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PLANS FOR USE OF ENDF/B IN REACTOR RESEARCH IN INDONESIA

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Abstract

Nuclear data are numerical constants of nature which quantify the nuclear behaviour of all elements and isotopes which make up the reactor medium and its environment, and which are needed as input for performing design calculations for safe and reliable operation of nuclear reactors. The nuclear data are available in the form of recommended values in specially formatted computerized files such as the Evaluated Nuclear Data File-B, known as ENDF/B. The development of base technology in the scheme of original reactor design calculations involves the mastering of the art of ENDF/B data processing. This paper briefly discusses the current status of this activity in Jakarta and gives an account of the future plans, with emphasis on the role of ENDF/B in reactor calculations.

I. Introduction

In order to perform rigorously all neutronic calculations starting from the basic evaluated data file such as ENDF/B, for any research or original designs, a long term project has been initiated in BATAN, Indonesia. The project involves starting from basic data file, mastering the art of ENDF/B data processing for reactor applications and creating the capability to use ENDF/B in original reactor design calculations. This paper

briefly discusses the current status of this activity and gives an account of the future plans.

Basic evaluated nuclear data libraries such as ENDF/B, JENDL-2, BROAND are available in ENDF/B format from international nuclear data centres such as IAEA, Vienna. The use of such evaluated nuclear data libraries in research reactor application calculations has been realized to be natural starting point for creating the base technology and for enhancing the local capability for indigenous design of reactors including research and power reactors. Although condensed multi-group libraries and few group constants are available from international distribution centres such as IAEA, NEA Data Bank, RSIC(USA) etc., it has not been possible to readily adapt these multigroup data unless the compatible neutronic code systems are also imported. Most important drawback is that the use of such condensed libraries continue dependence on foreign efforts in updating the libraries and the possibility of rigorously performing sensitivity calculations in a detailed manner starting from the basic evaluated data file is ruled out in such cases. Therefore plans to use the ENDF/B formatted basic neutron interaction cross section libraries for reactor design in Indonesia have been formulated and presented in this paper. The advantages for the Indonesian Nuclear Energy Programme in the direct use of ENDF/B are outlined. The present paper also outlines the progress made thus far in the installation of ENDF/B system in BATAN and presents details of proposed activities to further consolidate the routine use of ENDF/B in research reactor design calculations in BATAN. The experience gained thus far, in BATAN in the use of IAEA pre-processing codes LINEAR, RECENT, FIXUP, SIGMA1, GROUPIE etc., has been very useful in understanding

several aspects of ENDF/B processing for reactor applications and in appreciating the further efforts needed for the successful use of ENDF/B system in situations existing in Indonesia. Appreciation of physical principles and generation of group constants using the IAEA pre-processing codes and codes such as FEDGROUP-C, REX1, REX2 and REX3 or use of NJOY code sytem developed in USA need a long term committed team to realize the objectives. As part of nuclear data processing activities it is planned to take up intercomparison of nuclear data for important isotopes by processing ENDF/B formatted basic evaluated data libraries such as ENDF/B, JENDL-2, BROAND, ENDL etc., which are available from IAEA. These projects can make use of both IBM-PC/AT personal computers and the VAX 8550 computer with adequate memory. As part of strategy for long term planning and understanding of the basic evaluation process used in the creation of ENDF/B itself, work has been initiated in BATAN to understand and perform nuclear model based neutron interaction cross section calculations using well established user-friendly computer codes. As an example to cite here, the statistical optical model program ABAREX which is a well known user-friendly code has already been adapted on IBM-PC/AT in BATAN.

II. Role of Nuclear Data in Reactor Design

In reactor design, given a set of boundary conditions, one aims at solving the neutron balance equation for neutron density as function of position, direction of motion and energy of the neutron and time. There are seven independent variables in the integro-differential equation that one has to solve. Table 1 gives the transport equation (Bell and Glasstone, 1970) which describes the neutron population density in the form of an

Table 1 : The seven dimensional balance equation for the neutron in a nuclear reactor

The basic neutron balance equation for nuclear reactors in a volume element dV for neutron having energy between E and E+dE and flight direction in solid angle $d\vec{\Omega}$ around $\vec{\Omega}$ is given by:

$$\frac{\partial N(\vec{r}, E, \vec{\Omega}, t)}{\partial t} = -\vec{\Omega}\cdot\nabla N(\vec{r}, E, \vec{\Omega}, t)v - \Sigma_t N(\vec{r}, E, \vec{\Omega}, t)$$

Leakage — Loss due to absorption and scattering

$$+ \int\int \Sigma_s(E' \to E, \vec{\Omega}' \to \vec{\Omega}) N(r, E', \Omega', t) v' \, dE' \, d\vec{\Omega}'$$

source due to scattering from other directions and energies

$$+ \frac{\chi(E)}{4\pi} \iint v'\Sigma_f(E') N(\vec{r}, E', \vec{\Omega}, t) \bar{\nu}(E') \, dE' \, d\vec{\Omega}'$$

Fission source

$+ S(\vec{r}, E, \vec{\Omega}, t)$

External source

Symbols

v	: Neutron speed corresponding to energy E
N	: Neutron angular density
Σ_t	: Total neutron cross section
S	: External neutron source
$\vec{r}$	: space coordinate
$\vec{\Omega}$	: Unit vector in the direction of neutron
E	: Energy of the neutron
t	: time
$\Sigma_s(E' \to E, \vec{\Omega}' \to \vec{\Omega})$	: Scattering cross section for transfer of neutron from $(E', \vec{\Omega}')$ into $(E, \vec{\Omega})$
$\chi(E)$	: Fission neutron spectrum
$\bar{\nu}$	: Average number of neutrons produced per Fission

Note that the macroscopic cross sections depend on space $\vec{r}$, time t and energy E

Boundary conditions :

: $N(\vec{r}, E, \vec{\Omega}, t) = 0$ For energies greater than maximum of source energy (15 MeV)

: Continuity at the interface between two media

: $N(\vec{r}, E, \vec{\Omega}, t) = 0$ At the outer free surface for directions entering the system

integro-differential equation. Physicists try to solve this equation which is the starting point for the physics design of the reactor, by taking all the nuclear properties of the materials and the geometry of the reactor into account in order to obtain the important characteristics of the nuclear reactor such as the K-eff, namely, the effective multiplication factor, critical mass, fuel enrichment, neutron flux, Doppler and coolant void reactivity coefficients, dynamic characteristics during normal and abnormal operating conditions etc. The physics design of the reactor forms part of the overall engineering design of the reactor which includes the thermal hydraulics design of the system. The macroscopic cross sections needed as input to solve the transport equation in Table 1 are obtained from the specified atom densities of individual isotopes and the recommended numerical values of microscopic neutron-nuclear interaction cross section data of individual isotopes which are available in the evaluated nuclear data files. The limitation of the existing computers force reactor physicists to invoke the multigroup approximation to solve the balance equation. For solving the multigroup form of the balance equation, the multigroup cross sections are to be generated. The energy dependent cross sections of individual isotopes which are available in the evaluated nuclear data files are condensed into a few group constants through averaging procedures by following the law of conservation of reaction rates for each of the energy bins. The process of condensation is generally performed in a few steps. The basic file is pre-processed into 2000 fine groups followed by neutronic calculations in 2000 groups to get the spectrum characterising the homogeneous composition of each reagion of the reactor. The cross sections are then collapsed to 200 groups at

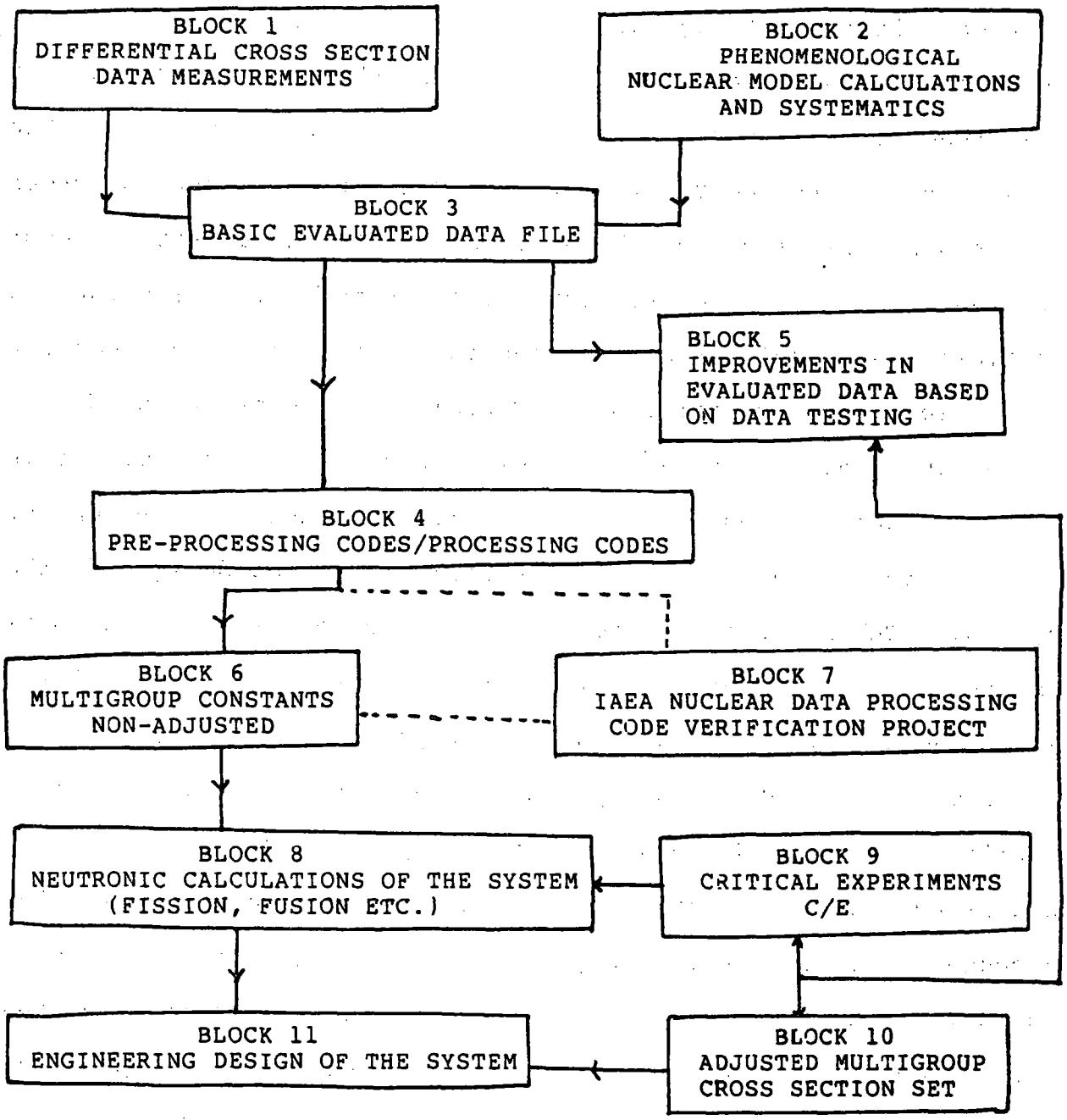

Fig. 1: Various stages/steps in the nuclear reactor calculations

which level the heterogeneity of the lattice may be treated. With further neutronic calculations, condensation is performed to generate a few froup constants for the reactor under consideration. Fig. 1 shows, in a general way, the various stages in the nuclear reactor calculations.

One of the well known codes developed in U. K. is the WIMS code which can be used for a wide spectrum of reactor

systems. Gaining sufficient experience with such code systems gives the necessary expertise that will enable Serpong to prepare and generate data from ENDF/B for light water and research reactor calculations. The use of FEDGROUP-C code which processes ENDF/B and produces results of multigroup cross sections in WIMS format or the use of the more general NJOY code system of U. S. A. which also has the option to produce results in WIMS format could be taken up in Serpong. The understanding of the use of such code systems demand a thorough knowledge of whole computational reactor physics when one takes up the task of practical thermal power and research reactor calculations.

III. Implementation

With the available facility like VAX-8550, Packard X-Y plotter, Microvax Workstation II/GPX, laser printer and desk-top PCs, the implementation of data base development project will be carried out in steps. At the request of BATAN, IAEA undertook a seven weeks mission (Dr. S. Ganesan) from 19 November 1988 to 6 January 1989 following a one week preparatory mission which was carried out from 24th September to 2nd October 1988 by Mr. Andrej Trkov.

The following capabilty as a result of this mission exists presently in Jakarta.

- The team in Jakarta can retrieve and understand the contents of ENDF/B formattted data. The team knows how to use the manuals.
- The team in Jakarta has acquired the confidence and are capable of correctly using the following pre-processing codes.

LINEAR: To linearize ENDF/B data

RECENT: To reconstruct resonance cross sections

SIGMA1: To obtain Doppler broadened cross sections at higher temperatures.

MERGER: To obtain ENDF/B tape for specific isotopes and reactions.

MIXER: To obtain total cross sections of mixture

VIRGIN: To study and interpret self-transmission experiments

FIXUP: To correct for negative cross sections.

LEGEND: To correct for negative angular distributions.

GROUPIE:To obtain multigroup cross sections and self-shielding factors which are temperature and composition dependent for total, elastic, fission, scattering and capture, based on Bondarenko definition; to obtain multi-band parameters.

It is planned that the programs EVALPLOT and COMPLOT will be commissioned as the hardware i. e. the X-Y plotter will be connected to mainframe computer. As an exercise to thoroughly master the retrieval of ENDF/B data and to satisfy the immediate plotting requirements, a team consisting of Mr. Bunjamin, Mr. Dewanto and Mr. Karsono and led by S. Ganesan successfully developed a code for plotting any desired cross section from file 3 of ENDF/B. The listing of this program is available from the authors. The successful development of the plotter program during the mission period also shows that the Serpong team has been able to write its own retrieval program to retrieve cross sections from File 3 of ENDF/B formatted files and make use of the available facilities i.e the MicroVax Work station and the laser printer to obtain the required plots. As illustration seven plots of ENDF/B data obtained recently in Serpong are shown in Figs. 2 and 8.

NOTE: ALL THE FOLLOWING GRAPHS WERE OBTAINED USING THE LOCALLY DEVELOPED PLOTTER PROGRAM:

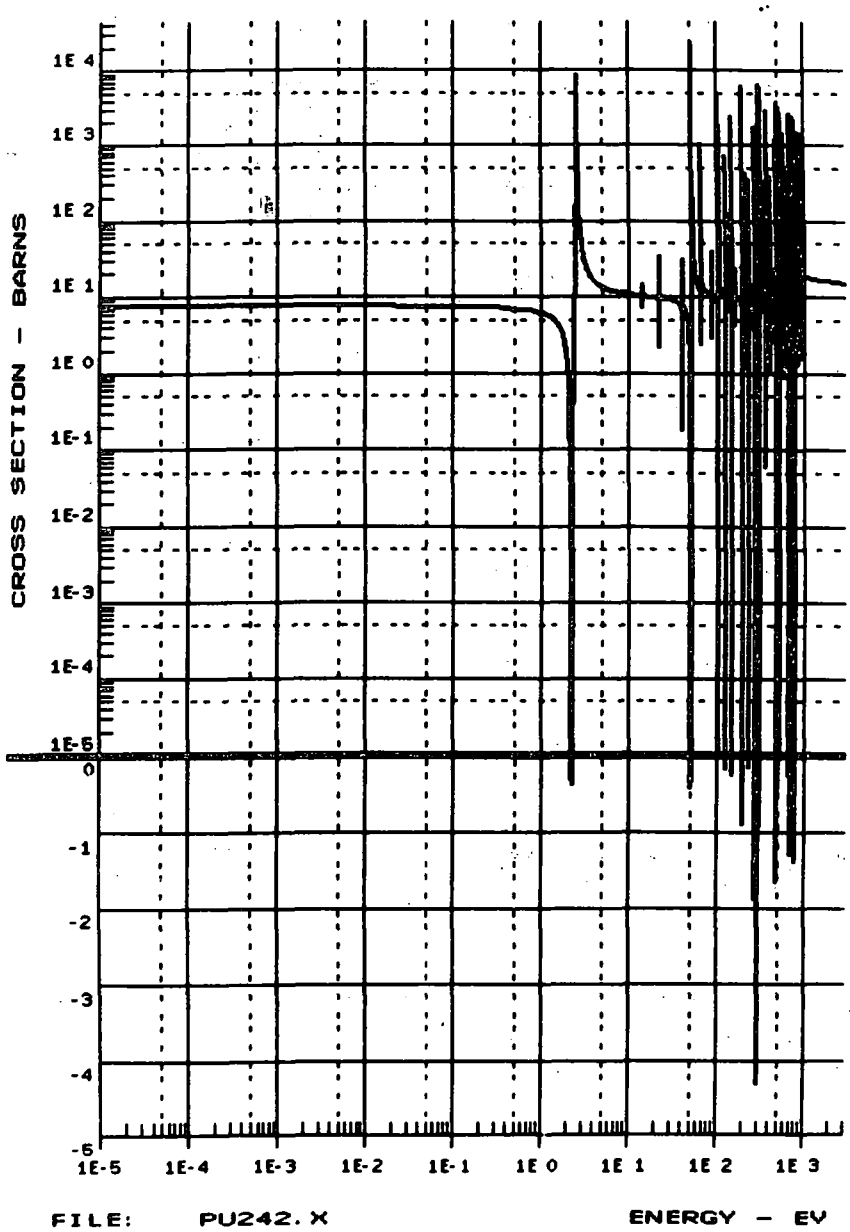

Fig.2: A plot of scattering cross sections retrieved from the output of RECENT for Pu-242 of ENDF/B-V Actinides (Rev. 2) in 1.0E-5 eV to 1E+3 eV energy region. The appearance of negative cross sections have been demonstrated.

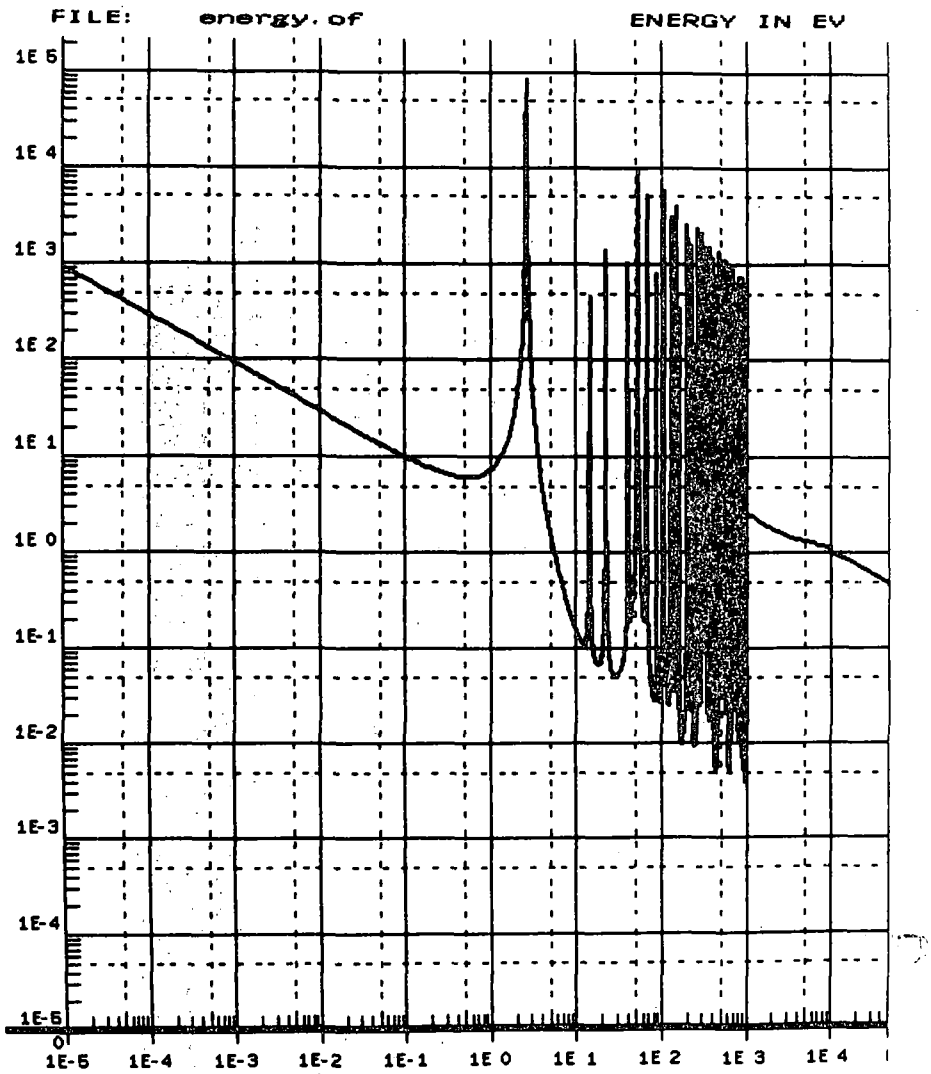

Fig.3: The capture cross sections at zero kelvin retrieved from the output of RECENT for Pu-242 of ENDF/B-V Actinides (Rev. 2) in 1.0E-5 eV to 1E+4 eV energy region.

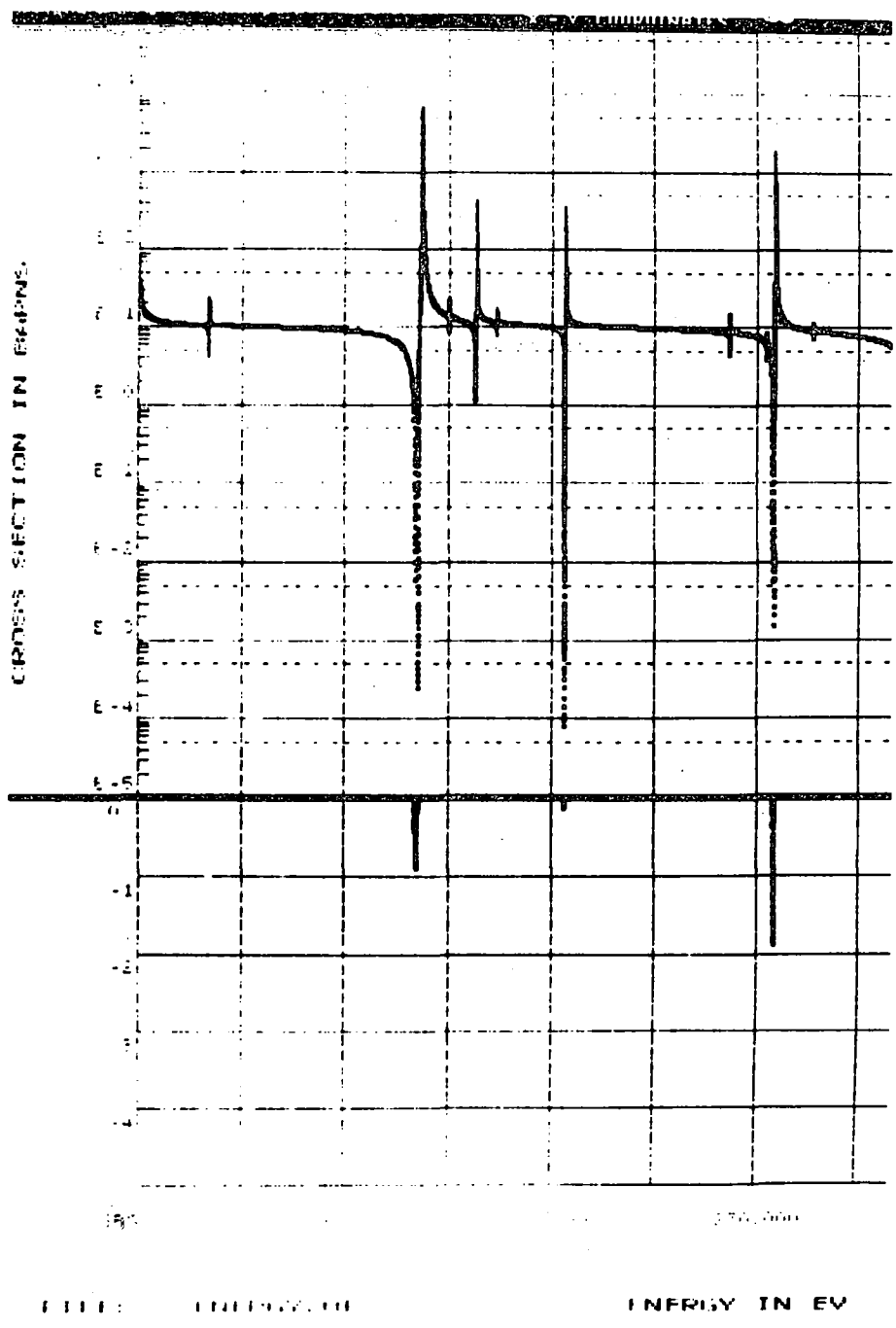

Fig.4: Same data as in Fig. 2, but zoomed in 150 eV to 290 eV energy region. This is shown to demonstrate that capability to zoom exists in the plotter program

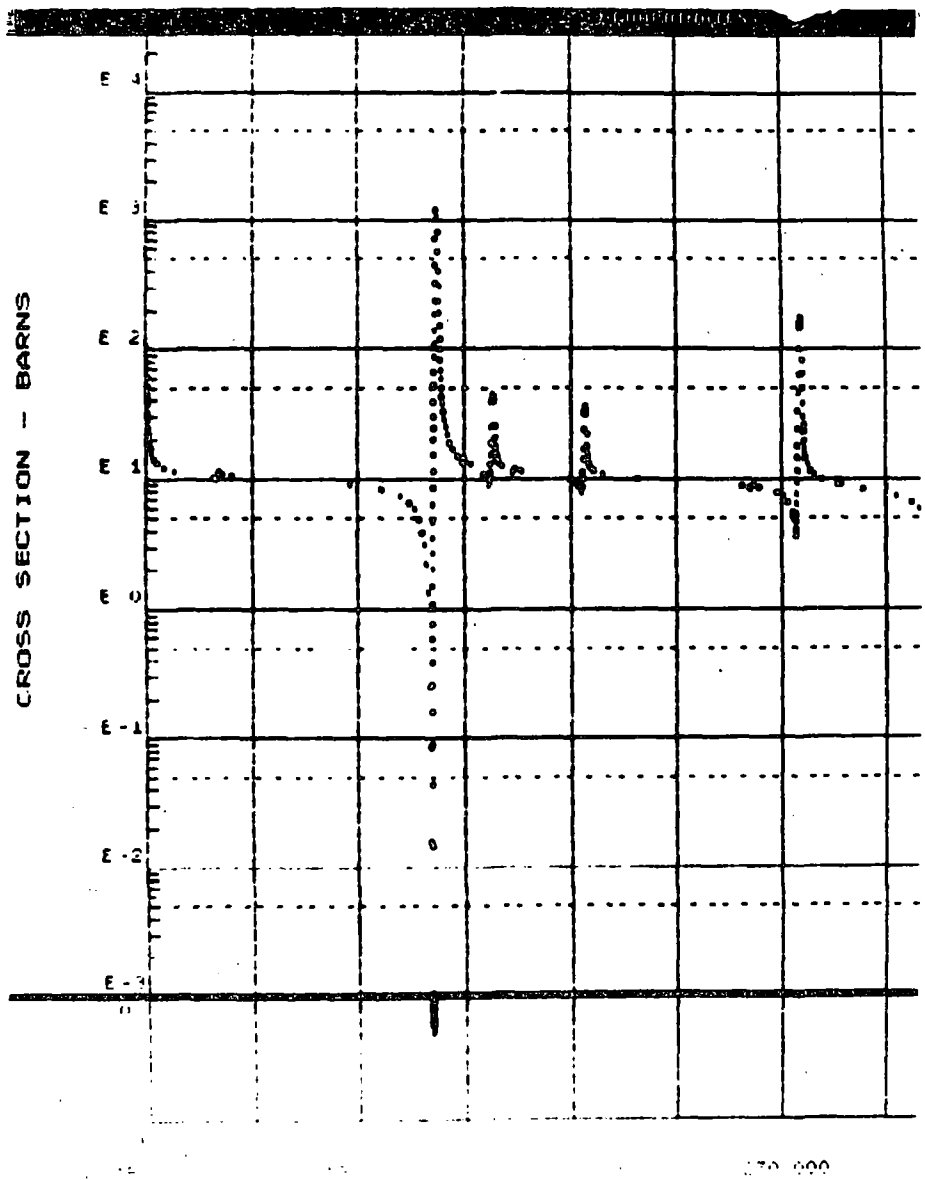

Fig.5: A plot of scattering cross sections retrieved from the output of SIGMA1 for Pu-242 of ENDF/B-V Actinides (Rev. 2) in 150 eV to 290 eV energy region. Same data and energy region as in Fig.4 but after Doppler broadening to 300 kelvin. The output of SIGMA1 was used to demonstrate that some of the negative cross sections may disappear at higher temperatures. In practice FIXUP is always run after RECENT before Doppler broadening.

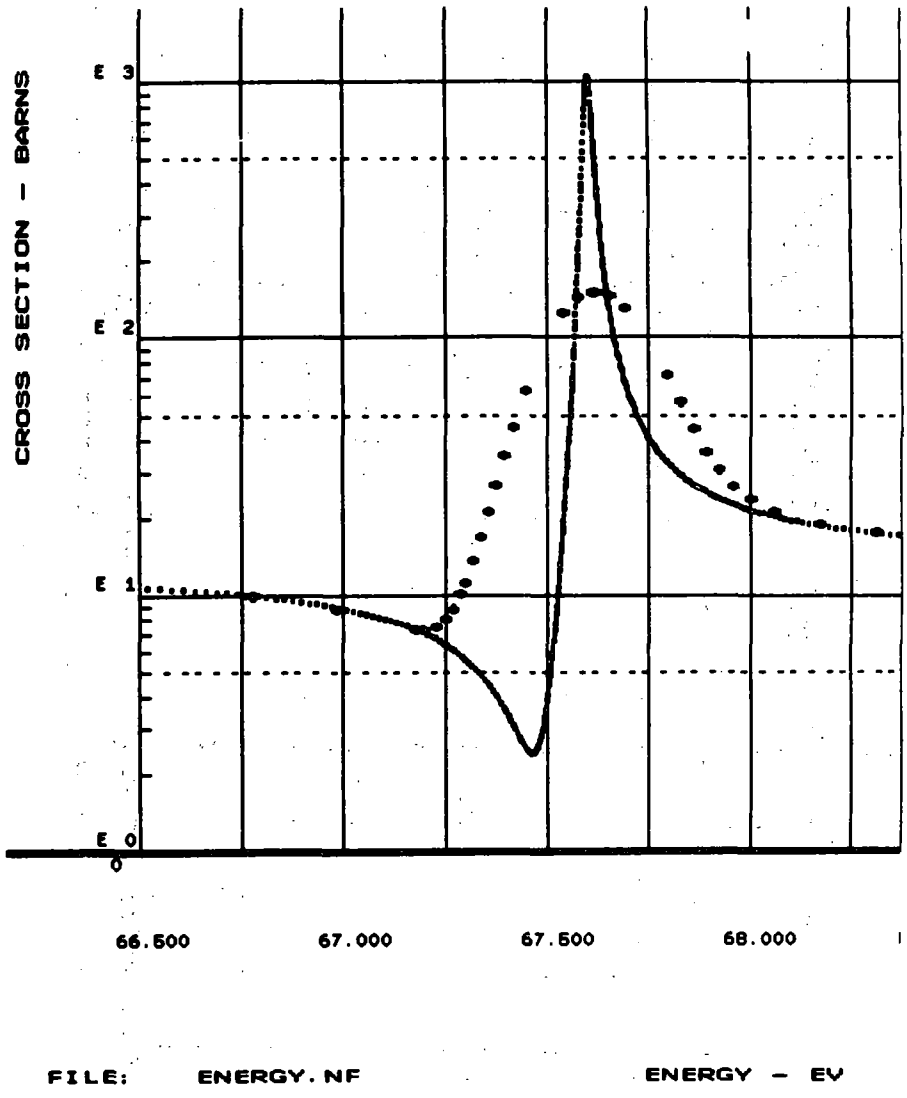

Fig. 6: Scattering cross sections of Pu-242 of ENDF/B-V Actinides (Rev. 2): Outputs of RECENT and SIGMA1 for a given energy region have been successfully superimposed in a single graph. The zoomed plot is shown for the energy region 66.5 eV to 68.25 eV energy region here, to demonstrate the effect of Doppler broadening. The plot shows that the number of points are less at higher temperatures due to smoothening as a result of Doppler broadening.

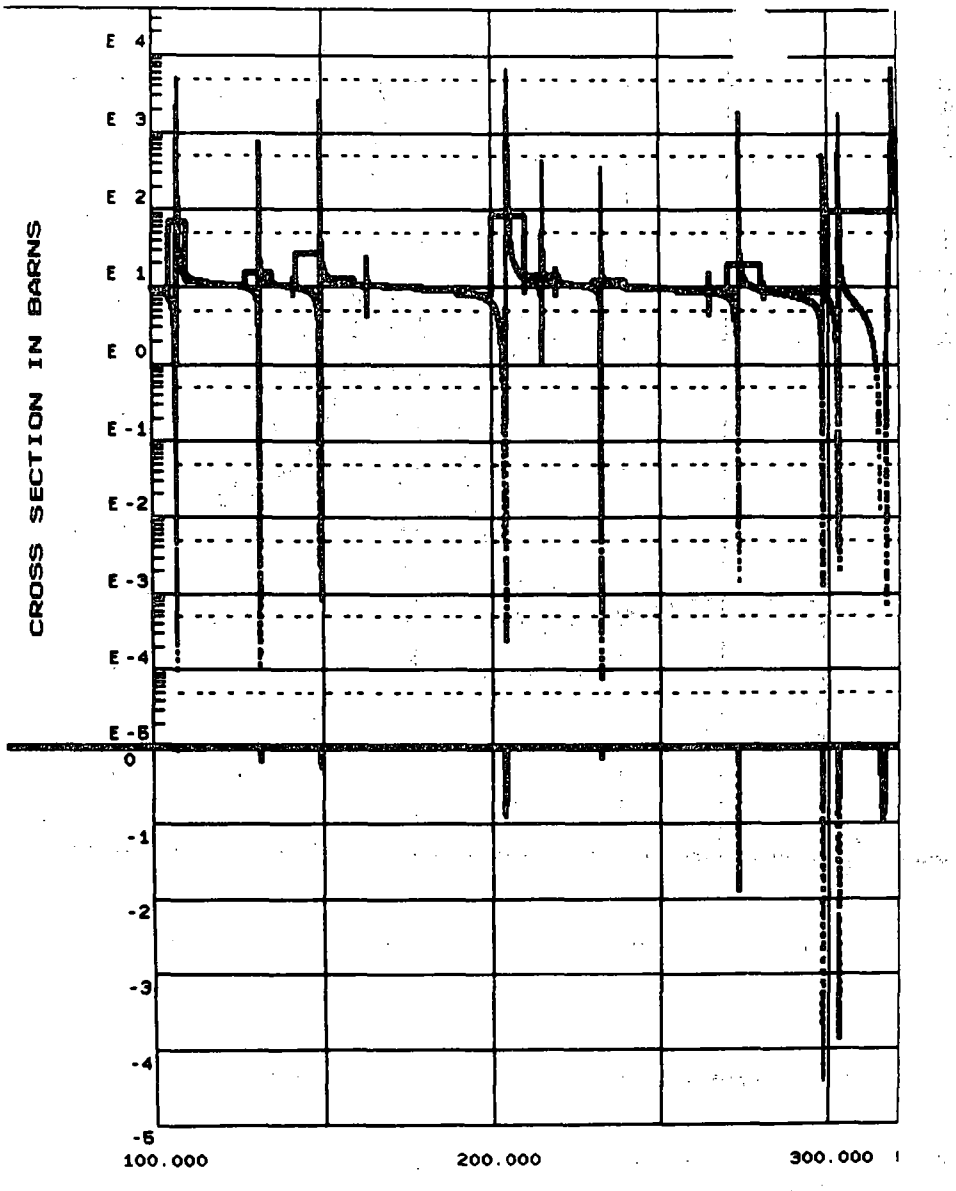

Fig. 7: Scattering cross sections of Pu-242 of ENDF/B-V Actinides (Rev. 2): The GROUPIE results in SAND II 620 group structure obtained using RECENT output is superimposed on the point data of RECENT file to show that generally the area covered by the negative cross sections are not large enough to make the group cross sections negative.

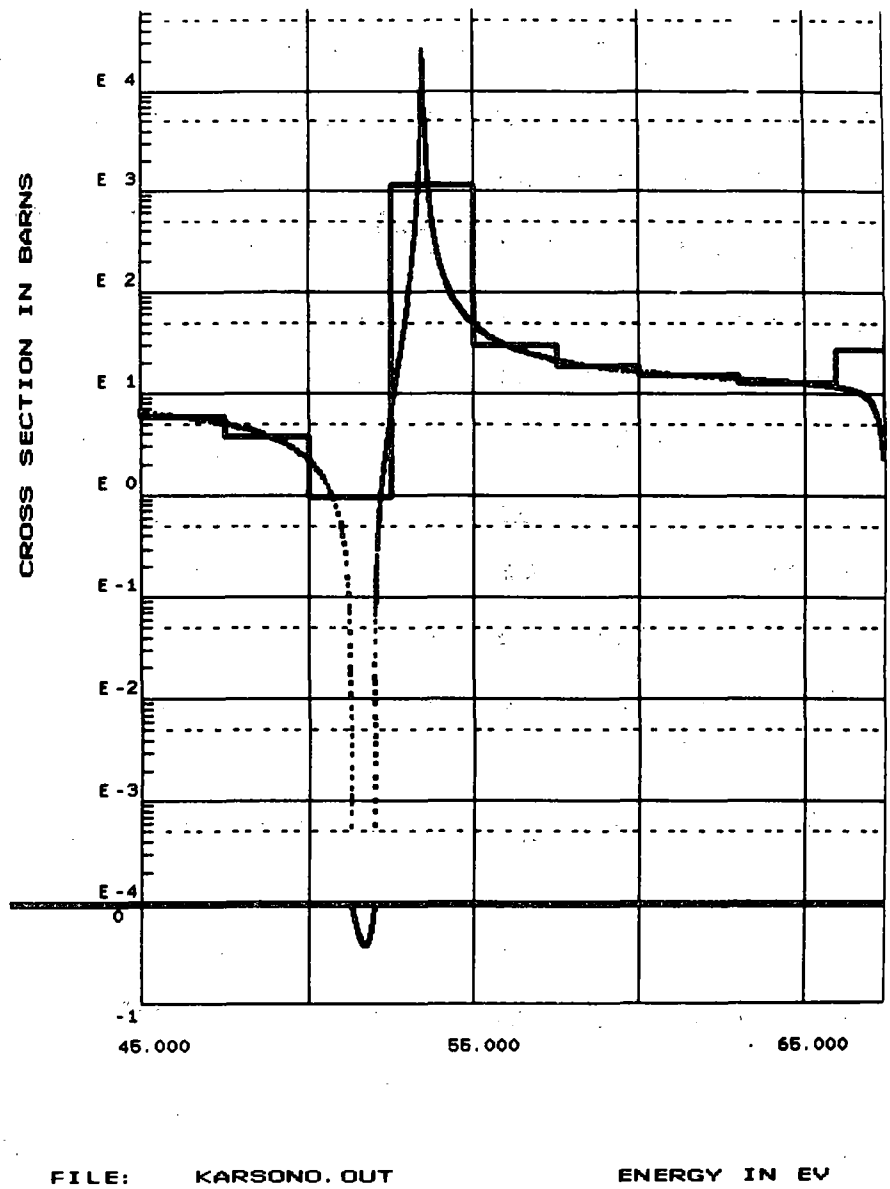

Fig. 8: Same as Fig. 7 but zoomed for 45 to 65 eV energy region.

A lot of other interesting graphs obtained during the mission period covering other reactions and other isotopes are not reproduced here to save space.

IV. Future Plans

The overall aim is to establish a workable and complete code system which will help BATAN perform rigorously all neutronic calculations for any research or original designs starting from the basic evaluated data file such as ENDF/B. Towards meeting this objective, several specific tasks have been identified. Following Fig. 1, which describes in general the various steps in the application calculations, we make below a few remarks for the prsent situation in BATAN.

Block-1 : Differential cross section data measurements

No experimental measurements of neutron interaction cross sections exists presently in BATAN. However, expeimenal raw data available from IAEA in the form of EXFOR will be used for evaluations in BATAN.

Block-2 : Phenomenological nuclear model calculations and systematics

As a beginning on a small scale, efforts are being initiated to perform nuclear model based predictions of cross sections. As a starting point, the two user friendly codes ALICE and ABAREX have been adapted in BATAN. It is planned to gain experience in the use of these codes.

Block-3 : Basic evaluated data files

- Carry out storage, retrieval and processing of new data files received from IAEA. Perform intercomparison of recent data files for a few important isotopes in point data and multigroup form.
- Further development of the local plotter program to cover other reactions and files may be made. In addition, when the hardware becomes available, attempts to commission the IAEA programs EVALPLOT and COMPLOT may be made.

Block-4 and Block-6 : Pre-processing codes and generation of multigroup constants:

-The pre-processing codes of IAEA, viz., LINEAR, RECENT, SIGMA1, GROUPIE, FIXUP etc., the code FEDGROUP-C and the Kalpakkam multigrouping codes REX1, REX2 have been successfully installed and commissioned in Serpong. Further experience is to be gained in using these codes for neutronic calculations. To start with, using these available processing code systems and a thorough understanding of group constants used by ANISN code, ENDF/B formatted files may be processed to produce multigroup library for use with ANISN code. As already stated, the other code systems with which experience at Serpong can be gained are the FEDGROUP-C code which processes ENDF/B and produces results of multigroup cross sections in WIMS format or the use of the more general NJOY code system of U. S. A. which has also the option to produce results in ANISN or WIMS formats.

Block-8 and 11: Neutronic calculations and engineering design

-A system of reactor design codes which can perform rigorous calculations starting from the basic evaluated data file and the transport equation shall be a long term development project. In the near future, existing codes like IAFUEL could be modified, replacing the four group condensed library by a compatible group constant set which will be generated from ENDF/B formatted files recently obtained from IAEA, using well tested available processing code systems.

- It is planned to generate compatible group constant set for use with IAFUEL code for the study of the behaviuor of G. A. Siwabessy research reactor when U O is replaced by U Si in the fuel elements.

- The ultimate goal would be mastering the network of computer programs for reactor analysis as shown in Fig. 9. For activities of each block in Fig. 9, related codes already exist in BATAN, but further experience and studies are desirable for activities stated for each block and to interface them for a full scale reactor design study.

THE NETWORK OF COMPUTER PROGRAMS

FOR REACTOR ANALYSIS

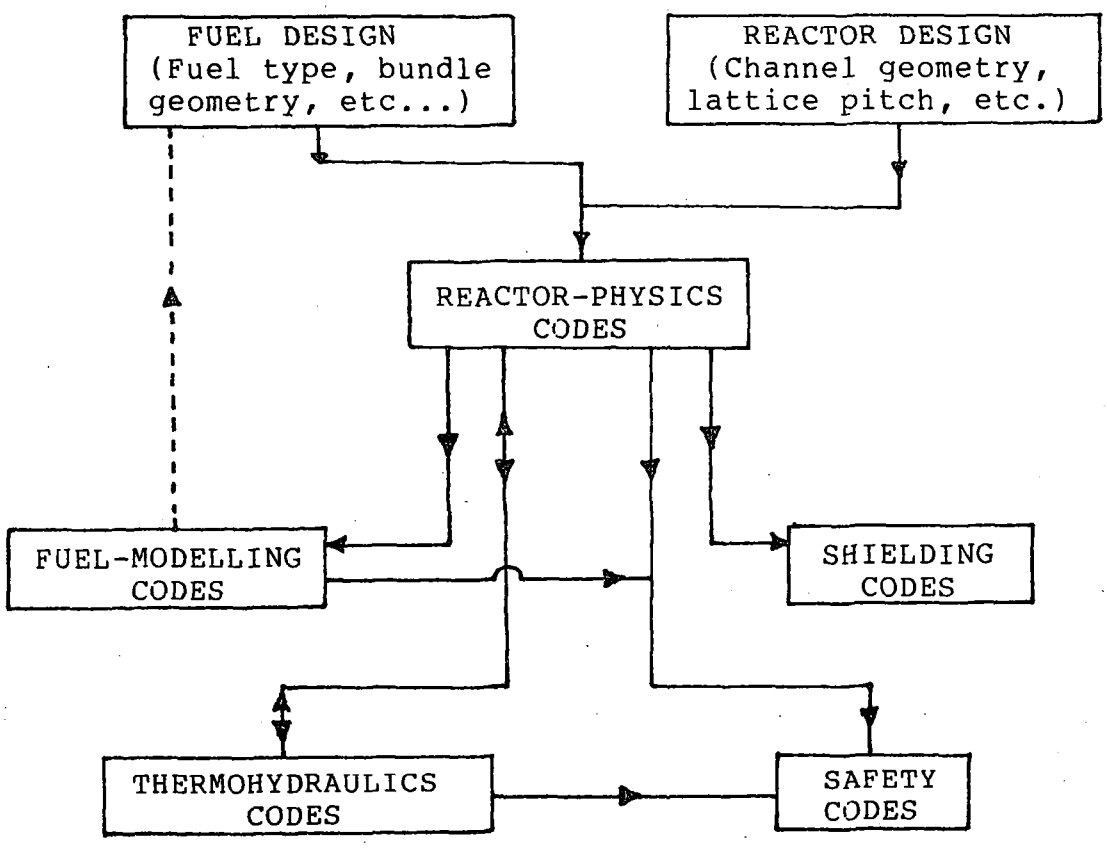

Fig. 9

V. Conclusions

Based on the availability of existing facilities and implementations, it is anticipated that the data base development and its applications can achieve a good progress in the future. Identifications of activities are formulated on the basis of a realistic programme. The code IAFUEL for the existing research reactor, viz., the G.A. Siwabessy reactor can be modified by updating the code IAFUEL's associated condensed group cross section library using complete set of group constants generated from ENDF/B formatted and recent basic evaluated data libraries, such as ENDF/B-V(Actinides), JENDL-2, BROAND, ENDL-86 etc.

Acknowledgement

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