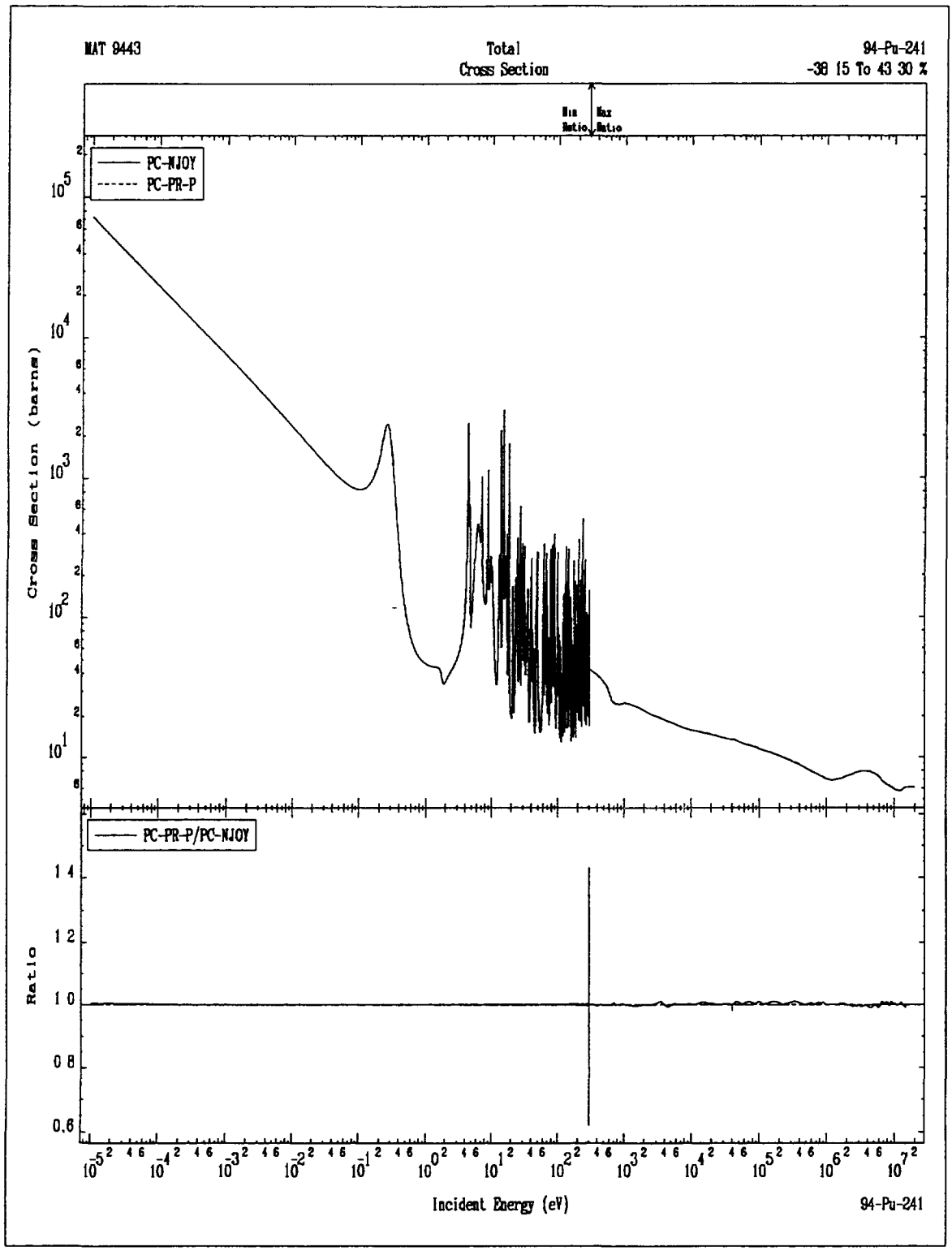

Fig. 6. Comparison of Total Cross Secion of ^{241}Pu Computed on PC-NJOY and PC-486-ENDF Pre-Processing System

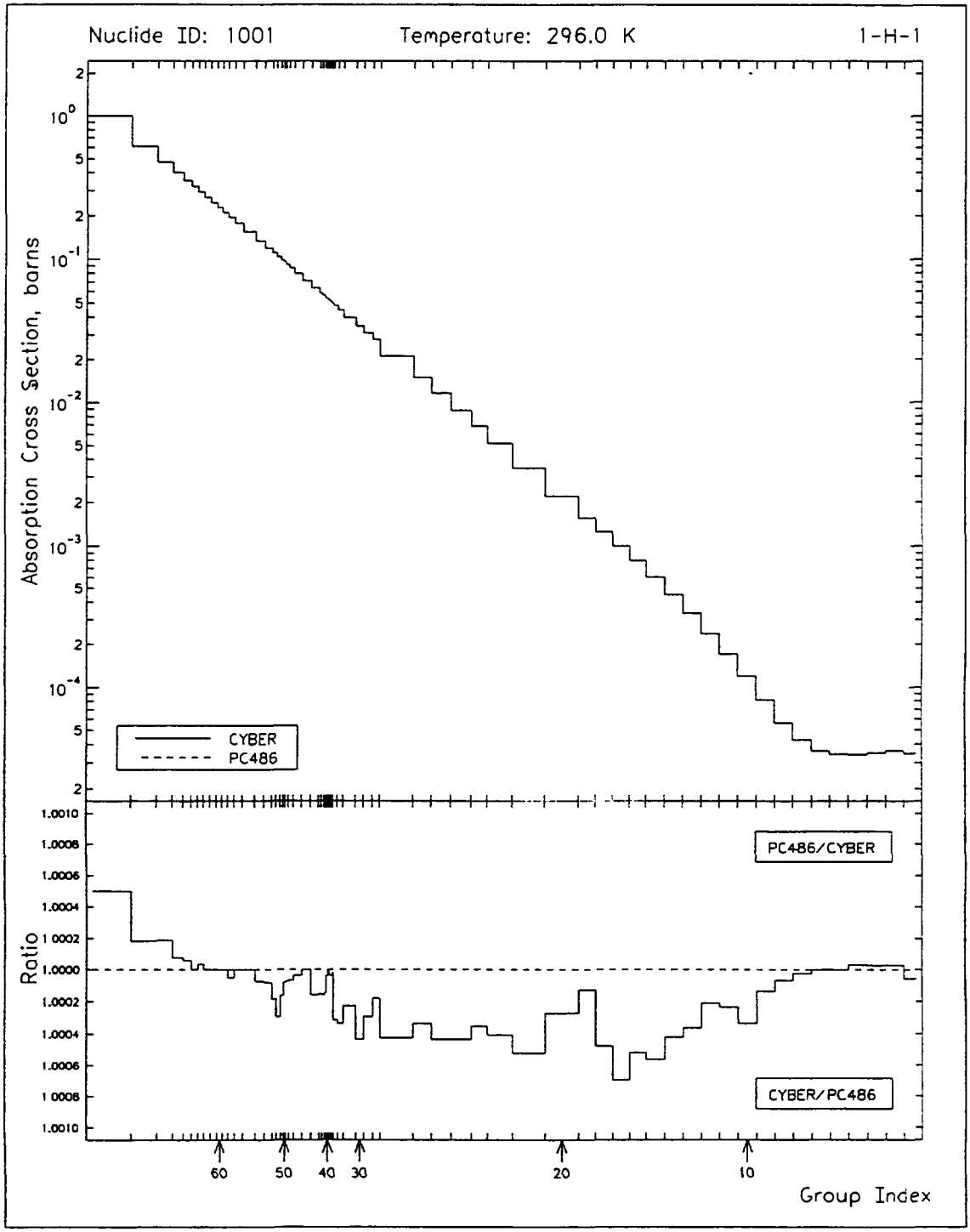

Fig. 7 Comparison of 69-group Absorption Cross Secions for Hydrogen Bound in Ligh Water Computed by CYBER and PC-486

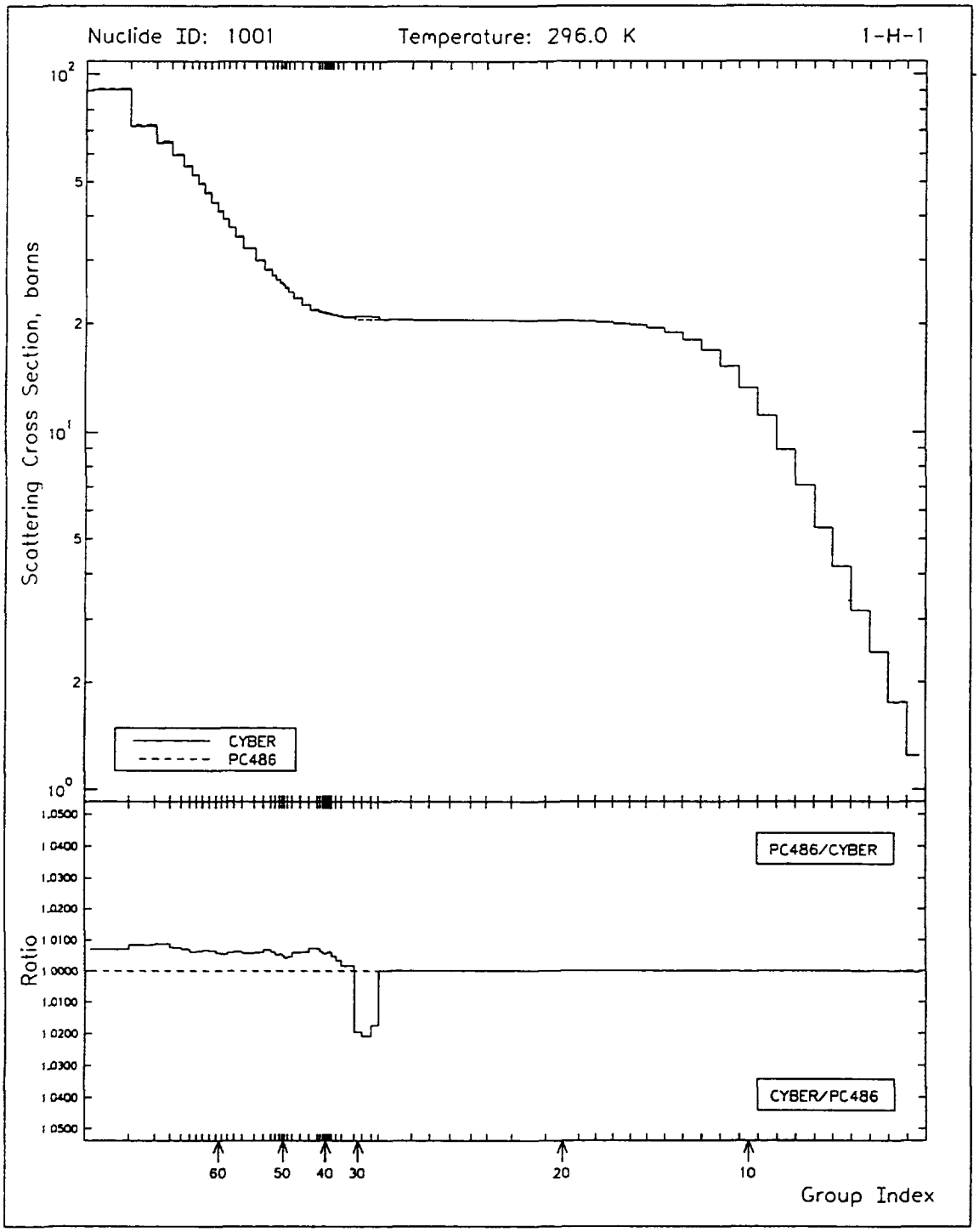

Fig. 8 Comparison of 69-group Scattering Cross Secions for Hydrogen Bound in Ligh Water Computed by CYBER and PC-486

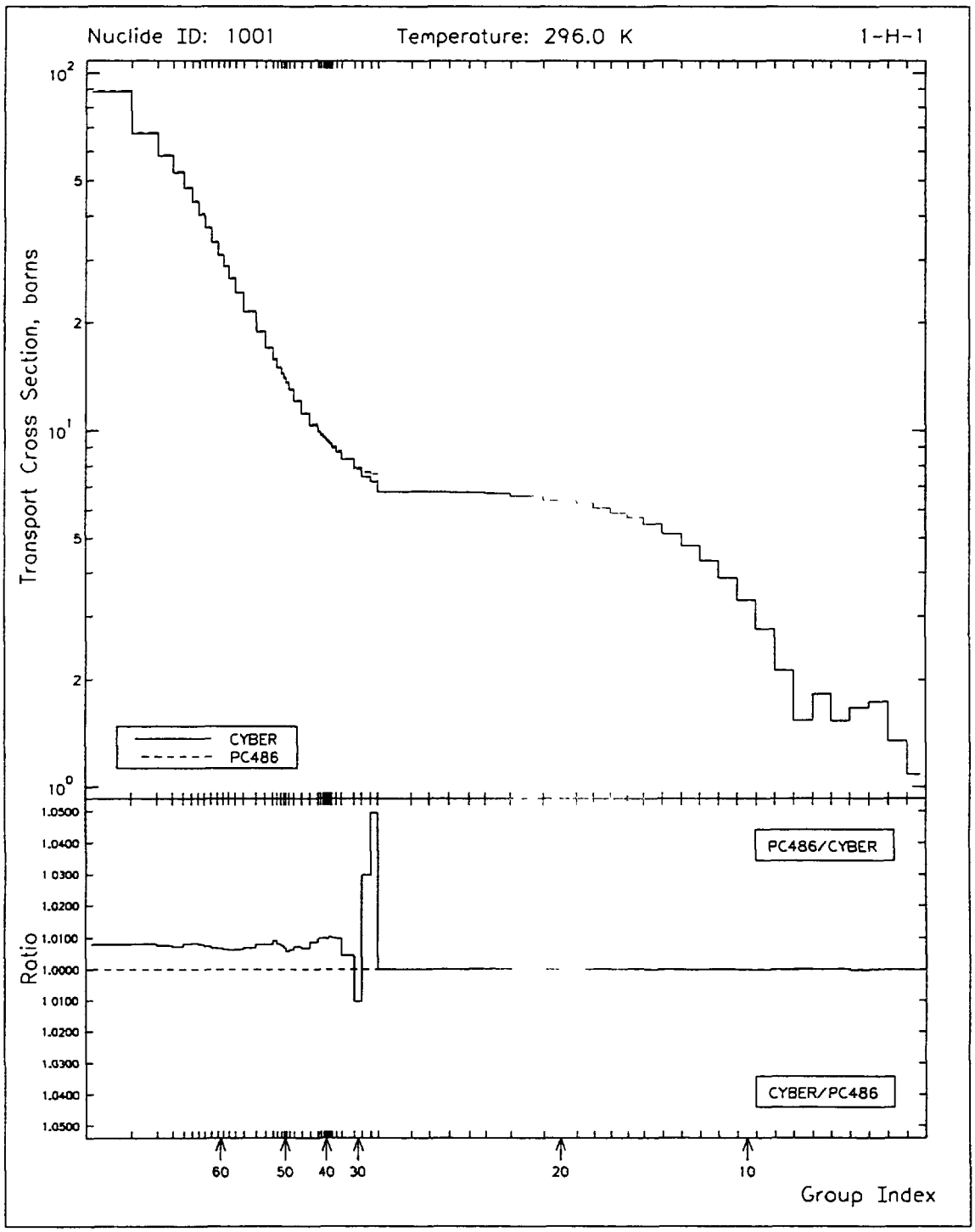

Fig.9 Comparison of 69-group Transport Cross Secions for Hydrogen Bound in Ligh Water Computed by CYBER and PC-486

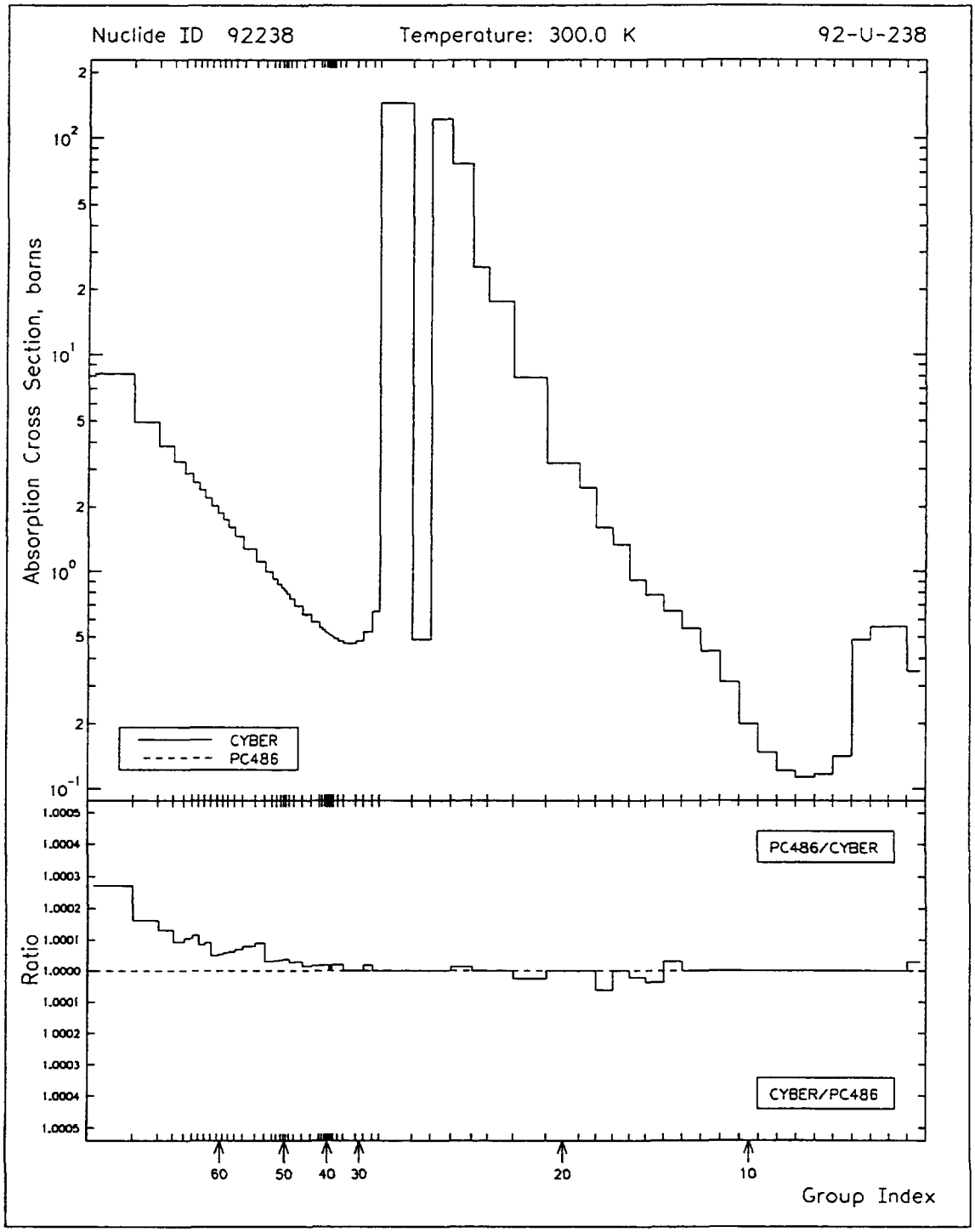

Fig. 10 Comparison of 69-group Absorption Cross Secions for ^{238}U Computed by CYBER and PC-486

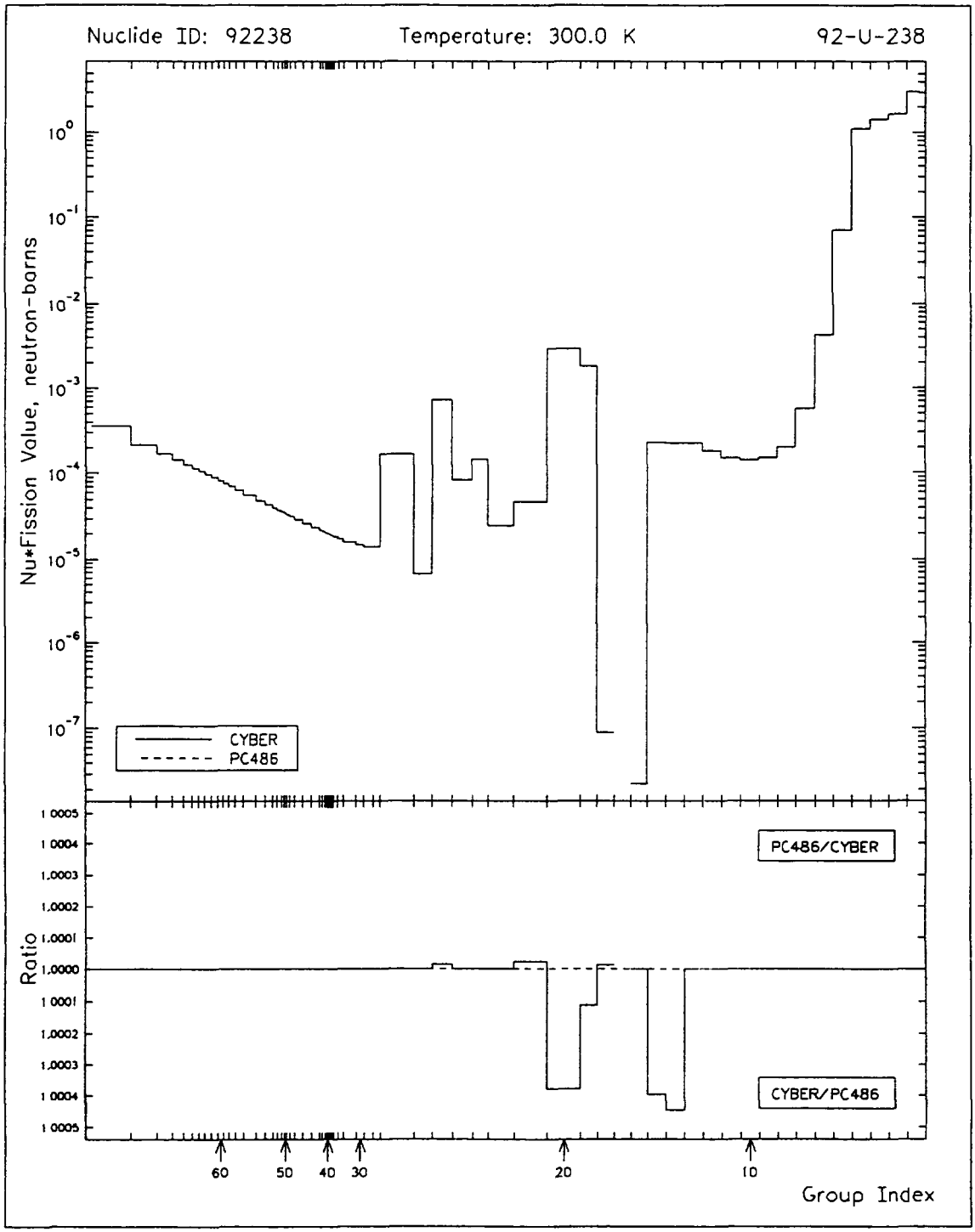

Fig. 11 Comparison of 69-group $\nu\sigma_f$ Values for ^{238}U Computed by CYBER and PC-486

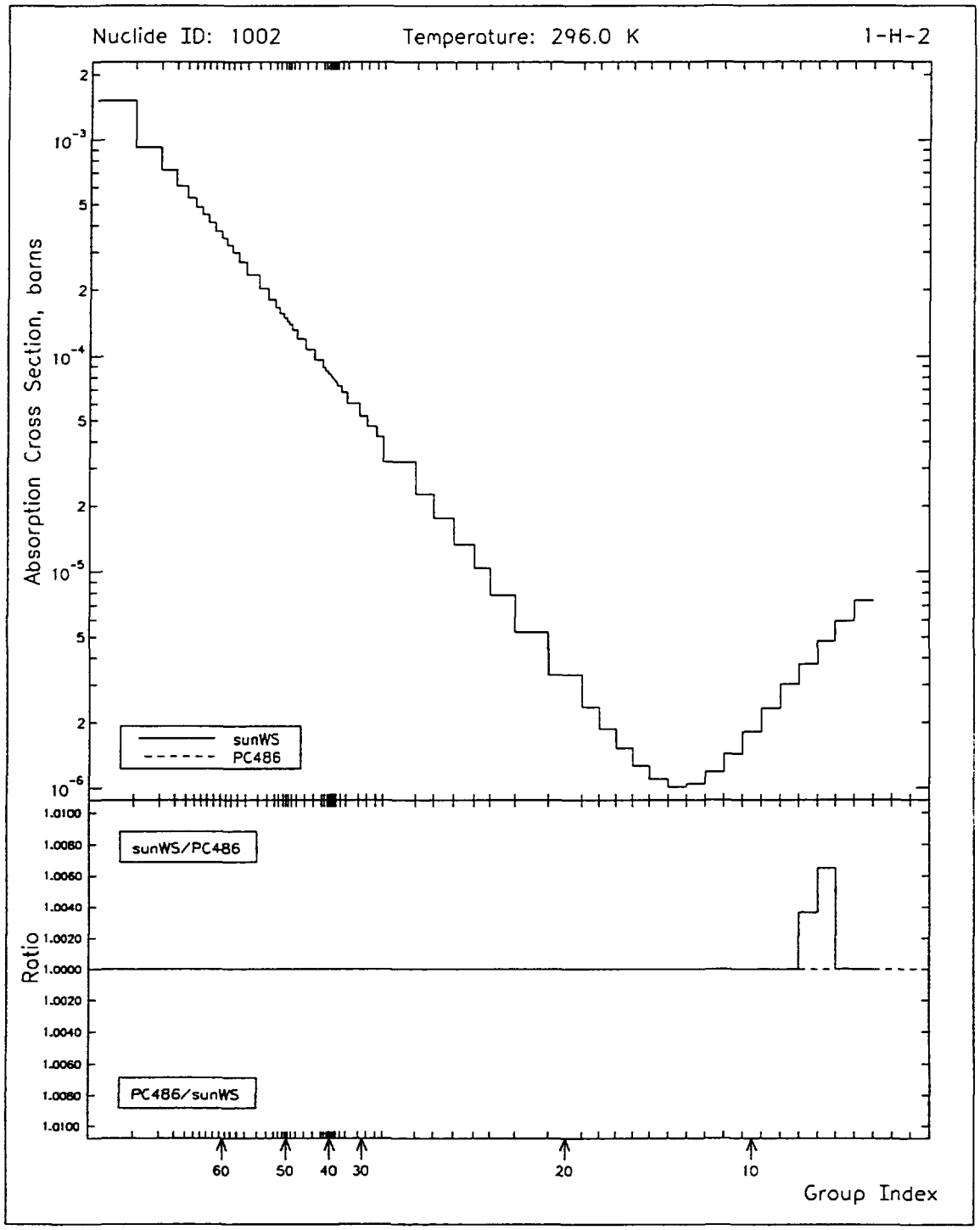

Fig. 12 Comparison of 69-group Absorption Cross Secions for Deuterium Bound in Heavy Water Computed by SUN and PC-486

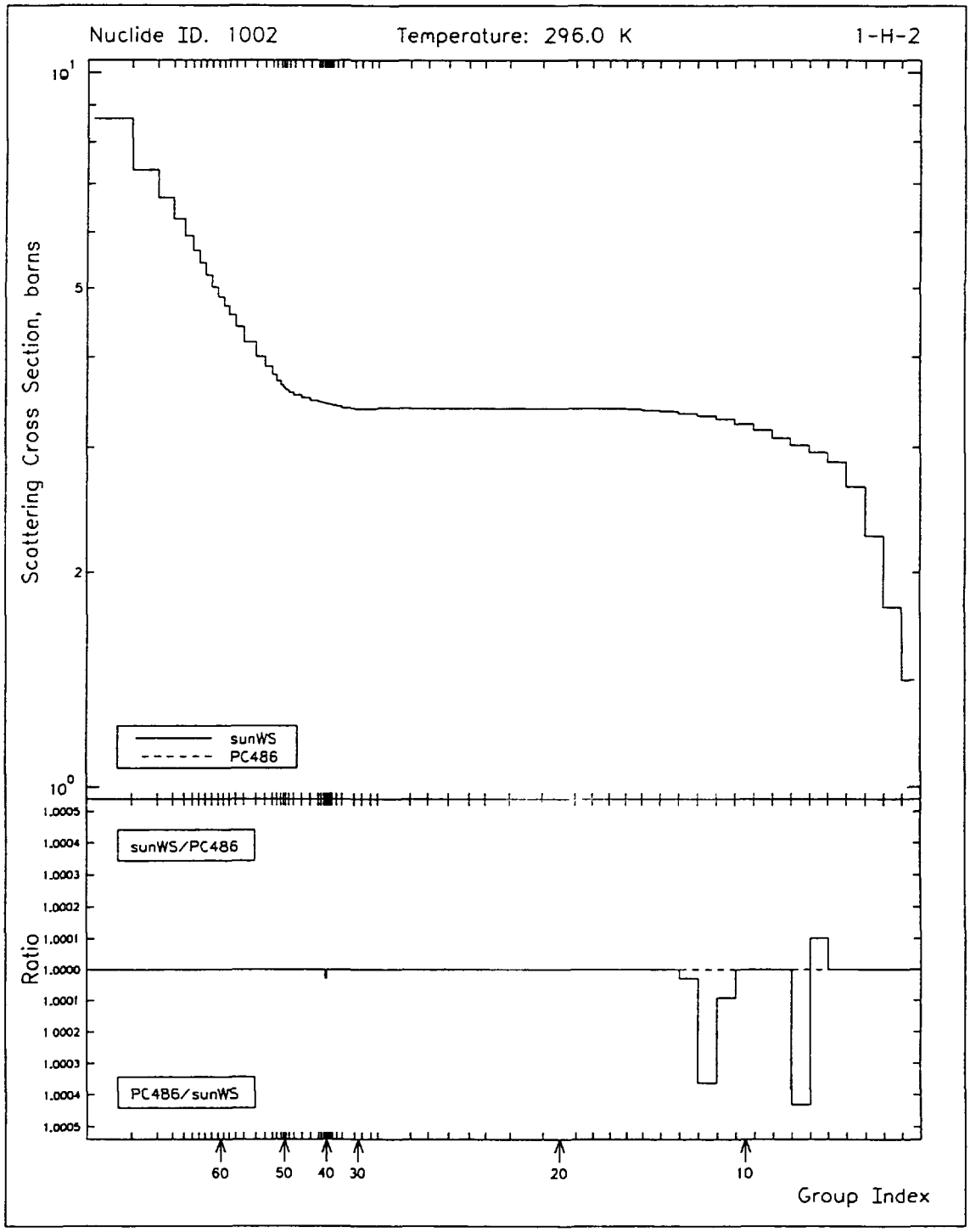

Fig.13 Comparison of 69-group Scattering Cross Secions for Deuterium Bound in Heavy Water Computed by SUN and PC-486

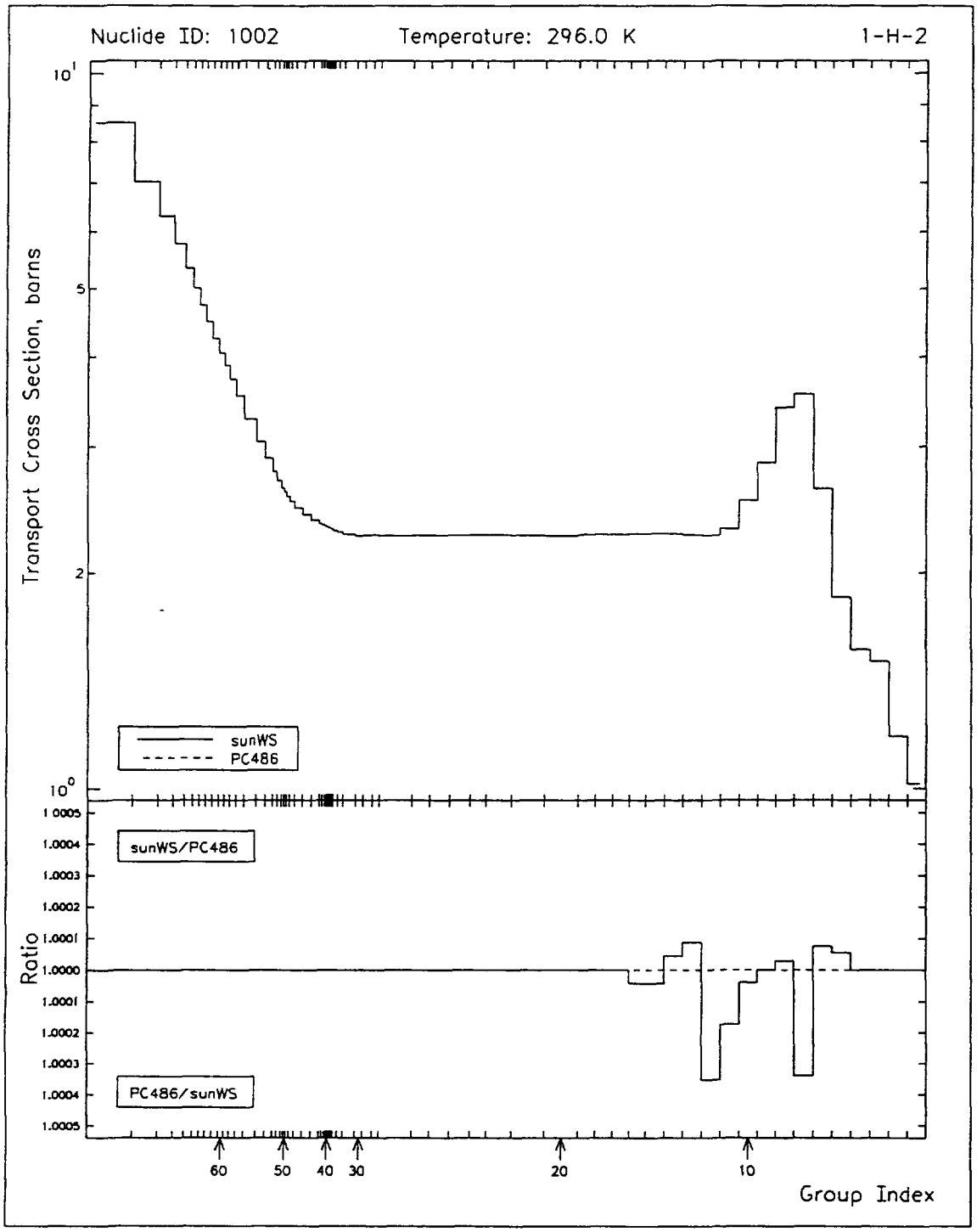

Fig. 14 Comparison of 69-group Transport Cross Secions for Deuterium Bound in Heavy Water Computed by SUN and PC-486

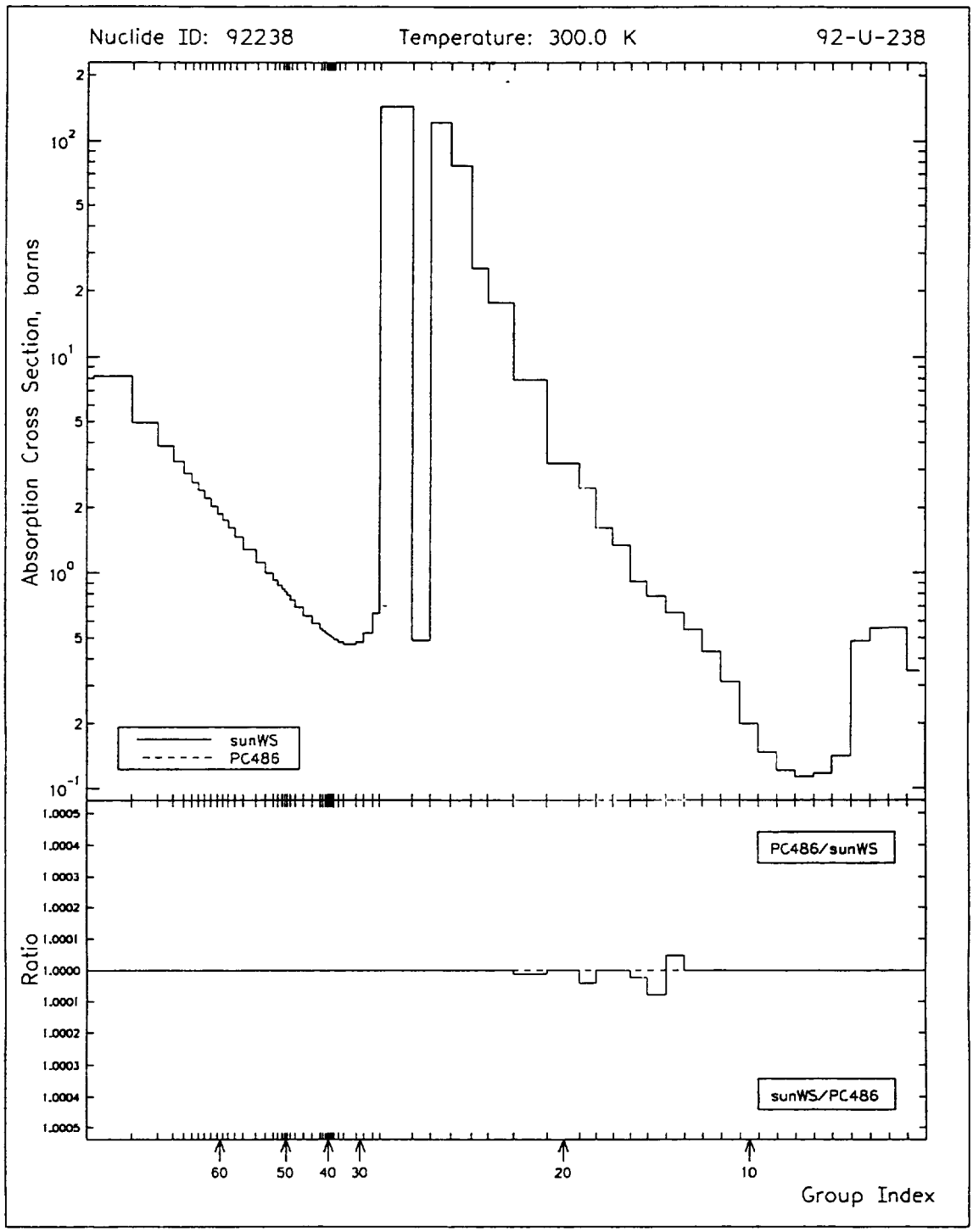

Fig. 15 Comparison of 69-group Absorption Cross Sections for ^{238}U Computed by SUN and PC-486

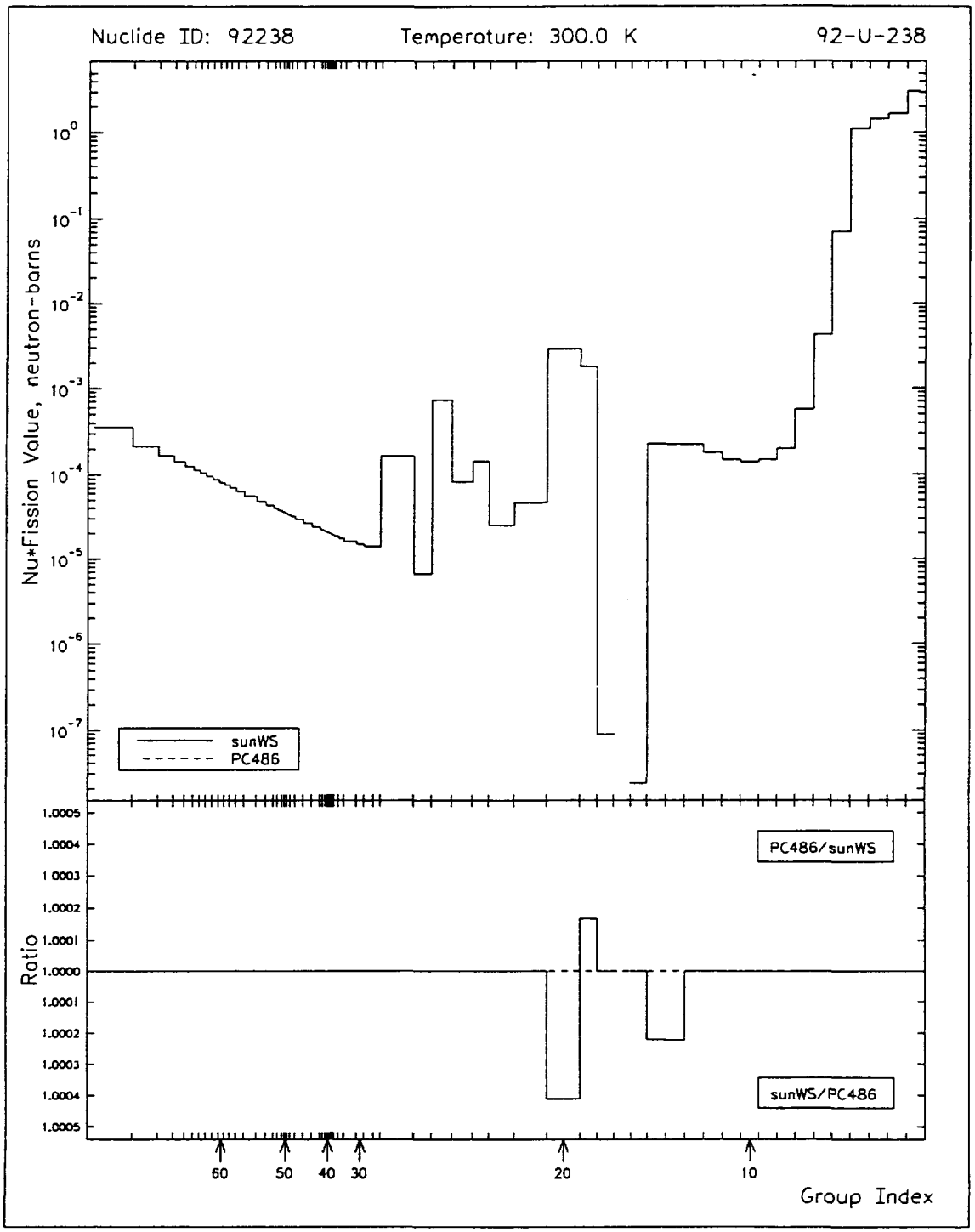

Fig. 16 Comparison of 69-group $\nu\sigma_f$ Values for ^{238}U Computed by SUN and PC-486

GROUP CONSTANT PREPARATION FOR THE ESTIMATE OF NEUTRON INDUCED DAMAGE IN STRUCTURAL MATERIALS

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Abstract

Neutron heating (kerma), displacement per atom cross sections (DPA), gas and γ-ray production are important parameters for the estimate of the damage produced by neutron induced nuclear reactions in the structural materials. The NJOY System for Nuclear Data Processing has been extensively used in order to compute the above quantities; here the theory, the algorithms and the connected problems are described.

Introduction

In the framework of the European Fusion Technology Programme NET, group damage data have been calculated for nuclides of interest in the design of shielding blanket of magnetic fusion devices. The source of the nuclear data is mainly EFF (European Fusion File), a subset of JEF (Joint Evaluated File), but some materials were taken from JENDL-3 and from ENDF/B-VI too.

The need of reviewing damage data comes from the availability of new nuclear data bases and of new processing codes by means of which a new library could be generated in place of the existing response function library, MACKLIB-IV /1/, which is based upon ENDF-B IV. The new data are available whithin the new format ENDF-6 which allows the inclusion of distributions of secondary particles - also charged particles - by means of double differential data (DDX) in File 6 (MF 6). After comparison with the result of MACK code/2/, the NJOY 3 System was used to basically process all the data; ancillary codes were also written for particular tasks within the Programme.

The damage data are here treated into the following classes:

1. Neutron kerma
2. DPA
3. Gas production cross sections
4. γ-ray production

Neutron kerma calculation

An important parameter in the nuclear machine design is the heating of the materials induced by the kinetic energy of charged particles produced by nuclear reactions.

The nuclear heating is composed by three parts:

1. the neutron heating, proportional to the neutron flux and induced by the kinetic energy of charged particles resulting from nuclear reactions;

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2. the photon heating, proportional to the flux of secondary photons produced in the nuclear reactions; this part is assumed to be deposited out of the site of the reaction and its contribution is subtracted from the neutron kerma. The computation of the photon heating assumes the presence of the gamma-ray production data in the source file;

3. the radioactive decay heating of isotopes produced in the neutron reactions leading to time dependent heating.

NJOY takes into account only the first two parts of the heating, i.e. the prompt heating.

At a given neutron energy E the neutron induced heating rate, $H_n(E)$, on a site around the incident particle impact for a given material can be expressed by:

$$H_n(E) = K(E)\phi(E) \qquad [1]$$

where $\phi(E)$ is the neutron flux, K(E) is the total neutron kerma factor defined as:

$$K(E) = \sum_{r=1}^{NR} K_r(E) \qquad [2]$$

and the sum over the partial kerma $K_r(E)$ is carried out over all the NR reactions induced by neutrons on the material at energy E. Each partial kerma is defined

$$K_r(E) = \sum_{l=1}^{NL} \overline{E}_{rl}(E)\sigma_r(E) \qquad [3]$$

where $\overline{E}_{rl}$ is the kinetic energy of l^{th} charged particle emitted in the reaction r, σ_r is the related cross section and the sum is carried out over all the NL charged products (including the recoil nucleus).

Eq.[3] represents the "direct method" for computing kerma factors and can be used only when charged particle distributions are available; in ENDF-6 terms this kind of data is given in the File 6 of double differential data in angle and energy (DDX). These data are included for not all materials in the nuclear data libraries, while in most cases neutron and photon spectral data only are given (for some nuclides photon production data are even missing).

When DDX are not given, the kinetic energy of the charged particles emitted in the reaction r is calculated by subtracting the energy carried away by neutrons and photons emitted in the reaction from the available energy. Eq.[3] is written as:

$$K_r(E) = (E + Q_{0r} - \overline{E}_{nr} - \overline{E}_{\gamma r})\sigma_r(E) \qquad [4]$$

Here E is the neutron incident energy, Q_{0r} is the mass-difference Q-value for the reaction r, $\overline{E}_{nr}$ is the energy of the secondary neutrons, including the multiplicity, and $\overline{E}_{\gamma r}$ is the energy of the photons, including the photon yield. Eq.[4] is the "energy balance method" for computing kerma factors. The total kerma K(E) [2] , should in principle be calculated as sum of partial kerma defined in eq.[4], but this can be performed only when neutron and photon spectral data are available for each separate reaction, which in general is not true. In fact, usually, photon production data are given in the lumped nonelastic (MT = 3) and inelastic (MT = 4) reactions. For that it is better to redefine K(E) in a more useful way:

$$K(E) = \sum_{r=1}^{NN} K_n(E) - \sum_{r'=1}^{NG} \sigma_{\gamma r'}(E)\overline{E}_{\gamma r'} \qquad [5]$$

where the first sum is carried out over all the NN reactions for which neutron (or possibly charged particle) production data are available and the second one is carried out over all the NG reactions for which photon production data are available. By this method the total kerma is always defined, while all the partial kerma factors are not;

moreover because of inconsistencies in the evaluations between cross section data of MF = 3 and photon production data of MF = 12 and MF = 13, K(E) is frequently incorrectly computed producing negative values for some nuclides.

When HEATR (the module of NJOY for kerma calculations) computes kerma factors for neutron induced reactions on a given material from nuclear data libraries in ENDF-6 format the algorithm of eq.[5] is used, where $K_n(E)$ is computed from DDX data in MF = 6, if any, by direct method, [3] , otherwise by the energy balance method:

$$K_n(E) = (E + Q_{0r} - \bar{E}_{nr})\sigma_r(E) \qquad [6]$$

($\bar{E}_{nr} = 0$ for no secondary neutron emission reactions such as (n,p), (n,α) ecc.). Kerma of (n,γ) reaction is calculated by kinematic formula.

HEATR performs a print check of total kerma K(E), MT = 301, on a coarse energy grid, optionally the upper and lower total kinematic limits, $K_U(E)$ and $K_L(E)$, are printed too; the quantity N(E):

$$N(E) = K_{r=102}(E) + \sum_{\substack{r=1 \\ r\neq 102}}^{NN} K_{max_r}(E) \qquad [7]$$

(where K_{max} are defined below) can be written on PENDF tape as MT = 443 and can be used in place of MT = 301 when the computed kerma are completely wrong (normally this is the case when no photon production data are given in the evaluation - overestimated kerma - or the energy carried away by photons is larger than the available energy - underestimated kerma, even negative) $K_U(E)$ and $K_L(E)$ are respectively the sum of the upper, $K_{max}(E)$, and the lower, $K_{min}(E)$, kinematic limits of the partial reactions.

Kinematic limits by reaction

The quantities K_{min} and K_{max} for each reaction are defined in the following

Note: σ_r, K_n, K_{min}, K_{max} are function of the incident neutron energy and should be read as $\sigma_r(E)$, $K_n(E)$, etc . $Q_{0r} \equiv QM$, is the mass-difference Q-value, $Q_r \equiv QI$ is the reaction Q-value (QM and QI are given in the C1 and C2 fields of each MF = 3 section, respectively, following the ENDF-6 standard; the LR term too is defined in that frame).

r = all reactions when MF = 6 is given

$$K_{min} = K_{max} = K_n$$

r = (n,n)

$$K_{min} = K_{max} = K_n$$

r = (n, γ)

$$K_{min} = \sigma_r \frac{E}{A+1}$$

$$K_{max} = \sigma_r \frac{E}{A+1} + \frac{1}{2}\sigma_r\left[Q_r + \frac{AE}{A+1}\right]^2 \frac{1}{m_n c^2 (A+1)}$$

r = discrete levels for the emission of neutrons and charged particles

$$K_{min} = \sigma_r[E + Q_r - \bar{E}_n]$$

$$K_{max} = K_{min} \qquad LR = 0$$

$$K_{max} = \sigma_r[E + Q_{0r} - \bar{E}_n] \qquad LR \neq 0$$

r = (n,n′) continuum

$$K_{min} = \sigma_r \frac{(E + \bar{E}_n)}{A}$$

$$K_{max} = K_{min} \qquad LR = 0$$

$$K_{max} = \sigma_r[E + Q_{0r} - \bar{E}_n] \qquad LR \neq 0$$

r = (n,x)

$$K_{min} = \sigma_r \frac{E}{A}$$

$$K_{max} = K_n$$

r = (n,fission)

$$K_{min} = \sigma_r(E + Q_r - \frac{1}{2}\bar{E}_n - 15 \times 10^6)$$

$$K_{max} = K_n$$

r = (n,2n) $(AWR \geq 10)$

$$K_{min} = 0$$

$$K_{max} = \sigma_r \frac{(E + \bar{E}_n)}{A - 1}$$

$$K_{max} = K_n \text{ (for } K_n \geq 0 \wedge K_{max} > K_n)$$

r = (n,3n) $(AWR \geq 10)$

$$K_{min} = 0$$

$$K_{max} = \sigma_r \frac{(E + 2\bar{E}_n)}{A - 2}$$

$$K_{max} = K_n \text{ (for } K_n \geq 0 \wedge K_{max} > K_n)$$

$r = (n,2n)$ (AWR < 10)

$r = (n,3n)$ (AWR < 10)

$r = (n,nx)$

$$K_{min} = 0$$

$$K_{max} = K_n$$

The contribution of the radioactive decay to kerma

Neutron induced reactions can produce radioactive nuclei. The decay of product nuclides can generate in the material the same effects of neutron induced reactions, leading to the release of decay energy and damage of the lattice because of the recoil of the nucleus.

As outlined above the NJOY code computes the heating production and the neutron induced nuclear reaction damage taking into account only the prompt effects directly induced by the reaction and by its products and neglecting the delayed effects of the decay of the product nucleus. A method of computing an additional term to kerma to account for the contribution of radioactive decay has been adopted.

A radioactive nucleus decays in three ways:

by α emission,

by β decay with electron or positron emission or electronic capture;

by isomeric transition between isomeric states with γ emission.

α and β decays are often associated with γ emitted by de-excitation of the final product.

The decay kerma computation is straightforward: the nuclear data files include the scheme of radioactive nuclide production (MF 8) and the average energies released by the three decay types (MF 8, MT 457); the sum times the cross section for the production of the radioactive nuclide gives the decay contribution to neutron kerma:

$$K_r^d(E_n) = (\bar{E}_\alpha + \bar{E}_\beta)_d \sigma_r(E_n)$$

where r is the neutron induced reaction of the target which produces the radioactive nuclide d. Thus the total prompt + delayed kerma is written as:

$$K(E_n) = K^p(E_n) + K^d(E_n)$$

with $K^p(E_n)$ defined above. $\bar{E}_\gamma$ as contribution to the decay heating is not included, since it is assumed γ-rays transported outside.

One has to put a time limit within which to take into account the decay terms; we assumed only nuclides which decay within 10^4 secs. to contribute to heating.

The computer programme DECKER [4] has been written for the retrieval of the data from the radioactive decay file, for the computation and the group averaging of the results

DPA

DPA s are not directly generated by NJOY which only computes the damage energy E_d from which DPA can be obtained.

Following the theory, the number of displacements depends on the available energy E_d and the energy required to displace an atom from its position E_d. In the NRT [5] model the DPA is given by the Kinchin-Pease formula:

$$DPA = N_d$$

N_d is:

$$0 \quad \text{for} \quad 0 < E_a < E_d$$

$$1 \quad \text{for} \quad E_d < E_a < 2E_d$$

$$k\frac{E_a}{2E_d} \quad \text{for} \quad E_a > 2E_d$$

where k is the displacement efficiency and is assumed to be 0.8, E_d is given in the literature; for reliable work the values of Greenwood et al./6/ have been used.

Material	E_d
Be-9	31
C	31
Al-27	27
Si	25
Ti	40
V	40
Cr	40
Mn-55	40
Fe	40
Ni	40
Cu	40
Zr	40
Nb-93	40
Mo	60
W	90

Table 1. : List of E_d values from Greenwood

The computation of the DPA from decay proceeds through the evaluation of the damage energy, which results to be a function of the kinetic energy of the recoil nucleus following the nuclear reaction. Nevertheless it has been shown that the contribution of delayed DPA to prompt term can be beglected being only of some percent.

Gas production

Gas production data includes p, d, T, ^3He, α particle production, are computed either from point or from group averaged data. The final values are the result of the sum of the concerned reactions times the multiplicity factor, as for example in the n,n3α reaction. Three groups of reactions should be taken into account.

1. reactions whose primary product is, among others, the particle to be considered (e.g. for hydrogen production n,p, n,n'p, n,2p, n,pα etc.);

2. reactions for which the particle to be considered is a secondary product as in the inelastic levels (MT 51 to 90) and in the continuum (MT 91) with the appropriate LR flag;

3. reactions which produce the requested particle as residual nucleus

Primary product particles

A simple scan of the reaction definition as given in ENDF/B and of the related MT is needed in order to group together the contributing reactions and their yields: the choice is straightforward and undoubtful. The reactions included in calculating hydrogen, tritium and helium production are shown in the table.

Hydrogen			Tritium			Helium		
MT	Reaction	m	MT	Reaction	m	MT	Reaction	m
28	(n, n p)	1	33	(n, n t)	1	22	(n, n α)	1
32	(n, n d)	1	36	(n, n t 2α)	1	23	(n, n 3α)	3
33	(n, n t)	1	105	(n, t)	1	24	(n, 2n α)	1
35	(n, n d 2α)	1	113	(n, t 2α)	1	25	(n, 3n α)	1
36	(n, n t 2α)	1	116	(n, p t)	1	29	(n, n 2α)	2
103	(n, p)	1				30	(n, 2n 2α)	2
104	(n, d)	1				34	(n, n 2^3_2He)	2
105	(n, t)	1				35	(n, n d 2α)	2
111	(n, 2p)	2				36	(n, n t 2α)	2
112	(n, p α)	1				106	(n, $^3_2$He)	1
113	(n, t 2α)	1				107	(n, α)	1
114	(n, d 2α)	1				108	(n, 2α)	2
115	(n, p d)	1				109	(n, 3α)	3
116	(n, p t)	1				112	(n, p α)	1
						113	(n, t 2α)	2
						114	(n, d 2α)	2

Table 2. : Reactions contributing to gas production with the primary particle, m is the multiplicity factor.

Secondary product particles

The type of secondary product in the discrete inelastic levels and in the continuum (MT 51 to 91) is specified by the LR flag which indicates levels which de-excite by particle emission rather than by γ emission. The type of emitted particle is deduced by the value of LR which becomes the MT number, so as a level with LR = 22 produces one α, with LR = 28 one proton and so on.

Isotope	Inelastic levels	LR
$^6_3$Li	1-6, 8-31	32
$^7_3$Li	2	33
$^{10}_5$B	5, 6, 8-35	22
$^{12}_6$C	2-23, 41	22
$^{16}_8$O	6-8, 10-15, 17-19, 21, 22, 24, 26, 27, 29, 31, 32, 34, 36, 38	22
	16, 20, 23, 25, 28, 30, 33, 35, 37	28

Table 3. : Reactions contributing to gas production with the secondary particle emitted from discrete inelastic levels and identified by LR flag. Note that LR = 32 for $^6_3$Li, LR = 33 for $^7_3$Li and LR = 22 for $^{10}_5$B produce gas particles as residual nuclei, too

Particles as residual nucleus

A typical case is the well known reaction:

$$^{9}_{4}Be + n \rightarrow 2n + ^{8}_{4}Be$$

in which the final product (^{8}Be) decays within 2×10^{-15}s. in 2α, so as the n,2n reaction becomes a α producer.

$$^{9}_{4}Be\ (n, 2n)\ vs.\ (n, \alpha)$$

On the contrary reactions of the type:

$$^{9}_{4}Be + n \rightarrow \alpha + ^{6}_{2}He$$

in which ^{6}He decays β^- within 0.85s in ^{6}Li and :

$$^{6}_{3}Li + n \rightarrow \alpha + ^{1}_{3}H$$

have been neglected because by taking into account the secondary neutron impact one has to make some assumptions about the flux intensity and the geometry of the medium which is not the case in a general purpose library, i.e. not yet problem oriented.

Below 20 MeV only the following reactions can be found to contribute to gas production with residual nuclei:

$$^{3}_{2}He + n \rightarrow p + ^{3}_{1}H$$

$$^{3}_{2}He + n \rightarrow d + ^{2}_{1}H$$

$$^{6}_{3}Li + n \rightarrow 2n, \alpha + ^{1}_{1}H$$

$$^{6}_{3}Li + n \rightarrow t + ^{4}_{2}He$$

$$^{7}_{3}Li + n \rightarrow 2n, \alpha + ^{2}_{1}H$$

$$^{7}_{3}Li + n \rightarrow 3n, \alpha + ^{1}_{1}H$$

$$^{9}_{4}Be + n \rightarrow 2n + ^{8}_{4}Be \rightarrow 2\alpha$$

Photon production

In addition to prompt γ-ray production computed by NJOY, an improvement to DECKER was developed that allows to take into account γ's from decay for further transport calculations, adding them to the prompt ones in the γ production matrices

From the point of view of the input data to DECKER, monoisotopic materials greatly differ from naturally occuring elements: in fact isotopes do not need any preprocessing since all reactions producing radioactive nuclei are easily found in the basic data file, while for the mixtures it is necessary to investigate isotope by isotope which reactions falling within the 20 MeV boundary are suitable for production of delayed γ-ray. The cross section data for those reactions are leased from EAF, the European Activation File.

The method of calculation is based upon the formulas given in the following. The total γ emission, produced from reactions induced by neutrons is:

$$Q_{tot} = \sum_{r=1}^{NR} \int_{0}^{\infty} \sigma^{r}(E_n)\Phi(E_n)I^{r}_{tot}(E_n)dE_n \qquad [10]$$

The intensity I_{tot}^{r} of delayed γ produced from radioactive decay is independent from the neutron energy and can be taken out the symbol of integral. In terms of neutron energy group i and photon energy group j, eq. [10] becomes.

$$Q_{tot} = \sum_{r=1}^{NR} \sum_{i=1}^{NG} \bar{\sigma}_i^r \Phi_i \sum_{j=1}^{NGG} I_j^r \qquad [11]$$

Eq. [11] can be written as :

$$Q_{tot} = \sum_{r=1}^{NR} \sum_{i=1}^{NG} \sum_{j=1}^{NGG} P_{i\to j}^r \Phi_i \qquad [12]$$

By comparison between eqs. [12] and [11] we have the following definition of the γ production matrix $P_{i\to j}^r$:

$$P_{i\to j}^r = \bar{\sigma}_i^r I_j^r \qquad [13]$$

In the case of delayed γ production, I_j^r is defined (in terms of the quantities in MF = 8 MT = 457 of ENDF - see ENDF-102 manual[8]):

$$I_j^r = FD \sum_{k}^{E_k \in j} RI(E_k) + FC \int_{\Delta E_j} RP(E)dE \qquad [14]$$

where FD is the discrete spectrum normalization factor (absolute intensity/relative intensity); RI is the intensity of discrete radiation produced (relative units); FC is the continuum spectrum normalization factor; RP is the spectrum of the continuum component of the radiation in units of probability/eV such that

$$\int RP(E)dE = 1$$

The group averaged cross sections $\bar{\sigma}_i^r$ are retrieved from GENDF file or computed from EAF-3

For a given nuclide and for each reaction r, the delayed kerma [8] and γ production matrix [13] are stored on tape in GENDF format and will be used in the production of the RFL-2 library presently in progress.

To quantify the contribution of delayed to prompt photons the code compares the following quantities, defined both for delayed (DECKER output) and prompt γ (GENDF file):

$$P_i^r = \sum_{j=1}^{NGG} \bar{P}_{i\to j}^r \qquad [15]$$

i.e. the sum over all γ groups for each neutron group.

The total delayed and prompt γ productions are calculated by summing eq. [15] over all the NR reactions:

$$I_{i,\gamma}^{tot} = \sum_{r=1}^{NR} P_i^r \qquad [16]$$

For some nuclides delayed kerma and γ production are relevant, but sometimes they can be due to reactions with an extremely long half-life. Moreover the contribution of delayed kerma is often more evident in the low neutron energy range.

As it can be seen in the table, all the reactions of each nuclide were considered, i.e. no selection based upon the time was made. So this question arises: since the activation

products have half-lives dispersed over a very large range (from fractions of second to thousands of years), which ones are important to be included, and which ones are to be discarded? This question is not to be answered by the library generator, but it is only the final user that knows what he intends to do.

From our point of view we limit ourselves to suggest three ways:

1 to generate a unique decay library including the radioactive decay contribution of nuclides having half-life shorter than a cutoff value to be decided;

2. to generate several time-dependent libraries, the first one containing the prompt contribution, the other ones including the radioactive decay contribution at different cutoff values;

3. to generate a unique library with prompt contribution only, associated with an ancillary file containing the radioactive decay data and a simple handling program enabling the user to select those nuclei having half-life below an input cutoff value and to add these data to the main library.

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Some problems met during the processing

We assume 90% of shortcomings due to the data and 10% to the processing codes: some materials lead to abend during the execution. A certain number of nuclides, eventhough apparently correctly processed, shows strange or unphysical behaviour.

It must be noted that some nuclides have a particular structure with respect to ENDF-6 standard format. In fact the DDX data of MT = 16, 22, 28, 91, 106, 649, 849 for these nuclides (i.e. all reactions emitting particles in the continuum) are not given separately but they are lumped into MT = 10. Modifications had to be introduced in HEATR module in order this structure is accepted.

- Li-7 and Be-9: MF = 6 of these nuclides does not strictly follow the structure of ENDF-6 rules. In fact MF = 6 represents data in the form:

$$\sigma(E,E',\cos\theta) = \sigma(E) \times y(E) \times g_0(E,E') \times f(E,E',\cos\theta)$$

where:

$\sigma(E)$	MF = 3 cross section
$y(E)$	yield (multiplicity)
$g_0(E,E')$	energy spectrum
$f(E,E',\cos\theta)$	angular distribution

with the conditions:

$$\int f(E,E',\cos\theta)\, d\cos\theta = 1$$

and.

$$\int g_0(E, E') \int f(E,E',\cos\theta)\, d\cos\theta\, dE' = 1$$

Both materials present:

1. zero energy dependent yield for the (n, 2n) reaction: this occurs at the threshold E_0 of the reaction and at that point is meaningless, but at $E_0 + \varepsilon$ the number of emitted neutrons must be 2; some interpolated value between 0 and 2 is assumed instead, leading to an underestimate of the emitted neutrons.
2. angular distributions not normalized to 1 for some $f(E,E',\cos\theta)$ at the first E' value (where the integral is frequently zero) or at some non boundary E' point (where the integral ranges from 10^{-3} to 0.9). this produce underestimates in the secondary neutron production too and some floating point errors.

- B-10. missing angular distributions for charged particle emission levels.

- B-11: MF = 13, MT = 4 is given and produces high partial kerma due to the lumping of inelastic levels; the total kerma is higher of about 16% with respect to the upper kinematic limit.

- N-14: idem as for B-10.

- N-15: photon production data for MT = 102 (n,γ) are missing, leading to an overestimate of several orders of magnitude at low energies.

- O-16. idem as for B-10.

- F-19: MT 61 to 71 are dominantly (n,nα) but the LR flags are set to zero: this leads to discrepancies in the kerma for those MT because of falsely evaluated photon emission MT = 102 (n,γ) photon production data are given in MF = 13 and its partial kerma shows fluctuations. The overestimate in the total kerma is about 70%.

- Na-23 <; < the partial kerma of MT = 54, 57 and 58 are low in the range 4 to 5 MeV because of discrepancies in the photon energy yield of MF = 12, probably due to clerical errors. Overestimates at higher energies occur because of the low value of MF = 13, MT = 3 which subtract less photon energy than expected The error is in the range of 25%.

- Natural Mg: MF = 13, MT = 3 for this naturally occurring element is given from 0.260 to 20 MeV: in this range the kerma is largely fluctuating from negative to very high values.

- P-31: MF = 13, MT = 3 is given and the kerma is fluctuating from low to high values: the error is about 85%.

- Natural Sulphur: MF = 13, MT = 3 is given and the kerma is overestimated of about 35%.

- Ar-40: missing secondary energy distributions (MF = 5) for (n,nα) (MT = 22) and (n,np) (MT = 28). Stop in HEATR.

- Natural K: the photon production data from MT = 51, 52 and 53 do not cover the same energy range of the corresponding Sections in MF = 3 giving raise to overestimates of the total kerma around 3 MeV. Above 7 MeV the partial kerma of MT = 91 shows higher values too which are reported on the total kerma for about 50%.

- Natural Ca: negative kerma values are generated possibly due to inconsistency between MF = 3 cross section data and the lumped MT = 3 reaction adopted for the photon production in the range 400 keV to 1 MeV.

- Natural Ti: the same situation occurs as for natural Ca: the problem ranges from 150 keV to higher energies.

- Natural V: MF = 13, MT = 3 is given from 326 keV to 20 MeV with some discrepancies within 20%.

- Co-59: MF = 13, MT = 3 is given and the kerma presents some discrepancies within 20%.

- Zr: this material is a mixture of isotopes making up the natural element; the Resolved Resonance ranges of the various isotopes have not the same boundaries, unresolved parameters are given for not all isotopes and some ranges overlap. Negative scattering and (n, γ) cross section values are generated in RECONR; furthermore more than 740000 energy points per reaction are produced, so as the simple averaging process in GROUPR requires 2' about for each MT and the elastic scattering matrix requires 1h 28'. In order to have a check of the material, a run with larger precision criteria has been prepared but the results are very bad.

- Mo this material too is a mixture of isotopes with both Resolved and Unresolved Resonance Ranges, the UNRESR module produce negative (n, γ) cross section for low σ_0 values; those values stop the run during the averaging process in GROUPR.

- Nb-93 MF = 13, MT = 3 is given from 29 keV to 20 MeV with negative values in the MeV region, so as the kerma can not be accepted.

- Ba-132: Unresolved Resonance parameters are given for this isotope without the Resolved ones. This unusual situation does not stop the code but all cross sections below the lower boundary of the unresolved resonance energy range are zero.

- Ta-181 MF=13, MT=3 is given and in its range the kerma is largely fluctuating from negative to very high values.

- Natural W idem as for Ta-181.

- W-182, 183, 184, 186: MF=13, MT=4 is given for this nuclide and includes MT=16 (n,2n) and MT=17 (n,3n). The kerma shows negative values in the MeV region.

- Bi-209: the kerma has some small negative peaks in the MeV region probably due to discrepancies between the MF=3 and MF=13 data. The large overestimate above 5 MeV is possibly due to the inclusion of MT=22 (n,nα) and MT=28 (n,np) (for which no MF=5 data were given) into MT=4 without proportionally correcting the photon production data.

NJOY Processed Multigroup Library for Fast Reactor Applications and Point Data Library for MCNP - Experience and Validation -

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Abstract

JEF-1-based 50-group cross section library for fast reactor applications and point data library for continuous-energy Monte Carlo code MCNP have been generated using NJOY91.38 system. They have been examined by analyzing measured integral quantities such as criticality and central reaction rate ratios for 8 small fast critical assemblies.

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JEF-1-based 50-group cross section library for fast reactor applications and point data library for continuous-energy Monte Carlo code MCNP have been generated using NJOY91.38 system. They have been examined by analyzing measured integral quantities such as criticality and central reaction rate ratios for 8 small fast critical assemblies.

1. Introduction

At the beginning of 1990s, recent versions of evaluated nuclear data libraries, such as ENDF/B-VI, JENDL-3, JEF-1 or BROND-2, were released to all users in the world without restrictions.
JEF-1, the first version of the Joint Evaluated File, was developed at the OECD/NEA Data Bank in co-operation with several laboratories in the member countries. Reactor Physicists have assessed the accurate results of JEF-1 data for use in fast reactors by simulating critical experiments/1,2/.
For an application purpose to fast reactor physics calculations at Korea Atomic Energy Research Institute (KAERI), a JEF-1-based multigroup cross section set was processed and examined by analyzing a wide range of fast critical experiments/3/.

In recent years, advantages in computer technology have made it possible to use continuous energy point data in conjuction with Monte Carlo codes, such as MCNP, for the analysis of a wide variety of problems in the applied nuclear sciences. Moreover, it is of great importance to generate a new data library based on the most current evaluated data, such as ENDF/B-VI, JENDL-3.1 or JEF-2.2.

A work for producing the point data libraries compatible with Monte Carlo code MCNP/4/ is in progress at KAERI.
In order to validate the processed data, a preliminary point data library has been derived from JEF-1 data. And then the two cross section sets,

multigroup and point data, have been examined by analyzing a number of small fast critical experiments.
In this report a short description of the data processing of the multigroup and the point library, and of validated results are given. All data processing and MCNP calculations were performed on a Hewlett-Packard HP710 workstation with the UNIX operating system. And the multigroup transport calculations were performed on Cyber-960-31.

2. Data Processing

NJOY/5/ is well known as the most general purpose and versatile nuclear data processing code system, and the NJOY91 is the latest version released. In this processing the 91.38 version of the NJOY was used.

2.1 CCCC-format Library

A 50 neutron group cross section set for fast reactor applications has been processed from JEF-1 data. The 50 group structure uses quater lethary spacing in the middle energy range, with one-half lethargy groups above 500 keV and below 275 eV. Weighting spectrum has a fusion peak at high energies, followed by a fission spectrum, a slowing-down spectrum typical to fast reactor, and a thermal tail.
Figure 1 shows the processing scheme in detail. In the processing of RECONR and BROADR module, tolerances of 0.1 percent was used for the reconstructions, linearizations and thinnings. The CCCCR module formats the multigroup data for CCCC standard interface files, ISOTXS and BRKOXS. The ISOTXS consists of infinite-dilution cross sections and scattering matrices, and the BRKOXS contains Bondarenko-type self-shielding factors. A sample input for NJOY/CCCCR running is given in Table 1.

2.2 ACE-format Library

Continuous energy Monte Carlo codes do not access the evaluated data, such as ENDF/B-format data, directly : these data must first be processed into an appropriate form. The NJOY system reconstructs, broadens, and formats the ENDF-format data into ACE's format, the

appropriate form for MCNP code. Point cross section set used in the Monte Carlo calculations was derived from JEF-1 data. The set consists of continuous energy point cross sections, 32 equally-probable cosine bins for angular distribution and 32 equally-probable energy bins for energy distributions etc..

The tolerance of 0.5 percent was used in the RECONR and BROADR module for all nuclides. Because ACER module of NJOY91.38 cannot process JEF-1 U-233 data of fission neutron energy distributions (MF=5, MT=18), the data were taken from ENDF/B-VI.
Depending primarily on the number of resolved resonances for each nuclide, the resulting energy grid may contain as few as about 2,800 points of U-233 or as many as about 20,500 points of U-238. The energy grid sizes of actinides processed using tolerances of 0.5 percent are listed in Table 2. The detailed processing scheme is shown in Fig. 2. A sample input for NJOY/ACER running is given in Table 3.

3. Benchmark Problems

Eight small fast critical assemblies recommended by Cross Section Evaluation Working Group/6/ were selected for this work. The experiments include high enriched U-233, U-235 and Pu-239 fueled, bare and natural uranium or thorium-232 reflected assemblies. The characteristics on spherical model are summarized in Table 4.

4. Analysis

Measured integral quantities for selected 8 assemblies have been analyzed by using discrete ordinates method with the 50-group data library and Monte Carlo method with the continuous energy point data library, respectively. The same kinds of calculations for small fast critical experiments were performed by S. Pelloni/2/ using the discrete ordinates transport code, 174 neutron group VITAMIN-E structured problem related cross section libraries.
Data analyzed in this work are effective multiplication factors and fission reaction rate ratios at the core center.

4.1 SPHINX Calculations

The SPHINX code/7/ incorporates both one-dimensional diffusion and transport (discrete ordinates) theory. The one-dimensional diffusion theory neutronics of the 1DX code/8/ were incorporated into the SPHINX diffusion theory module. And the SPHINX transport module contains the one-dimensional transport theory neutronics of the ANISN code/9/. The calculations in this analysis were based on the transport theory and performed with P_3S_{16} approximation.

4.2 MCNP Calculations

MCNP is general-purpose, continuous energy, generalized-geometry, coupled neutron/photon Monte Carlo transport code. It was used in neutron tranport only in these calculations. The MCNP code computes k_{eff} by three different estimators, collision, absorption, and track length. To select the proper value of k_{eff}, the optimum k_{eff}-values were produced by the average of the absorption estimator with one of the other two. 30,000 particles and 50,000 particles were traced on bare and reflected cores, respectively.

5. Results and Discussion

In order to test the reliability of multigroup data processing, the results of SPHINX (discrete ordinates) calculations with 50-group cross section library have been intercompared with those of previously published values using a similar data and method, at first.
Table 5 shows the calculated effective multiplication factors. The central fission reaction rates in U-238, Pu-239, U-233 and Np-237 relative to the central fission reaction rates in U-235 are given in Table 6. In this intercomparison, KAERI's result is in good agreement with Pelloni's. And then, the results of multigroup discrete ordinates calculations and continuous energy Monte Carlo calculations have been intercompared for testing of point library processing in Tables 7 and 8. Considering the standard deviations in Monte Carlo calculations and disregarding some values in U-233 cores, the results of Monte Carlo calculations using the point data library processed in this work give generally reasonable results. The source of discrepancies in the calculation of U-233 cores is believed to be related to the above mentioned fission spectrum data

difference.
As a benchmarking, the ratios of calculated to experimantal (C/E) values are summarized in Table 9. Compared with SPHINX results, MCNP results on plutonium fueled cores show generally an improvement.

In addition, U-233 evaluated data have been checked by using ENDF/B-VI data and the result is listed in Table 10. The result shows that the U-233 related values using ENDF/B-VI data represent an improvement compared with those of JEF-1.

6. Conclusion

50-group cross section library for fast reactor applications and point data library for continuous energy Monte Carlo code MCNP have been generated from JEF-1 data by using NJOY91.38 processing system. For validating the usefulness of processed libraries, the results from calculations using multigroup data have been compared with previously published values, and the intercomparisons of multigroup discrete ordinates and continuous energy Monte Carlo calculations have been performed.

As a result, two kinds of data processing using CCCCR and ACER modules in NJOY91.38 show generally reasonable results.

Further work is still in progress to process the latest evaluated data, ENDF/B-VI(Rev. 2), JENDL-3.1 or JEF-2.2.

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Table 1. A Sample Input of NJOY/CCCCR for SPHINX Library

```
 0
 5
*moder*
 20 -21
*reconr*
-21 -22
*pendf tape for jef-1*/
 4925 1/
 0.001/
*92-U-235*/
 0/
*broadr*
-22 -23
 4925 1/
 0.001/
 300.
 0/
*unresr*
-21 -23 -22
 4925 1 7 1
 300.
 1.e10 1.e4 1.e3 1.e2 10. 1. 0.1
 0/
*groupr*
-21 -22 0 -25
 4925 5 0 8 3 1 7 1
*92-U-235*/
 300.
 1.e10 1.e4 1.e3 1.e2 10. 1. 0.1
 3/
 6/
 0/
 0/
*ccccr*
-25 21 22 0
 1 1 *njoy91.38*/
 50 0 1 3 1
* U235* * U235* *jef-1* * 4925* 4925 10.89
 50 0 50 -1
 2 2.35044+2 3.23620-11 1.76840-12 0.0 1.e10 0.0
 1 6
 300.
 1.e4 1.e3 1.e2 10. 1. 0.1
*stop*
```

Table 2. ACER-produced Energy Grid with 0.5% tolerances

Nuclide	Number of Energy Grid	
	JEF-1	BMCCS1
Th-232	15,880	4,132
U-233	2,796	708
U-234	11,360	58
U-235	8,424	4,418
U-236	15,485	155
U-238	20,483	5,631
Np-237	8,750	-
Pu-239	13,215	3,611
Pu-240	15,551	4,800
Pu-241	3,207	197
Pu-242	8,713	-

Table 3. A Sample Input of NJOY/ACER for MCNP Library

```
 0
 5
*moder*
 20 -21
*reconr*
-21 -22
*pendf tape for jef-1*/
 4925 1/
 0.005/
*reconstructed from 0 K*/
 0/
*broadr*
-22 -23
 4925 1 0 1 0/
 0.005/
 300.
 0/
*acer*
-21 -23 0 25 26/
 1/
*U-235 T=300 K from JEF-1*/
 4925 300./
 0.01 0/
 /
*stop*
```

Table 4. Small Critical Assembly Characteristics on Spherical Model

Assembly	Main Fissile Fuel	Fertile -to- Fissile Ratio	Approx. Core Volume (liter)	Core Radius (cm)	Reflector Thickness (cm)	Comments
FLATTOP-23	U^{2-53}	0.019	0.34	4.317	19.52	98.13% U^{233} metal, natural U reflector
FLATTOP-PU	Pu	0.050	0.39	4.533	19.597	Pu metal with 4.5% Pu^{240}, natural reflector
THOR	Pu	0.054	0.63	5.310	23.57	Pu metal with 5.1% Pu^{240}, Th^{232} reflector
JEZEBEL-23	U^{233}	0.019	0.90	5.983	-	Bare sphere with 98.13% U^{233} metal
FLATTOP-25	U^{235}	0.072	0.96	6.116	18.014	93.31% U^{235} metal, natural U reflector
JEZEBEL	Pu	0.047	1.09	6.385	-	Bare sphere with 4.5% Pu^{240} metal
JEZEBEL-PU	Pu	0.258	1.20	6.599	-	Bare sphere with 20.1% Pu^{240} metal
GODIVA	U^{235}	0.066	2.80	8.741	-	Bare sphere with 93.77% U^{235} metal

Table 5. Comparison of Calculated k_{eff} using Discrete Ordinates Method

Assembly	Pelloni*	KAERI	% difference
FLATTOP-23	0.9766	0.9780	-0.14
FLATTOP-PU	1.0054	1.0030	0.24
THOR	0.9923	0.9934	-0.11
JEZEBEL-23	0.9620	0.9687	-0.69
FLATTOP-25	0.9984	0.9982	0.02
JEZEBEL	1.0095	1.0105	-0.10
JEZEBEL-PU(20.1)	1.0024	1.0028	-0.04
GODIVA	0.9995	0.9987	0.08

* Data from Ref. 2.

Table 6. Comparison of Calculated Reaction Rate Ratios using Discrete Ordinates Method

Reaction Rate Ratio	Assembly	Pelloni*	KAERI	% difference
U-238f / U-235f	FLATTOP-23	0.1753	0.1734	1.10
	FLATTOP-PU	0.1772	0.1749	1.32
	THOR	0.1948	0.1925	1.19
	JEZEBEL-23	0.1959	0.1946	0.67
	FLATTOP-25	0.1582	0.1575	0.44
	JEZEBEL	0.2087	0.2069	-0.87
	JEZEBEL-PU(20.1)	0.2021	0.2011	0.50
	GODIVA	0.1745	0.1745	0.00
Np-237f / U-235f	FLATTOP-23	0.8209	0.8154	0.67
	FLATTOP-PU	0.8131	0.8064	0.83
	THOR	0.8864	0.8783	0.92
	JEZEBEL-23	0.8973	0.8927	0.52
	FLATTOP-25	0.7832	0.7814	0.23
	JEZEBEL	0.9268	0.9214	0.59
	JEZEBEL-PU(20.1)	0.9070	0.9030	0.44
	GODIVA	0.8510	0.8509	0.01
Pu-239f / U-235f	FLATTOP-25	1.3873	1.3861	0.09
	JEZEBEL	1.4304	1.4282	0.15
	GODIVA	1.4126	1.4124	0.01
U-233f / U-235f	FLATTOP-25	1.5161	1.5154	0.05
	JEZEBEL	1.4994	1.4994	0.00
	GODIVA	1.5129	1.5126	0.02

* Data from Ref. 2.

Table 7. Comparison of k_{eff} using SPHINX and MCNP

Assembly	Discrete Ordinates (SPHINX)	Monte Carlo (MCNP)	% difference
FLATTOP-23	0.9780	0.9730 (0.0010)*	0.51
FLATTOP-PU	1.0030	0.9980 (0.0010)	0.50
THOR	0.9934	0.9971 (0.0010)	-0.37
JEZEBEL-23	0.9687	0.9617 (0.0011)	0.73
FLATTOP-25	0.9982	0.9957 (0.0009)	0.25
JEZEBEL	1.0105	1.0106 (0.0012)	0.01
JEZEBEL-PU(20.1)	1.0028	1.0036 (0.0011)	-0.08
GODIVA	0.9987	0.9964 (0.0011)	0.23

* Fractional standard deviation

Table 8. Comparison of Calculated Reaction Rate Ratios using SPHINX and MCNP

Reaction Rate Ratio	Assembly	SPHINX	MCNP	% difference
U-238f / U-235f	FLATTOP-23	0.1734	0.1832 (0.0127)*	-5.35
	FLATTOP-PU	0.1749	0.1804 (0.0130)	-3.05
	THOR	0.1925	0.1974 (0.0128)	-2.48
	JEZEBEL-23	0.1946	0.2049 (0.0185)	-5.03
	FLATTOP-25	0.1575	0.1573 (0.0177)	0.13
	JEZEBEL	0.2069	0.2131 (0.0191)	-2.91
	JEZEBEL-PU(20.1)	0.2011	0.2000 (0.0199)	0.55
	GODIVA	0.1745	0.1791 (0.0259)	-2.57
Np-237f / U-235f	FLATTOP-23	0.8154	0.8396 (0.0106)	-2.88
	FLATTOP-PU	0.8064	0.8201 (0.0108)	-1.67
	THOR	0.8783	0.8926 (0.0108)	-1.60
	JEZEBEL-23	0.8927	0.9218 (0.0157)	-3.16
	FLATTOP-25	0.7814	0.7865 (0.0143)	0.65
	JEZEBEL	0.9214	0.9436 (0.0162)	-2.35
	JEZEBEL-PU(20.1)	0.9030	0.9103 (0.0169)	0.80
	GODIVA	0.8509	0.8688 (0.0215)	-2.06
Pu-239f / U-235f	FLATTOP-25	1.3861	1.3882 (0.0128)	0.15
	JEZEBEL	1.4282	1.4358 (0.0153)	-0.53
	GODIVA	1.4124	1.4198 (0.0197)	-0.52
U-233f / U-235f	FLATTOP-25	1.5154	1.5159 (0.0127)	-0.03
	JEZEBEL	1.4994	1.4974 (0.0152)	0.13
	GODIVA	1.5126	1.5113 (0.0197)	0.09

* Fractional standard deviation

Table 9. Comparison of C/E-values of Reaction Rate Ratios using SPHINX and MCNP

Reaction Rate Ratio	Assembly	SPHINX	MCNP
U-238f / U-235f	FLATTOP-23	0.9076	0.9589
	FLATTOP-PU	0.9714	1.0023
	THOR	0.9870	1.0125
	JEZEBEL-23	0.9133	0.9613
	FLATTOP-25	1.0573	1.0556
	JEZEBEL	0.9682	0.9971
	JEZEBEL-PU(20.1)	0.9762	0.9707
	GODIVA	1.0596	1.0872
Np-237f / U-235f	FLATTOP-23	0.9162	0.9434
	FLATTOP-PU	0.9600	0.9763
	THOR	0.9547	0.9702
	JEZEBEL-23	0.9138	0.9435
	FLATTOP-25	1.0281	1.0349
	JEZEBEL	0.9578	0.9809
	JEZEBEL-PU(20.1)	0.9815	0.9895
	GODIVA	1.0166	1.0380
Pu-239f / U-235f	FLATTOP-25	1.0117	1.0133
	JEZEBEL	0.9863	0.9916
	GODIVA	1.0074	1.0127
U-233f / U-235f	FLATTOP-25	0.9471	0.9474
	JEZEBEL	0.9502	0.9489
	GODIVA	0.9513	0.9505

Table 10. Testing of U-233 Data in ENDF/B-VI (C/E values)

Assembly	Keff	U-238f / U-235f	Np-237f / U-235f	U-233f / U-235f
FLATTOP-23	0.9936 (0.9766)*	1.0191 (0.9178)	0.9802 (0.9224)	-
JEZEBEL-23	0.9915 (0.9620)	1.0381 (0.9193)	0.9899 (0.9184)	-
FLATTOP-25	-	-	-	0.9803 (0.9476)
JEZEBEL	-	-	-	0.9848 (0.9502)
GODIVA	-	-	-	0.9855 (0.9515)

* Data from Ref. 2.

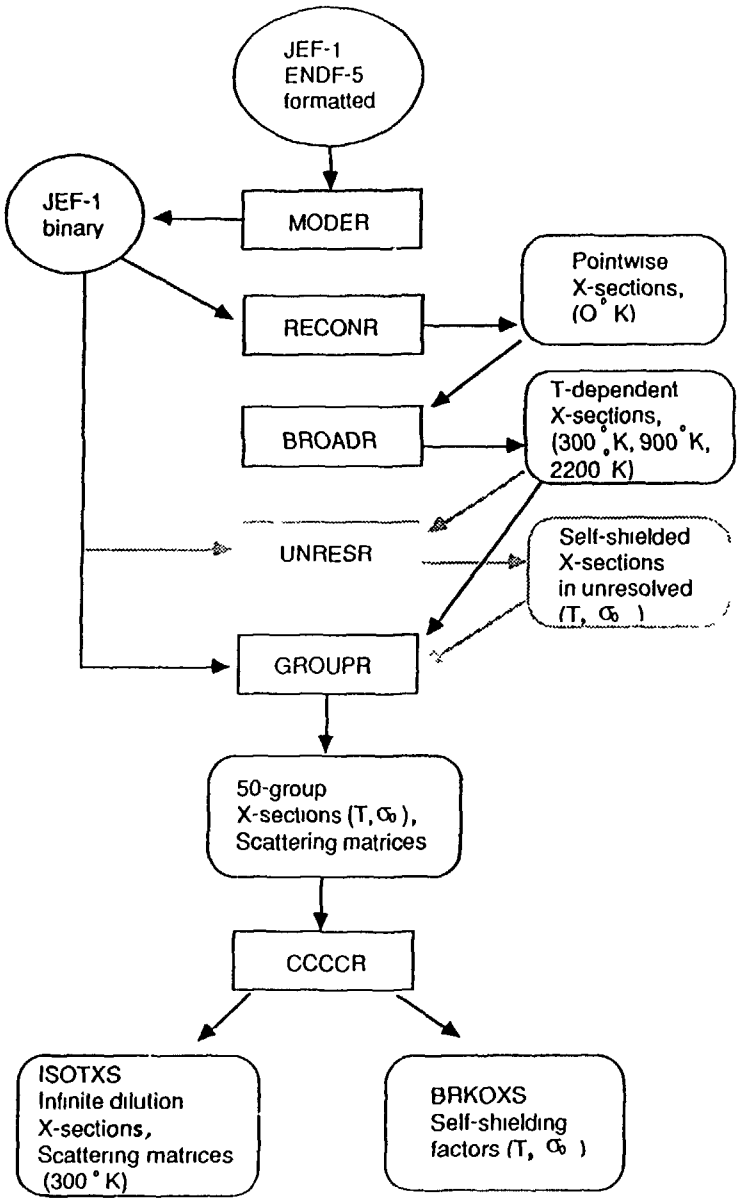

Figure 1. Flow Diagram for Generating CCCC-Format SPHINX Library

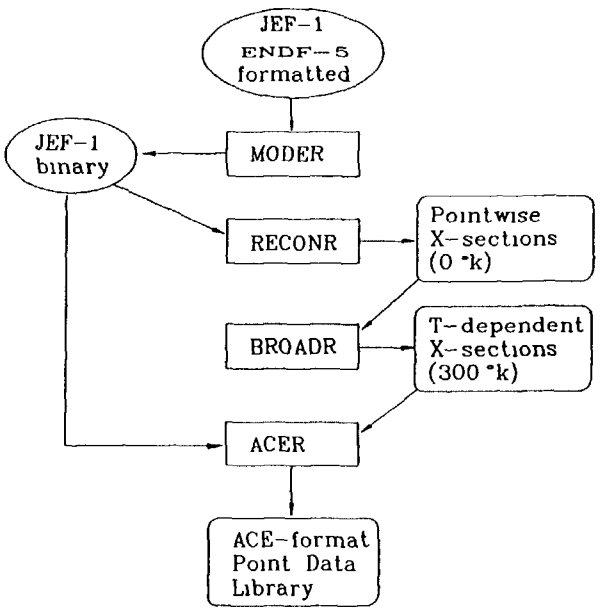

Fig 2 Flow Diagram for Generating ACE-format MCNP library

Processing of Nuclear Data for Reactor Applications

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Processing of Nuclear Data for Reactor Applications

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1 Introduction

Nuclear data centres played a very important role at the time when large volumes of experimentally measured data were produced and had to be compiled in sorted and computer-readable format, so that the evaluators could have a systematic and easy access to all new experimental data. The data centres bridged the gap between the experimentalists and the evaluators, so that new data could be incorporated into new evaluations relatively quickly.

Since then the bulk of the data base has been filled. The demand is now focused on more specific materials and reactions, with special emphasis on highly accurate absolute measurements for normalization purposes, or to resolve the discrepancies in the existing measured data. The work on theoretical nuclear models continues because recent developments promise significant improvements in the data prediction capabilities of such codes (particularly at high energies where measurements are difficult and costly), but a general decline in the support to the experimental facilities is observed worldwide.

A demand for new data is created when existing data are found to be deficient. However, user groups which can take advantage of the newly available evaluated data files are relatively few and highly specialized. They are often involved in new technologies such as nuclear fusion, where extensive experimental verification of the adequacy of the data can not be foreseen in the near future. On the other hand, many reactor physicists still use adjusted multigroup data libraries based on old evaluated data files.

Existing integral experiments which are adequately described in the literature can be used for data validation purposes, but the errors in the calculated parameters arising from the nuclear data, from data processing and from the physical modelling approximations must be decoupled as much as possible.

The problem of processing evaluated data files for specific applications has always been underestimated. Cullen has shown in the Code Verification Project [1], that data processing errors can be very significant. The work proved many of the data processing codes obsolete. Furthermore, new and more accurate measurements required extension of the evaluated data formats, such as ENDF-6 [2]. There are very few codes which can process the new data.

2 Validation of processed nuclear data

Processing of nuclear data for reactor applications is a complex task, which requires close cooperation between the specialists in nuclear data and in reactor physics to ensure correct definitions and consistent data processing. The ultimate test of the data is the comparison of the calculated results with the measurements on simple experimental configurations which can be modelled without significant approximations. Detailed description of such experiments can be found in the literature [3]. In the present discussion the TRX-1, TRX-2, BAPL-1, BAPL-2 and BAPL-2 thermal reactor benchmarks are considered. The following measured parameters are available for comparison:

- k_{eff} the effective multiplication factor,
- ρ^{28} the ratio of the epithermal to thermal capture reaction rates in ^{238}U,
- δ^{25} the ratio of the epithermal to thermal fission reaction rates in ^{235}U,
- δ^{28} the ratio of the total fission reaction rates in ^{238}U and ^{235}U,
- C^* the ratio of the capture reaction rates in ^{238}U and fission reaction rates in ^{235}U (i.e. the conversion ratio), measured in the TRX lattices only.

The thermal cutoff energy for the parameters defined above is $0.625\,eV$ where applicable.

In the literature the results of analysis for the above benchmarks can be found, with the aim to test the performance of multigroup libraries based on new evaluated data files. Some examples are given below.

JEF-1 Library analysis has been performed by Pelloni [4]. A number of different reactor physics code systems were used: BOXER, WIMS-D, TRAMIX/SOLVERB and MICROX-2. They are all based on deterministic methods. A summary of results is reproduced in Table 1. They are encouraging, but the spread in the results for the same benchmark is still quite significant, considering that the same evaluated data base is used. The error can be attributed to the differences in the physical models in the transport codes, but there is most certainly a contribution from the assumptions and processing errors in preparing the corresponding data libraries for individual codes. From the analysis no definite conclusions about the JEF-1 data can be made. At most one can say that in some cases the new library performs better than the older, adjusted data sets.

JENDL-3 Library analysis was performed by the JAERI LWR Integral Data Testing Working Group [5]. For the benchmarks of interest, several deterministic transport codes were used: WIMS-E, CASMO, SRAC and TGBLA. In addition the VIM Monte-Carlo calculation was included. A summary of results is reproduced in Table 2. Considerable systematic differences from the measurements can be observed. The origin of some of the differences was identified in the paper, but was

Table 1: Summary of results for JEF-1 library using different processing codes for selected benchmark lattices and comparison with measured values.

LATTICE	k_{eff}	ρ^{28}	δ^{25}	δ^{28}	C^*
TRX-1	1.00000(.30)	1.320(1.6)	0.0987(1.0)	0.0946(4.3)	0.797(1.0)
BOXER	0.99830(-.17)	1.350(+2.3)	0.1001(+1.4)	0.0994(+5.1)	0.798(+.13)
WIMS-D	0.99500(-.50)	1.362(+3.2)	0.0974(-1.3)	0.0995(+5.2)	0.802(+.62)
TRAMIX	0.99350(-.65)	1.380(+4.5)	0.0999(+1.2)	0.0959(+1.3)	0.808(+1.3)
MICROX-2	0 99200(-.80)	1 396(+5.7)	0.1016(-2.9)	0.0987(+4.3)	0.811(+1.8)
TRX-2	1.00000(.10)	0.837(1.9)	0.0614(1.3)	0.0693(5.1)	0.647(.93)
BOXER	0 99890(-.01)	0.840(+.35)	0.0613(- 16)	0.0703(+1.4)	0.643(-.62)
WIMS-D	0.99510(-.49)	0.853(+1.9)	0.0597(-2.7)	0.0705(+1.7)	0.650(+.46)
TRAMIX	0.99560(-.44)	0.863(+3.1)	0.0612(-.33)	0 0680(-1.9)	0.650(+.46)
MICROX-2	0 99460(-.54)	0 874(+4.4)	0 0621(-.33)	0 0698(+.72)	0.653(+.93)
BAPL-1	1.00000(.10)	1 390(.72)	0.0840(2 4)	0 0780(5.1)	0.000
BOXER	1.00470(+.47)	1.399(+ 65)	0 0844(+.48)	0 0776(-.51)	0.809
WIMS-D	0 99890(-.11)	1.428(+2 7)	0 0820(-2.4)	0 0773(-.90)	0.819
TRAMIX	0 99910(-.09)	1.438(+3 5)	0 0841(- 12)	0.0746(-4.4)	0.821
MICROX-2	0.99420(-.58)	1.463(+5 3)	0 0857(+2 0)	0.0770(-1.3)	0.828
BAPL-2	1.00000(.10)	1 120(.89)	0 0680(1 5)	0.0700(5.7)	0.000
BOXER	1 00490(+.49)	1.160(+3.5)	0.0667(-1.9)	0.0666(-4.9)	0.736
WIMS-D	0 99910(-.09)	1.189(+6 2)	0.0669(-1.6)	0.0665(-5.0)	0.745
TRAMIX	1.00000(+.00)	1.195(+6 7)	0.0685(+ 73)	0.0641(-8 4)	0.747
MICROX-2	0.99530(+.47)	1.215(+8.5)	0 0698(+2.6)	0.0662(-5.4)	0 752
BAPL-3	1.00000(.10)	0 906(1.1)	0.0520(1.9)	0.0570(5.3)	0.000
BOXER	1.00520(+.52)	0 907(+ 11)	0.0527(+1.4)	0 0545(-4 3)	0 657
WIMS-D	0.99970(-.03)	0 931(+2.8)	0.0514(-1.2)	0.0545(-4.4)	0 665
TRAMIX	1 00080(+.34)	0 937(+3 4)	0 0526(+1.2)	0.0525(-7.9)	0.666
MICROX-2	0.99740(+.08)	0 952(+5.1)	0 0535(+2.9)	0.0541(-5.1)	0 671

NOTE Measured values and their % uncertainty (in brackets) for each lattice are given in row-1. Then follow the results using different processing codes, and in brackets the % difference from measurements.

not pursued any further. It is perhaps surprising that even the Monte-Carlo results do not agree too well with the measurements.

WIMS Library Update Project (WLUP) results: An analysis has been performed for the selected benchmarks, based on ENDF/B-VI, JEF-1, JEF-2, JENDL-3.1 and CENDL-2 evaluated data files [6,7]. The results are reproduced in Tables 3 and 4. No significant differences are observed when different evaluated data files are used. Furthermore, the calculated results agree quite well with the measurements. In fact, most of the calculated parameters lie within the experimental uncertainty of the measurements. This proves that good results can be obtained in modelling some integral experiments even with the use of relatively simple and inexpensive calculational tools and that most of the discrepancies between calculations and measurements originate most likely from the assumptions in the preparation of the multigroup libraries for such codes, rather the the mathematical models.

Table 2: Summary of results for JENDL-3 library using differen ing codes for selected benchmark lattices and comparison measured values.

LATTICE	k_{eff}	ρ^{28}	δ^{25}	δ^{28}	C^*
TRX-1	1.00000(.30)	1.320(1.6)	0 0987(1.0)	0.0946(4.3)	0.797(1.0)
WIMS-E	1.00190(+.19)	1.324(+.28)	0.0962(-2.6)	0.0986(+4.2)	0.787(-1.3)
CASMO	0.99940(-.06)	1.348(+2.1)	0.0962(-2.6)	0.0990(+4.7)	0.794(-.38)
SRAC	0.99560(-.44)	1.345(+1.9)	0.0966(-2.1)	0.1008(+6.6)	0.794(-.38)
TGBLA	0.99360(-.64)	1.354(+2.6)	0.0962(-2.6)	0.0986(+4.2)	0.797(+0.0)
VIM	0.99240(-.76)	1.347(+2.1)	0.0972(-1 6)	0.1014(+7.2)	0.795(-.19)
TRX-2	1.00000(.10)	0.837(1.9)	0.0614(1.3)	0.0693(5.1)	0.647(.93)
WIMS-E	1.00210(+.21)	0.830(-.86)	0.0591(-3.7)	0.0703(+1.4)	0.636(-1.7)
CASMO	0 99880(-.12)	0.843(+.76)	0.0592(-3.6)	0 0708(+2.1)	0.641(-1.0)
SRAC	0.99790(- 21)	0.839(+.19)	0.0593(-3.4)	0.0719(+3.8)	0 639(-1 2)
TGBLA	0.99650(- 35)	0 842(+ 57)	0.0586(-4 6)	0.0705(+1.7)	0.641(- 95)
VIM	0.99170(-.83)	0.837(+0 0)	0.0602(-1 9)	0 0728(+5 0)	0.639(-1.2)

NOTE: Measured values and their % uncertainty (in brackets) for each lattice are given in row-1. Then follow the results using different processing codes, and in brackets the % difference from measurements (reconstructed from a diagram).

3 Validation of WLUP Results

Data processing methods have been verified using ENDF/B-IV library. This particular choice of the basic evaluated data file has the advantage, that it is relatively cheap to process and that most of the data processing codes (including some older ones) can accept its format. However, the most important advantage is a very detailed analysis of the selected benchmark experiments, which is available from the literature [8]. Several laboratories from United States participated in the analysis, many of them using Monte-Carlo codes. These results based on ENDF/B-IV data serve as a reference numerical benchmark to validate the data processing methods for WIMS.

An ENDF/B-IV based WIMS multigroup library was prepared with the `NJOY91.38` code, which included some modifications in the `WIMSR` module). With this library the benchmarks were analyzed and the results were compared to the reference numerical benchmark values. The results which are presented in Table 5 indicate that there is very good agreement between WIMS results and the reference (mainly Monte-Carlo) results. This is considered an indication of correct data processing.

The same data processing methods were then applied for generating WIMS libraries using ENDF/B-VI, JEF-1, JEF-2, JENDL-3.1 and CENDL-2 evaluated data files. The results are directly comparable and reflect differences in the data alone, since the data processing methods and even the input instructions for the the transport code are the same.

4 Conclusions

The nuclear data field should not end with the release of an evaluated data file. It should include data verification, which extends beyond the formal checking of the

Table 3: Summary of WIMS results using different evaluated data libraries for selected benchmark lattices and comparison with measured values.

LATTICE	k_{eff}	ρ^{28}	δ^{25}	δ^{28}	C^*
TRX-1	1.00000(.30)	1.320(1.6)	0.0987(1.0)	0.0946(4.3)	0.797(1.0)
CENDL-2	0.99997(-.00)	1.348(+2.1)	0.0980(-.72)	0.0960(+1.4)	0.789(-1.1)
ENDF/B-VI	0.99472(-.53)	1.345(+1.9)	0.0985(-.18)	0.0969(+2 5)	0.794(-.36)
JEF-1	0.99960(-.04)	1.336(+1.2)	0.0988(+.12)	0.0978(+3.4)	0.793(-.53)
JEF-2	0.99682(- 32)	1.339(+1.4)	0.0985(-.23)	0.0958(+1.3)	0.798(+.09)
JENDL-3 1	0.99457(-.55)	1.344(+1.8)	0.0970(-1.8)	0.0978(+3.3)	0.793(-.45)
WIMS-D/4	1 00227(+.23)	1.279(-3.1)	0.0990(+.30)	0.0965(+2.0)	0.780(-2.1)
TRX-2	1.00000(.10)	0 837(1.9)	0.0614(1.3)	0.0693(5.1)	0.647(.93)
CENDL-2	0.99911(-.09)	0 850(+1.6)	0.0603(-1.8)	0.0682(-1.6)	0.638(-1.4)
ENDF/B-VI	0.99429(- 58)	0 847(+1.2)	0.0606(-1.4)	0 0689(-.65)	0 643(- 68)
JEF-1	0.99844(-.16)	0.842(+.60)	0.0608(-.99)	0.0699(+.81)	0.642(-.74)
JEF-2	0.99581(-.42)	0.844(+ 84)	0.0606(-1.3)	0.0684(-1.3)	0.646(- 12)
JENDL-3.1	0.99452(-.55)	0.847(+1.2)	0.0596(-2.9)	0.0697(+ 55)	0.642(- 77)
WIMS-D/4	0.99654(-.35)	0.808(-3.5)	0.0610(-.64)	0.0695(+ 30)	0.636(-1 7)
BAPL-1	1.00000(.10)	1.390(.72)	0.0840(2.4)	0.0780(5.1)	0.000
CENDL-2	1.00549(+.54)	1 399(+.63)	0.0825(-1 8)	0 0740(-5 2)	0 799
ENDF/B-VI	1.00052(+ 05)	1.392(+.14)	0.0828(-1.4)	0.0747(-4.3)	0.804
JEF-1	1.00567(+.56)	1.385(-.36)	0.0832(-.93)	0.0758(-2 8)	0.803
JEF-2	1.00286(+.28)	1.385(- 36)	0.0828(-1.4)	0.0741(-5 1)	0.808
JENDL-3 1	1.00137(+ 14)	1.392(+ 14)	0.0816(-2.9)	0.0757(-3.0)	0.803
WIMS-D/4	1.00292(+ 29)	1 358(-2.3)	0.0840(+.05)	0.0755(-3.2)	0.800
BAPL-2	1.00000(.10)	1.120(89)	0.0680(1.5)	0.0700(5 7)	0.000
CENDL-2	1.00424(+.42)	1.167(+4 2)	0.0674(-.96)	0.0636(-9.1)	0.730
ENDF/B-VI	0.99936(-.06)	1.161(+3.7)	0.0676(-.56)	0.0642(-8.3)	0.734
JEF-1	1.00435(+.43)	1.156(+3.2)	0.0679(-.10)	0.0653(-6 7)	0.734
JEF-2	1.00157(+.16)	1.155(+3 1)	0.0676(-.54)	0.0638(-8.8)	0.737
JENDL-3.1	1.00039(+.04)	1.161(+3.7)	0.0666(-2.1)	0.0652(-6.9)	0.733
WIMS-D/4	1.00049(+.05)	1.133(+1.2)	0.0687(+1.0)	0.0652(-6.8)	0.732
BAPL-3	1.00000(.10)	0.906(1.1)	0.0520(1.9)	0.0570(5.3)	0.000
CENDL-2	1.00347(+.34)	0.919(+1.5)	0.0518(-.35)	0.0521(-8.5)	0.653
ENDF/B-VI	0.99885(-.12)	0.914(+.88)	0.0520(+.04)	0.0527(-7.6)	0.658
JEF-1	1.00341(+ 34)	0.911(+.55)	0.0523(+.48)	0.0536(-5.9)	0 657
JEF-2	1.00073(+.07)	0.910(+ 44)	0.0520(+.08)	0.0524(-8 0)	0.661
JENDL-3 1	1.00000(+.00)	0 915(+.99)	0.0512(-1.5)	0.0535(-6.2)	0.657
WIMS-D/4	0.99807(- 19)	0.894(-1.3)	0.0529(+1.7)	0.0538(-5.6)	0.657

NOTE: Measured values and their % uncertainty (in brackets) for each lattice are given in row-1. Then follow the results using libraries based on different evaluated data files and the old WIMS library, and in brackets the % difference from measurements.

Table 4: Summary of average measurement uncertainties and average differences between calculated and measured values over all the benchmark lattices.

LATTICE	k_{eff}	ρ^{28}	δ^{25}	δ^{28}	C^*
Average	0.16	1.32	1.69	5.12	0.97
CENDL-2	0.24(0.25)	1.46(0.53)	-1.13(0.59)	-4.60(4.04)	-1 22(0.15)
ENDF/B-VI	-0.25(0.26)	1.03(0.63)	-0.69(0.59)	-3.67(4.09)	-0.52(0.16)
JEF-1	0.23(0.28)	0.50(0.56)	-0.28(0.58)	-2.24(3.88)	-0.63(0.11)
JEF-2	-0.05(0.28)	0 58(0.64)	-0 68(0.59)	-4.39(3.88)	-0.02(0.11)
JENDL-3 1	-0.19(0.29)	1.03(0.59)	-2 23(0 57)	-2.43(3.91)	-0.61(0.16)
WIMS-D/4	0.00(0.24)	-2.55(0.84)	0 49(0.81)	-2.66(3 36)	-1.90(0.23)

NOTE Average uncertainty over all lattices for each of the parameters is given. Then follow the mean differences between the calculated values and the measurements and in brackets the standard deviation for each of the evaluated libraries considered.

Table 5: Summary of WIMS results based on ENDF/B-IV data for selected benchmark lattices and comparison with reference results.

LATTICE	k_{eff}	ρ^{28}	δ^{25}	δ^{28}	C^*
TRX-1	0.98760(.32)	1.382(.43)	0.0994(50)	0.0955(.63)	0.806(.25)
ENDF/B-IV	0.98706(-.05)	1.380(-.14)	0 1002(+ 80)	0.0928(-2.8)	0.804(-.25)
TRX-2	0.99350(.31)	0.863(.58)	0.0609(49)	0.0676(.44)	0 647(.31)
ENDF/B-IV	0.98916(-.44)	0.868(+.58)	0.0615(+1.0)	0.0660(-2.4)	0.649(+.25)
BAPL-1	0.99140(.30)	1.433(2.0)	0.0835(1.6)	0.0735(.95)	0.817(1.3)
ENDF/B-IV	0.99321(+.18)	1.429(-.28)	0.0843(+.98)	0.0717(-2.5)	0.814(-.31)
BAPL-2	0.99320(.09)	1.188(1.3)	0.0678(1.8)	0.0631(.79)	0 742(.81)
ENDF/B-IV	0.99325(+.01)	1.191(+ 25)	0.0688(+1.4)	0.0617(-2.3)	0 743(+.08)
BAPL-3	0.99395(.21)	0.936(1.7)	0.0522(.38)	0.0516(.78)	0.664(1 1)
ENDF/B-IV	0.99373(-.02)	0 938(+.21)	0.0529(+1.3)	0.0506(-2.0)	0 664(0 00)

NOTE: For each lattice the reference solution and the % uncertainty is given in the first row. The results using the updated ENDF/B-IV based library and the % difference from reference are given in the second row.

formats and other trivial errors. The goal should be to produce at least partial processed data libraries for a few specific and commonly used codes so that a selection of benchmark test cases can be executed. This should bring the evaluated data files much closer to the end user.

Several experimental measurements on reactors are reported in the literature, such as those discussed in this work, which can be modelled quite easily even with lattice codes. These can serve as benchmarks for an inexpensive validation of data. The validity of such calculations can be demonstrated once and for all by comparison to a detailed Monte-Carlo calculation. An extended data base of suitable benchmarks could be established, with standardized inputs for different transport codes.

In the past, the nuclear data centres very effectively bridged the gap between the experimentalists and the evaluators by setting up a computerized data base. With the bulk of the job finished and the demand for new data being given lower priority, the nuclear data centres could play a new important role in bridging the gap between the evaluated data and the reactor physicists.

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