Titolo

BUGENDF70.BOLIB - An ENDF/B-VII.0 Broad-Group Coupled (47 n + 20 γ) Cross Section Library in FIDO-ANISN Format for LWR Shielding and Pressure Vessel Dosimetry Applications

Descrittori

Tipologia del documento: Rapporto tecnico

Collocazione contrattuale:

Argomenti trattati: Fisica nucleare, dati nucleari, fisica dei reattori nucleari
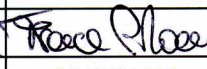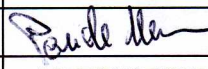
Sommario

The ENEA-Bologna Nuclear Data Group generated a broad-group coupled neutron and photon working cross section library in FIDO-ANISN format, dedicated to LWR shielding and pressure vessel dosimetry applications. This library, named BUGENDF70.BOLIB, was obtained in the neutron and photon energy group structures (47 n + 20 γ) of the ORNL BUGLE-96 similar library and it is based on the US ENDF/B-VII.0 library of evaluated nuclear data. BUGENDF70.BOLIB was produced through cross section collapsing of the ENEA-Bologna VITENDF70.BOLIB general-purpose library in AMPX format for nuclear fission applications using the updated and corrected ENEA-Bologna 2007 Revision of the ORNL SCAMPI nuclear data processing system. This library, based on the Bondarenko (f-factor) neutron resonance self-shielding method, has the same neutron and photon energy group structures (199 n + 42 γ) as the ORNL VITAMIN-B6 library in AMPX format (based on ENDF/B-VI.3 evaluated nuclear data) from which the BUGLE-96 library was generated at ORNL. A preliminary testing of the BUGENDF70.BOLIB library was performed through three-dimensional fixed source transport calculations with the TORT-3.2 discrete ordinates (S_N) code on the PCA-Replica 12/13 and VENUS-3 engineering neutron shielding benchmark experiments, specifically conceived to check the nuclear data and transport codes used in LWR radiation shielding and radiation damage analyses.

Note

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REV.	**DESCRIZIONE**	**DATA**			**REDAZIONE**	**CONVALIDA**	**APPROVAZIONE**

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BUGENDF70.BOLIB - An ENDF/B-VII.0 Broad-Group Coupled (47 n + 20 γ) Cross Section Library in FIDO-ANISN Format for LWR Shielding and Pressure Vessel Dosimetry Applications

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January 2013

1 - INTRODUCTION

Some years ago the ENEA-Bologna Nuclear Data Group started nuclear data processing and validation activities addressed to generate and/or to test broad-group coupled neutron/photon working cross section libraries, specifically dedicated to radiation shielding and radiation damage calculations for the light water nuclear fission reactors (LWRs) and, in particular, for the reactor pressure vessel (RPV) dosimetry analyses. The generation of working cross section libraries in ENEA-Bologna was initially /1/ and recently /2/ dedicated to the LWR radiation shielding and radiation damage applications since it was considered firstly important to produce working libraries for the most diffused types of nuclear power reactors all over the world. In fact about 81% of the 437 total world nuclear power reactor units are LWRs, i.e., 272 PWRs (about 62%) and 84 BWRs (about 19%), as reported in the December 16, 2012, updating of the IAEA-PRIS database.

It was considered useful (see in particular /3/) to offer new updated working cross section libraries specifically addressed, in particular, to improve the calculation accuracy of the radiation damage parameters like fast neutron flux, fast neutron fluence, iron displacement per atom rate (DPA/s) and total iron displacement per atom (DPA) in the structural components of the future and present operating LWRs. In fact in this kind of applications even more accuracy is required than in radiation shielding calculations where more conservative calculation approaches are usually applied. For example, in radiation damage calculations for the most important LWR component, the reactor pressure vessel, the calculation accuracy obtained is a fundamental parameter, directly linked with the RPV End-of-Life (EoL) prediction which is in its turn connected with well known fundamental nuclear safety aspects and relevant economic impacts.

The present technical report represents the user's manual of the ENEA-Bologna BUGENDF70.BOLIB broad-group coupled neutron/photon working cross section library in FIDO-ANISN /4/ format, dedicated to the previously cited LWR radiation shielding and radiation damage applications and, in particular, to the RPV dosimetry analyses. This library was obtained through problem-dependent cross section collapsing from the ENEA-Bologna VITENDF70.BOLIB /5/ fine-group cross section library for nuclear fission applications, based on the ENDF/B-VII.0 /6/ evaluated nuclear data library and on the Bondarenko /7/ (f-factor) method for the treatment of neutron resonance self-shielding and temperature effects. The VITENDF70.BOLIB library is a "pseudo-problem-independent" coupled neutron/photon library in AMPX format, i.e., a fine-group library prepared with enough detail in energy, temperatures and neutron resonance self-shielding so as to be applicable to a wide range of

physical systems. VITENDF70.BOLIB is characterized by the same neutron and photon energy group structures (199 neutron groups + 42 photon groups) as the ORNL DLC-0184/VITAMIN-B6 /8/ library in AMPX format, based on the ENDF/B-VI.3 /9/ evaluated data library. The BUGENDF70.BOLIB coupled working library adopts the neutron and photon energy group structures (47 neutron groups + 20 photon groups) of the ORNL DLC-0185/BUGLE-96 /8/ broad-group working library in FIDO-ANISN format, derived from the VITAMIN-B6 library and specifically conceived for the same previously cited applications in LWRs.

In particular broad-group working libraries, properly generated for the cited applications, are necessary to permit the use of the deterministic transport codes. The three-dimensional (3D) deterministic transport codes like, for example, the ORNL TORT /10/ discrete ordinates (S_N) code, can presently offer rigorous and reliable calculation solutions also for complex geometries, which could be described, up to few years ago, exclusively by the 3D stochastic transport codes like, for example, the LANL MCNP /11/ Monte Carlo code. In fact, if the 3D deterministic transport codes are properly assisted by dedicated pre/post-processor systems of programs (e.g., the ENEA-Bologna BOT3P /12/ /13/ /14/ /15/ and the Japanese TORTWARE /16/ systems) for the automatic generation and graphical verification of the spatial mesh grids of the reactor geometrical model, their performance can be highly competitive with that of the 3D Monte Carlo codes.

Despite the fact that, all over the world, the development of the 3D deterministic codes is going on and their use is increasingly appreciated also by the industrial organizations, it is not easy to find updated problem-dependent broad-group working cross section libraries for fission reactor shielding applications, free-released by the international distribution agencies (e.g., UNO-IAEA NDS, OECD-NEA Data Bank and ORNL-RSICC). In particular the packages of modular systems containing deterministic transport codes are distributed by the previously cited agencies without including working cross section libraries and this influences negatively the diffusion of the deterministic transport culture and expertise. This is mainly due to the fact that, in order to generate a problem-dependent broad-group working cross section library for the previously mentioned applications, it is necessary to collapse the fine-group cross sections of a multi-purpose library, as recommended by specific standards. This operation is performed using problem-dependent neutron and photon spectra for the various spatial regions of a specific type of reactor, pre-calculated with typical compositional, geometrical and temperature parameters. These data are very often considered confidential and this implies that these libraries are mainly produced for internal use and commercial purposes within the industrial organizations.

In this contribution, based on the compositional, geometrical and temperature parameters reported in the BUGLE-96 /8/ library user's manual for a typical PWR and a typical BWR, it was intentionally intended to follow the same data processing procedures and methodologies adopted at ORNL for the generation of the BUGLE-96 library which was widely and successfully used in LWR radiation shielding and radiation damage applications since 1996.
The LANL NJOY-99.259 /17/ and the ORNL SCAMPI /18/ nuclear data processing systems were selected and used to generate the group cross section libraries. NJOY-99.259 was employed, in particular, for the generation of the VITENDF70.BOLIB fine-group mother library. The so called ENEA-Bologna 2007 Revision /19/ of the SCAMPI system, an updated and corrected version specifically able, e.g., to process properly the recent nuclear data files

of the ENDF/B-VII.0 /6/ and JEFF-3.1.1 /20/ (see also /21/ and /22/) evaluated data libraries, was used to generate the BUGENDF70.BOLIB broad-group working library, through problem-dependent cross section collapsing. The AMPX format was adopted for the fine-group multi-purpose cross section library in order to have the compatibility with the AMPX-77 /23/ and SCAMPI nuclear data processing systems and with the SCALE-6 /24/ nuclear safety system, while the well known and diffused FIDO-ANISN format was selected for the broad-group working library.

A limited preliminary testing of the BUGENDF70.BOLIB library was successfully performed with the 3D TORT-3.2 discrete ordinates transport code on the PCA-Replica 12/13 /25/ /26/ (Winfrith, UK) and VENUS-3 /27/ (Mol, Belgium) engineering neutron shielding benchmark integral experiments, whose compositional and geometrical specifications were taken from the fission reactor shielding section of the ORNL-RSICC/ OECD-NEA Data Bank SINBAD /28/ /29/ international database of shielding benchmark experiments. It is underlined that the testing of the BUGENDF70.BOLIB library on the cited benchmark experiments is particularly meaningful since they were specifically conceived to verify and possibly to improve the accuracy of the calculation methodologies and nuclear data used in PWR radiation shielding and radiation damage calculations.

Finally, concerning the availability of similar recent libraries, it is stressed that the BUGENDF70.BOLIB and VITENDF70.BOLIB libraries, based on ENDF/B-VII.0 data, were respectively produced by the same data processing procedure employed to generate in parallel the similar BUGJEFF311.BOLIB /30/ and VITJEFF311.BOLIB /31/ ENEA-Bologna libraries, based on OECD-NEA Data Bank JEFF-3.1.1 data. The BUGJEFF311.BOLIB, VITJEFF311.BOLIB and VITENDF70.BOLIB libraries were already transferred for free distribution, in the reported chronological order, to OECD-NEA Data Bank and ORNL-RSICC. In addition, at the end of 2011 the BUGLE-B7 /32/ and VITAMIN-B7 /32/ ORNL libraries were diffused by ORNL-RSICC and then by OECD-NEA Data Bank. As the corresponding ENEA-Bologna counterparts, the BUGLE-B7 and VITAMIN-B7 libraries are based on ENDF/B-VII.0 data and adopt respectively the same neutron and photon group structures as the BUGLE-96 and VITAMIN-B6 libraries. On the other hand, differently from the ENEA-Bologna similar libraries processed with the cited NJOY and SCAMPI systems, the BUGLE-B7 and VITAMIN-B7 libraries were processed with the ORNL AMPX-6.1 nuclear data processing system not yet freely diffused.

1.1 - Background

Since many years, at the international level, a decrease of open activity was observed /33/ for debating production, features and performances of energy group-averaged working cross section libraries dedicated to radiation shielding and radiation damage applications in nuclear fission reactors.

This situation is unjustified, taking into account the actual increased performances of computers and 3D deterministic transport codes, the potential availability of collapsed and self-shielded group cross section libraries for different spectral, compositional and temperature conditions and, finally, the requirements of the nuclear safety authorities which encourage nuclear safety calculations possibly performed with transport codes based on different methods (stochastic and deterministic).

The methodology for the generation of energy group-averaged cross section libraries for nuclear radiation protection calculations for nuclear power plants, recommended by the ANSI/ANS-6.1.2-1999 (R2009) /34/ American National Standard and adopted in the present work, consists of a two stage process: 1) the processing of evaluated data files into a fine-group, pseudo-problem-independent library, followed by 2) the collapsing of the fine-group library cross sections into the broad-group, problem-dependent cross sections of the desired working library for the specific application. The problem-dependent library is then derived from the fine-group library by taking into account temperature and neutron resonance self-shielding specifications and collapsing into a smaller number of groups. As underlined in the VITAMIN-B6 and BUGLE-96 /8/ library user's manual, this approach removes from the end user the need to deal with the complex task of producing a group-averaged library from the evaluated data files, which contain a mix of point and functional data. This approach also reduces the user's responsibility to the operation of a finite number of well-defined processing codes. Hence a higher level of standardization and reliability is achieved since the user can focus only on those features which are special to his application.

In other words, following the previously cited standard, a working cross section library for a specific application should not be obtained directly in broad-group (tenths of groups) energy structures, using generic neutron and photon weighting fluxes derived from analytical functions (e.g., Maxwellian thermal neutron spectrum, 1/E neutron slowing-down spectrum, neutron fission spectrum, 1/E photon spectrum, etc.). In fact it is considered a more accurate approach to obtain the broad-group cross sections of the working libraries through proper cross section collapsing of fine-group (hundreds of groups) pseudo-problem independent cross section libraries, based, e.g., on the Bondarenko /7/ (f-factor) method for the treatment of neutron resonance self-shielding and temperature effects. It is recommendable, in particular, to collapse the fine-group cross sections using in-core and ex-core neutron and photon spectra, properly pre-calculated with a transport code for the specific compositional, geometrical and temperature data of the various spatial regions of a specific reactor type.

These libraries are required to run the deterministic transport codes which are, e.g., included in the following systems of deterministic codes: the US packages DOORS-3.2 /35/ and PARTISN-5.97 /36/ or the Russian package CNCSN 2009 /37/, distributed by the OECD-NEA Data Bank and ORNL-RSICC. On the other hand these packages, unlike those which include Monte Carlo codes like MCNP /11/, do not contain any working cross section library. The production of fine-group coupled neutron/photon pseudo-problem-independent libraries, based on the Bondarenko neutron resonance self-shielding method (e.g., of the type similar to the VITAMIN-B6 /8/ library) continues in several research institutes (ENEA, KAERI, ORNL, etc.). On the other hand, the free availability of derived broad-group working libraries of collapsed and self-shielded cross sections (e.g., of the type similar to the BUGLE-96 /8/, BUGJEFF311.BOLIB /30/, BUGLE-B7 /32/ or BUGENDF70.BOLIB libraries) is practically absent.

The previously cited decrease of open activity dedicated to the generation of broad-group working cross section libraries for fission reactor shielding applications is very probably due to the fact that it is traditionally considered highly convenient to perform 3D radiation shielding analyses for complex geometries with the combinatorial geometry approach included in the Monte Carlo codes (e.g., MCNP) using continuous-energy (point-wise)

processed cross section libraries, independent in practice from the specific neutron and photon spectral environment.

Differently from the 3D Monte Carlo codes using a single processed continuous-energy cross section library to treat the different spectral environments of interest, when deterministic transport codes are employed it is not practically meaningful to use only one broad-group spectrum-independent working cross section library for any kind of application.

As reported in the user's manual of the VITAMIN-B6 and BUGLE-96 libraries, the generation of group-averaged cross section libraries with broad-group energy structures is primarily justified for reasons of economy. Despite the impressive performance of modern supercomputers it is still often impractical to perform two and three-dimensional radiation transport analyses using point-wise data or finely structured multi-group data, especially if fine resolution is needed for the space or angular meshes. In particular the 3D deterministic transport codes, could have convergence problems when fine-group working libraries are used together with hundreds of thousands of volumetric spatial meshes, possibly needed to describe accurately complex in-core and ex-core reactor geometrical models used to perform radiation shielding and radiation damage analyses.

Even for one-dimensional analyses, it is often more efficient to use few-group data to perform the initial scoping analysis and then advance to finer group data as accuracy requirements become more stringent. The establishment of reference broad-group libraries is desirable to avoid duplication of effort, both in terms of the library generation and verification, and to assure a common database for comparisons among participants to a specific calculation program.

Taking into account this background and following the previously cited recommended methodology of generating a broad-group working cross section library dedicated to a specific reactor type, it is then necessary to know compositional, geometrical and temperature data, typical of the specific nuclear reactor, which are not normally freely released. These data, i.e., the typical homogenized atomic densities and the specific temperatures of the nuclide mixtures for the various in-core and ex-core spatial regions along the one-dimensional reactor radial geometry at the core midplane, are necessary for two fundamental reasons.

The first reason is that these data are necessary to pre-calculate the in-core and ex-core neutron and photon spectra in proper locations of each representative reactor zone of a specific reactor type in order to perform then a problem-dependent cross section collapsing of the fine-group cross sections of the pseudo-problem-independent mother library.

The second reason is that the mentioned data are necessary to calculate the correct problem-dependent neutron resonance self-shielding of the broad-group cross sections of the working library for a specific nuclear reactor type.

The fact that these data are still considered confidential by the industrial organizations implies that a wide and open data processing effort to produce this kind of libraries is not presently possible. This fact in its turn induces the risk that the culture and the technical expertise related to deterministic calculations may progressively disappear within the research and development organizations and university institutions whilst there is still the interest that they continue to be developed, within the industrial organizations, in a commercial and self-

referential perspective that does not guarantee, in general, a completely satisfactory approach to nuclear safety.

Moreover, despite the free availability of systems (e.g., NJOY/TRANSX /17/ /38/, AMPX-77 /23/, SCAMPI /18/ /19/, SCALE /24/, etc.) which permit the problem-dependent nuclear data processing in order to obtain broad-group working libraries of collapsed and self-shielded cross sections from fine-group general-purpose cross section libraries, the expertise about the nuclear data processing systems and methods is not generally widespread at the industrial level.

When, on the contrary, the industrial organizations are equipped with the necessary human resources and technical tools to perform detailed nuclear data processing addressed to generate multi-purpose fine-group libraries and derived collapsed working libraries of self-shielded cross sections for a specific type of fission reactor (the so called "custom-made" working cross section libraries), it is in any case necessary to follow the quality assurance approach.
This implies that, in order to reduce possible errors in the data entry during the problem-dependent data processing phase of a custom-made working library, it is in parallel recommended using of already processed working libraries dedicated to the same applications, with parameterized sets of self-shielded cross sections for a similar type of nuclear fission reactor.

In other words it is in any case important to verify and intercompare the results obtained with the custom-made working libraries together with the results obtained through working cross section libraries with parameterized sets of self-shielded cross sections, like for example BUGLE-96, BUGJEFF311.BOLIB, BUGLE-B7 or BUGENDF70.BOLIB, dedicated to LWR shielding and pressure vessel dosimetry.

Concerning the deterministic codes, it is really an upsetting fact that currently the 3D deterministic transport codes cannot be fully used due to the lack of broad-group working libraries also when their use should be strongly recommended and, in any case, competitive with the use of the 3D Monte Carlo stochastic codes. It is interesting to note that the industrial organizations continue to be interested in the use /39/ or even in the development /40/ of the 3D deterministic codes. Moreover they directly develop /41/ or outsource to external nuclear data processing working groups, under specific contracts, the generation of broad-group working cross section libraries for radiation shielding and radiation damage calculations with the deterministic transport codes. In fact they must fulfil quality assurance procedures with respect to the nuclear safety authority requirements and when deterministic codes are employed there is no need, as in the case of the Monte Carlo codes, to justify the validity of the statistics adopted since the deterministic codes are based on rigorous analytical solutions of the neutral particle transport equations. Moreover it is underlined that the 3D deterministic codes applied to radiation shielding and radiation damage analyses assure, with only a single run, a simultaneous and accurate average dose determination in every spatial position of the reactor geometrical model. It is then very important to underline that the deterministic transport codes permit reliable and effective sensitivity and uncertainty analyses, particularly recommended in the data validation activity and in a modern and rigorous approach to the industrial project of a nuclear reactor.

The 3D deterministic transport codes (e.g., the TORT /10/ code in the DOORS package, the PARTISN parallel code, etc.), which necessarily use the group-wise cross section libraries, increased in recent years their calculation performances in an impressive way and expanded their applicability to handle complex geometries, reaching in many cases the detail offered by the 3D Monte Carlo codes (e.g., MCNP). This result was achieved through the use of pre/post-processor systems of ancillary programs (e.g., the ENEA-Bologna BOT3P /12/ distributed by OECD-NEA Data Bank and ORNL-RSICC), dedicated to the 2D and 3D deterministic transport codes. In particular, with the support of BOT3P, based on combinatorial geometry algorithms, it is now easily possible to generate automatically detailed spatial mesh grids not only for the 2D and 3D transport codes of the DOORS system but also for any other possible transport code (through simple interfaces dedicated to manage the BOT3P binary output files), together with the graphical verification of the input data of the geometrical model.

During the last 10-15 years, the 3D discrete ordinates (S_N) transport codes increased their competitiveness with respect to the corresponding 3D Monte Carlo stochastic codes, obtaining comparable or even more convenient performances in terms of CPU times, with the same calculation precision, similar description capability of complex geometries and suitable simulation of different neutron and photon spectral conditions. Moreover 3D discrete ordinates codes like the US ATTILA /42/ commercial code, with unstructured spatial grids (finite elements) can now treat not only the neutral but also the charged particle transport as the more conventional discrete ordinates codes of the Russian CNCSN 2009 system of deterministic codes with structured spatial grids: the 1D code ROZ-6.6, the 2D codes KASKAD-S-2.5 (serial) and KASKAD-S-3.0 (parallel multi-threaded) and the 3D codes KATRIN-2.0 (serial) and KATRIN-2.5 (parallel multi-threaded).

Since deterministic transport codes are going to be employed in the analysis of the Generation IV nuclear reactor projects within the European Union research activities, it would be highly recommended that a specific interest dedicated to the generation of broad-group working cross section libraries should be promoted.

During the last years, the ENEA-Bologna Nuclear Data Group has performed several actions addressed to generate practical tools to increase, in particular, the performance and competitiveness of the 2D and 3D deterministic transport codes, following the recommendations proposed by the OECD-NEA Data Bank.

1. Several fine-group cross section libraries /2/ /5/ /30/ /31/ /43/ /44/ /45/ /46/ /47/ /48/ for nuclear fission applications were generated and are at present freely distributed by OECD-NEA Data Bank and ORNL-RSICC.

2. A pre/post-processor system /12/ /13/ /14/ /15/ of programs for the automatic spatial mesh generation, dedicated to the 2D and 3D deterministic transport codes, was developed and it is now freely distributed by OECD-NEA Data Bank and ORNL-RSICC.

3. Transport analyses dedicated to fission reactor neutron shielding benchmark experiments /1/ /2/ /49/ /50/ /51/ /52/ /53/ were performed also within the activities /51/ of the OECD-NEA Nuclear Science Committee TFRDD Task Force /3/ on nuclear fission reactor ageing problems.

4. The whole set of the IRDF-2002 /54/ dosimetry cross sections was processed /55/ in the 47 neutron group structure of the BUGLE-96 cross section library using a flat weighting neutron spectrum and neutron weighting spectra calculated with JEFF-3.1.1 and ENDF/B-VII.0 data, at one quarter of the thickness of a typical PWR pressure vessel: these data are partially included in the package of the present BUGENDF70.BOLIB library.

1.2 - ENDF/B-VII.0 Evaluated Nuclear Data Library

The cross section advances in the ENDF library support needs in a wide variety of applied technologies. Complete cross section evaluations are needed in radiation transport simulation codes that are used to model the neutronics, activation and nuclear transmutations, energy deposition and absorbed dose, etc.. The applications include advanced reactor design, nuclear waste transmutation and fuel cycles, nuclear criticality safety, fusion, medical applications (isotope production, external beam therapy, etc.), nonproliferation and national security, space physics, radiation protection and shielding.

The work performed for the ENDF/B-VII.0 /6/ library represented a coordinated effort for five years by researchers from many US institutions, organized by CSEWG (Cross Section Evaluation Working Group) which is the organization that oversees the development of this database. The principal advances were dictated by specific programmatic priorities set by several laboratories and by the Department of Energy (DOE). The DOE Office of Science, Office of Nuclear Physics' US Nuclear Data Program (USNDP) provided the bulk of the support for bringing the various capabilities developed at different laboratories together under CSEWG, and for the National Nuclear Data Center to maintain and archive the ENDF databases at Brookhaven National Laboratory. Most of the underlying research was supported by the DOE National Nuclear Security Agency's Advanced Simulation and Computing (ASC), Nuclear Criticality Safety and Nonproliferation Research and Engineering programs, in addition to the Office of Science. The DOE Nuclear Energy (NE) office supported work related to advanced fuel cycles and advanced reactors. Important support also came from the DOE Naval Reactor Laboratories and from the National Institute for Standards and Technology (NIST).

The development of complete, evaluated cross section data files depends upon a variety of expertises: nuclear experimentation; nuclear theory and model predictions; statistical analysis; radiation transport physics; computer code and database development; processing of nuclear data; and fundamental and integral validation against experiments that include criticality and neutron transmission (shielding) measurements. This effort has brought together scientists from these different disciplines to create the ENDF/B-VII.0 library.

A cross section library is developed not only for the purpose of providing accurate basic physics data, isotope-by-isotope, but also to perform well, as an ensemble, in applied simulations. This is particularly important for nuclear criticality applications, where for some critical assemblies the performance may depend sensitively upon H-1, O-16, U-235, U-238, etc. data. Since all cross sections are known only to a certain level of precision, significant attention was paid to ensure that the evaluated cross sections perform together well as a group in validation simulations of these critical assemblies.

The ENDF/B-VII.0 evaluated nuclear data library was developed in the period 2002-2006 and it was released (see /6/) in December 2006.

The ENDF/B-VII.0 library includes all the data in the internationally-accepted ENDF-6

format /56/. It contains 14 sublibraries ordered according to the NSUB sublibrary identification number, defined by the ENDF-6 format. The total number of materials in ENDF/B-VII.0 has increased considerably with respect to the previous (VI.8) version of the ENDF/B library, largely thanks to the new decay data sublibrary. Although the ENDF/B library is widely known for evaluated neutron cross sections, a considerable amount of non-neutron data are contained as well.

As discussed below, out of the total of 14 sublibraries, there are two new sublibraries, 7 sublibraries were considerably updated and extended, while the remaining 5 sublibraries were taken over from ENDF/B-VI.8 without any change:

1. The photonuclear sublibrary is entirely new. It contains evaluated cross sections for 163 materials (all isotopes) mostly up to 140 MeV. The sublibrary has been supplied by Los Alamos National Laboratory (LANL) and it is largely based on the IAEA-coordinated collaboration completed in 2000. This project mostly used the evaluation methodology and modelling tools for photonuclear reactions developed at LANL.

2. The photo-atomic sublibrary has been taken over from ENDF/B-VI.8. It contains data for photons from 10 eV up to 100 GeV interacting with atoms for 100 materials (all elements). The sublibrary has been supplied by Lawrence Livermore National Laboratory (LLNL).

3. The decay data sublibrary has been completely re-evaluated and considerably extended by the National Nuclear Data Center, Brookhaven National Laboratory (BNL).

4. The spontaneous fission yields were taken over from ENDF/B-VI.8. The data were supplied by LANL.

5. The atomic relaxation sublibrary was taken over from ENDF/B-VI.8. It contains data for 100 materials (all elements) supplied by LLNL.

6. The neutron reaction sublibrary represents the heart of the ENDF/B-VII.0 library. The sublibrary has been considerably updated and extended, with a number of entirely new evaluations. It contains 393 materials, including 390 isotopic evaluations and 3 elemental ones (C, V and Zn). These evaluations can be considered to be complete (the only exception is Es-253 that contains (n,γ) dosimetry cross sections) since they contain data for all important reaction channels including energy spectra and angular distributions for use in neutronics calculations. Important improvements were made to the actinide nuclides by LANL, often in collaboration with ORNL. Evaluations in the fission product range (Z = 31-68) have been entirely changed. ENDF/B-VII.0 contains fission product evaluations for 219 materials, with 71 materials evaluated by BNL, 2 by LLNL, 1 by LANL-BNL and the remaining 145 materials produced by the international project (OECD-NEA WPEC). Of the 393 materials, about 2/3 of the evaluations are based upon recent important contributions from the US evaluators. The remaining evaluations were adopted from other sources (mostly the JENDL-3.3 library). Livermore provided β-delayed γ-ray data for U-235 and Pu-239, for the first time in ENDF/B.

7. Neutron fission yields were taken over from ENDF/B-VI.8. The data were supplied by LANL.

8. The thermal neutron scattering sublibrary contains thermal scattering law data with 20 evaluations, largely supplied by LANL, with several important updates and extensions (in seven evaluations also based on the work /57/ by IKE Stuttgart, Germany). The remaining evaluations were taken over from the ENDF/B-VI.8 library.

9. The neutron cross section standards sublibrary is new. Although standards traditionally constituted part of the ENDF/B library, in the past these data were stored on a tape with a specific tape number. As the concept of tapes has been abandoned in ENDF/B-VII.0, the neutron cross sections standards sublibrary has been introduced. Out of 8 standards materials, 6 were newly evaluated, while the He-3(n,p) and C-nat(n,n) standards were taken over from ENDF/B-VI.8. The standard cross sections were completely adopted by the neutron reaction sublibrary except for the thermal cross section for U-235(n,f) where a slight difference occurs to satisfy thermal data testing. These new evaluations come from the international collaboration coordinated by the IAEA and OECD-NEA WPEC; the US effort was led by NIST and LANL.

10. The electro-atomic sublibrary was taken over from ENDF/B-VI.8. It contains data for 100 materials (all elements) supplied by LLNL.

11. The proton-induced reactions were supplied by LANL, the data being mostly to 150 MeV. There are several updates and several new evaluations.

12. The deuteron-induced reactions were supplied by LANL. This sublibrary contains 5 evaluations.

13. The triton-induced reactions were supplied by LANL. This sublibrary contains 3 evaluations.

14. Reactions induced with He-3 were supplied by LANL. This sublibrary contains 2 evaluations.

The major US laboratory contributors to the ENDF/B-VII.0 library are cited herewith. A dominant contributor to the evaluations is LANL, who provided the many actinide evaluations in the neutron reaction sublibrary, almost all the evaluations in the neutron thermal scattering sublibrary, many photonuclear and all the charged particle evaluations. BNL contributed the decay data sublibrary and many fission product evaluations in the neutron sublibrary; ORNL contributed neutron resonances for several actinides of key importance; LLNL contributed 3 atomic sublibraries (carried over from previous evaluations), and NIST played the leading role in developing neutron cross section standards. BNL performed Phase 1 testing (data verification), LANL was the leading laboratory in Phase 2 testing (data validation) and BNL is responsible for archival and dissemination of the library.

1.3 - Cross Section Processing and Testing

The calculation approach used to produce the BUGENDF70.BOLIB library, based on the ENDF/B-VII.0 /6/ evaluated nuclear data library, is consistent (see also 1.1) with the US ANS standard "Neutron and Gamma-Ray Cross Sections for Nuclear Radiation Protection Calculations for Nuclear Power Plants" (ANSI/ANS-6.1.2-1999 (R2009) /34/). Specifically the ENDF/B-VII.0 data were first processed into a fine-group cross section set (the VITENDF70.BOLIB /5/ library) similar to the VITAMIN-B6 /8/ pseudo-problem-independent library and then collapsed into a broad-group set (the BUGENDF70.BOLIB library) similar to the BUGLE-96 /8/ working library, derived from VITAMIN-B6.

The selected approach employs both the following modular nuclear data processing systems: the LANL NJOY-99.259 /17/ system and the ENEA-Bologna 2007 Revision /19/ of the ORNL SCAMPI /18/ system. Several modules of NJOY were used to process the neutron interaction, the photon production and the photon interaction data from the ENDF/B-VII.0 formats to a group-averaged format.
In order to process correctly modern evaluated nuclear data like the ENDF/B-VII.0 data files, it was necessary to develop an updated and corrected version of the SCAMPI system, originally developed at ORNL from the AMPX-77 /23/ system and already employed at ORNL to generate the BUGLE-96 library from the VITAMIN-B6 library. In particular, the previously cited so called "ENEA-Bologna 2007 Revision of SCAMPI", able to generate and to read data in AMPX format, was developed and was released to OECD-NEA Data Bank and ORNL-RSICC.
In particular the ENEA-Bologna 2007 Revision of SCAMPI, through the revised and corrected SMILER module, was employed to read the double-precision GENDF binary files from the NJOY-99.259 nuclear data processing system, to translate the intermediate NJOY file into the AMPX master format for the VITENDF70.BOLIB fine-group library and, finally, to calculate the total (prompt + delayed) neutron fission spectra and average numbers of neutrons emitted per fission, taking into account that their delayed neutron components could not be previously obtained with the original ORNL SMILER version of the SCAMPI system. The BONAMI module was used to self-shield the VITENDF70.BOLIB cross sections taking into account the compositional, geometrical and temperature specifications typical of PWR and BWR calculation models. These self-shielded cross sections in the VITENDF70.BOLIB neutron and photon fine-group energy structures were used in transport calculations to determine the problem-dependent weighting spectra employed to generate the BUGENDF70.BOLIB neutron and photon broad-group collapsed cross sections through the revised and corrected MALOCS module, which was used, in particular, to perform the cross section collapsing of the rectangular fission matrices.

A detailed description of the data processing performed for the generation of the VITENDF70.BOLIB fine-group library is given in Chapter 2 while the specifications and processing methods used to generate the BUGENDF70.BOLIB broad-group working library are described in Chapter 3. Finally, the results of a preliminary but important validation effort dedicated to the BUGENDF70.BOLIB library is presented in Chapter 4.

2 - VITENDF70.BOLIB FINE-GROUP LIBRARY SPECIFICATIONS

The VITENDF70.BOLIB /5/ library is a fine-group pseudo-problem-independent cross section library, based on the Bondarenko /7/ (f-factor) method for the treatment of neutron resonance self-shielding and temperature effects. This library, generated in ENEA-Bologna by the Nuclear Data Group, is a coupled neutron/photon library in AMPX format for nuclear fission applications, based on the US ENDF/B-VII.0 /6/ evaluated nuclear data library. It has the same neutron and photon energy group structures (199 neutron groups + 42 photon groups) and general basic features as the ORNL DLC-184/VITAMIN-B6 /8/ American library in AMPX format, based on the US ENDF/B-VI.3 /9/ evaluated nuclear data library.

The VITENDF70.BOLIB library was generated through an updated automatic calculation procedure based on the LANL NJOY-99.259 /17/ nuclear data processing system, with the updating "upnea049", and the ENEA-Bologna 2007 Revision /19/ of the ORNL SCAMPI /18/ nuclear data processing system. VITENDF70.BOLIB was extensively tested on many thermal, intermediate and fast neutron spectrum criticality safety benchmark experiments.

A revised version of the SMILER module of the ENEA-Bologna 2007 Revision of the SCAMPI system was used to translate the fine-group data from the GENDF format into the AMPX master library format of the VITENDF70.BOLIB library. In parallel, automatic multiplication of the term of ℓ-th order of the Legendre polynomial (P_ℓ) expansion of the scattering cross section matrix by the ($2\ell + 1$) factor was performed by the SMILER module for all the processed data files of the nuclides contained in the library.

The cross section files of VITENDF70.BOLIB in AMPX format can be exclusively treated by the updated ENEA-Bologna 2007 Revision of the SCAMPI system which assures a high level of flexibility in the production of working cross section libraries and an evident consistency with the VITAMIN-B6 library generation methods. It is underlined, in particular, that the ENEA-Bologna revised version of the SMILER module, contained in the ENEA-Bologna 2007 Revision of the SCAMPI system, permits to obtain separately the prompt neutron fission spectrum (MF=6, MT=18), the delayed neutron fission spectrum (MF=5, MT=455) and the total neutron fission spectrum needed, e.g., in the fixed source transport calculations for reactor radiation shielding applications. On the contrary, from the original ORNL SMILER version (see /18/), used to generate VITAMIN-B6 and VITJEF22.BOLIB /48/, it is possible to obtain only the prompt neutron component of the fission spectrum.

At present, the availability of the VITENDF70.BOLIB library permits to obtain derived working libraries of collapsed and self-shielded cross sections, through the ENEA-Bologna 2007 Revision of the SCAMPI system. More specifically, the cross sections can be collapsed by the MALOCS module, can be self-shielded by the BONAMI module and finally can be generated in the AMPX or FIDO-ANISN /4/ format. The cross sections in AMPX format can be used by the XSDRNPM one-dimensional (1D) discrete ordinates transport code, included in the ORNL AMPX-77 /23/ and SCAMPI nuclear data processing systems, or in the ORNL SCALE-6 /24/ nuclear safety system. Concerning the cross sections in FIDO-ANISN format, they can be used by the discrete ordinates (S_N) deterministic codes ANISN-ORNL (1D), DORT (2D) and TORT (3D) of the DOORS /35/ system, by the PARTISN /36/ (1D, 2D and 3D) parallel time-dependent discrete ordinates system and, finally, by the MORSE /58/ 3D Monte Carlo stochastic code.

2.1 - Name

The fine-group pseudo-problem-independent library which generated the BUGENDF70.BOLIB broad-group working library is designated as VITENDF70.BOLIB /5/. "VIT" suggests that the main features of the library are similar to those of the ORNL VITAMIN-B6 /8/ library and to the ENEA fine-group libraries /31/ /44/ /48/ in AMPX format with the same neutron and photon energy group structures. The "ENDF70" designation conveniently reflects the origin of the evaluated data: the ENDF/B-VII.0 /6/ evaluated nuclear data library. Finally, "BOLIB" means BOlogna LIBrary and so it is indicative of the place of production of the library.

2.2 - Materials, Temperatures and Background Cross Sections

A set of 183 cross section files, derived from the ENDF/B-VII.0 /6/ evaluated nuclear data library, was processed for the VITENDF70.BOLIB /5/ fine-group library. In particular the complete list of the included nuclides is reported in TAB. 2.1 together with the corresponding Z atomic numbers, the ENDF/B-VII.0 MAT numbers, the AMPX identifiers and a flag (YES/NO) indicating the presence of gamma-ray production data in the specific evaluated nuclear data file. It is underlined that, in this set of processed data files, only two data files correspond to evaluated natural elements (C-nat and V-nat) whereas the other ones correspond to single isotope evaluated data files.

The Bondarenko /7/ (f-factor) method was used for handling neutron resonance self-shielding and temperature effects. As for VITAMIN-B6 /8/, all the 177 standard (not bound) nuclides were processed at the 4 temperatures of 300 °K, 600 °K, 1000 °K and 2100 °K and most materials were processed with 6 to 8 values for the background cross section σ_0. These parameters are indicated in detail in TAB. 2.2, where it is possible to verify that nearly all materials were processed with the following values of σ_0: 1, 10, 1.0E+2, 1.0E+3, 1.0E+4 and 1.0E+10 barns.

With respect to the σ_0 values used in the generation of the Fe-56 processed files in the VITAMIN-B6 (AMPX format), VITJEF22.BOLIB /48/ (AMPX format) and MATJEF22.BOLIB /47/ (MATXS format) older libraries, an additional σ_0 numerical value equal to 0.01 barns was used in the production of the Fe-56 processed file included in VITENDF70.BOLIB as it was done for the VITJEFF31.BOLIB /44/ (AMPX format) and VITJEFF311.BOLIB /31/ (AMPX format) libraries and for the MATJEFF31.BOLIB /43/ (MATXS format) library. This additional σ_0 numerical value improves further, through a more precise self-shielding factor interpolation, the neutron self-shielding of the Fe-56 cross sections in natural iron. Moreover the possibility of a more accurate self-shielding calculation for Fe-56 was considered useful also in LWR radiation damage analyses in the carbon steel of the pressure vessel and in the stainless steel of the reactor internals (see /8/ and /30/).
For consistency with most other similar libraries, it was decided to use infinitely dilute background cross sections (σ_0 = 1.0E+10 barns) for nuclides with the atomic number Z less than 7, with the exception of B-11. Hence, only a background cross section with a numerical value of 1.0E+10 barns was used for each of these nuclides.

Thermal scattering cross sections were produced for six additional bound nuclides which were processed at all the temperatures (see TAB. 2.3) available in the ENDF/B-VII.0 thermal

scattering law data file (see /6/): H-1 in light water, H-1 in polyethylene, H-1 in zirconium hydride (not contained in the VITAMIN-B6, VITJEF22.BOLIB and MATJEF22.BOLIB libraries), H-2 in heavy water, C in graphite and Be in beryllium metal.

Finally, it is important to note that in total reactor power and heating calculations, the corresponding results can be heavily affected by the lack of gamma production data in some ENDF/B-VII.0 evaluated data files. Concerning this, it is recommended to verify carefully if the ENDF/B-VII.0 data files of the nuclides involved in the calculations include gamma production data (see TAB. 2.1).

TAB. 2.1

ENDF/B-VII.0 Nuclides Processed for the VITENDF70.BOLIB Library.

Z	Nuclide	ENDF/B-VII.0 MAT	AMPX Identifier	Gamma-Ray Production
1	H-H$_2$O	125/1	1001	YES
	H-CH$_2$	125/37	1901	YES
	H-ZrH	125/7	1401	YES
	D-D$_2$O	128/11	1002	YES
	H-3	131	1003	NO
2	He-3	225	2003	NO
	He-4	228	2004	NO
3	Li-6	325	3006	YES
	Li-7	328	3007	YES
4	Be-9	425	4009	YES
	Be-9 (Thermal)	425/26	4309	YES
5	B-10	525	5010	YES
	B-11	528	5011	YES
6	C-nat	600	6012	YES
	C-nat (Graphite)	600/31	6312	YES
7	N-14	725	7014	YES
	N-15	728	7015	YES
8	O-16	825	8016	YES
	O-17	828	8017	NO
9	F-19	925	9019	YES
11	Na-23	1125	11023	YES
12	Mg-24	1225	12024	YES
	Mg-25	1228	12025	YES
	Mg-26	1231	12026	YES
13	Al-27	1325	13027	YES
14	Si-28	1425	14028	YES
	Si-29	1428	14029	YES
	Si-30	1431	14030	YES
15	P-31	1525	15031	YES
16	S-32	1625	16032	YES
	S-33	1628	16033	YES
	S-34	1631	16034	YES
	S-36	1637	16036	YES
17	Cl-35	1725	17035	YES
	Cl-37	1731	17037	YES
19	K-39	1925	19039	YES
	K-40	1928	19040	YES
	K-41	1931	19041	YES
20	Ca-40	2025	20040	YES
	Ca-42	2031	20042	YES
	Ca-43	2034	20043	YES
	Ca-44	2037	20044	YES
	Ca-46	2043	20046	YES
	Ca-48	2049	20048	YES
22	Ti-46	2225	22046	YES
	Ti-47	2228	22047	YES
	Ti-48	2231	22048	YES
	Ti-49	2234	22049	YES
	Ti-50	2237	22050	YES

TAB. 2.1 Continued

ENDF/B-VII.0 Nuclides Processed for the VITENDF70.BOLIB Library.

Z	Nuclide	ENDF/B-VII.0 MAT	AMPX Identifier	Gamma-Ray Production
23	V-nat	2300	23000	YES
24	Cr-50	2425	24050	YES
	Cr-52	2431	24052	YES
	Cr-53	2434	24053	YES
	Cr-54	2437	24054	YES
25	Mn-55	2525	25055	YES
26	Fe-54	2625	26054	YES
	Fe-56	2631	26056	YES
	Fe-57	2634	26057	YES
	Fe-58	2637	26058	YES
27	Co-59	2725	27059	YES
28	Ni-58	2825	28058	YES
	Ni-60	2831	28060	YES
	Ni-61	2834	28061	YES
	Ni-62	2837	28062	YES
	Ni-64	2843	28064	YES
29	Cu-63	2925	29063	YES
	Cu-65	2931	29065	YES
31	Ga-69	3125	31069	NO
	Ga-71	3131	31071	NO
39	Y-89	3925	39089	YES
40	Zr-90	4025	40090	YES
	Zr-91	4028	40091	YES
	Zr-92	4031	40092	YES
	Zr-94	4037	40094	YES
	Zr-96	4043	40096	YES
41	Nb-93	4125	41093	YES
42	Mo-92	4225	42092	YES
	Mo-94	4231	42094	YES
	Mo-95	4234	42095	YES
	Mo-96	4237	42096	YES
	Mo-97	4240	42097	YES
	Mo-98	4243	42098	YES
	Mo-100	4249	42100	NO
47	Ag-107	4725	47107	YES
	Ag-109	4731	47109	YES
48	Cd-106	4825	48106	YES
	Cd-108	4831	48108	NO
	Cd-110	4837	48110	NO
	Cd-111	4840	48111	YES
	Cd-112	4843	48112	NO
	Cd-113	4846	48113	NO
	Cd-114	4849	48114	NO
	Cd-115m	4853	48115	YES
	Cd-116	4855	48116	NO
49	In-113	4925	49113	NO
	In-115	4931	49115	NO
50	Sn-112	5025	50112	NO
	Sn-114	5031	50114	NO

TAB. 2.1 Continued

ENDF/B-VII.0 Nuclides Processed for the VITENDF70.BOLIB Library.

Z	Nuclide	ENDF/B-VII.0 MAT	AMPX Identifier	Gamma-Ray Production
	Sn-115	5034	50115	NO
	Sn-116	5037	50116	NO
	Sn-117	5040	50117	NO
	Sn-118	5043	50118	NO
	Sn-119	5046	50119	NO
	Sn-120	5049	50120	NO
	Sn-122	5055	50122	NO
	Sn-123	5058	50123	NO
	Sn-124	5061	50124	NO
	Sn-125	5064	50125	YES
	Sn-126	5067	50126	NO
56	Ba-138	5649	56138	NO
63	Eu-151	6325	63151	NO
	Eu-152	6328	63152	NO
	Eu-153	6331	63153	YES
	Eu-154	6334	63154	NO
	Eu-155	6337	63155	NO
64	Gd-152	6425	64152	YES
	Gd-154	6431	64154	YES
	Gd-155	6434	64155	YES
	Gd-156	6437	64156	YES
	Gd-157	6440	64157	YES
	Gd-158	6443	64158	YES
	Gd-160	6449	64160	YES
68	Er-162	6825	68162	YES
	Er-164	6831	68164	YES
	Er-166	6837	68166	YES
	Er-167	6840	68167	YES
	Er-168	6843	68168	YES
	Er-170	6849	68170	YES
72	Hf-174	7225	72174	NO
	Hf-176	7231	72176	NO
	Hf-177	7234	72177	NO
	Hf-178	7237	72178	NO
	Hf-179	7240	72179	NO
	Hf-180	7243	72180	NO
73	Ta-181	7328	73181	YES
	Ta-182	7331	73182	NO
74	W-182	7431	74182	YES
	W-183	7434	74183	YES
	W-184	7437	74184	YES
	W-186	7443	74186	YES
75	Re-185	7525	75185	NO
	Re-187	7531	75187	NO
79	Au-197	7925	79197	YES
82	Pb-204	8225	82204	YES
	Pb-206	8231	82206	YES
	Pb-207	8234	82207	YES
	Pb-208	8237	82208	YES

TAB. 2.1 Continued

ENDF/B-VII.0 Nuclides Processed for the VITENDF70.BOLIB Library.

Z	Nuclide	ENDF/B-VII.0 MAT	AMPX Identifier	Gamma-Ray Production
83	Bi-209	8325	83209	YES
90	Th-230	9034	90230	NO
	Th-232	9040	90232	YES
91	Pa-231	9131	91231	YES
	Pa-233	9137	91233	YES
92	U-232	9219	92232	YES
	U-233	9222	92233	YES
	U-234	9225	92234	YES
	U-235	9228	92235	YES
	U-236	9231	92236	YES
	U-237	9234	92237	YES
	U-238	9237	92238	YES
93	Np-237	9346	93237	YES
	Np-238	9349	93238	NO
	Np-239	9352	93239	NO
94	Pu-236	9428	94236	NO
	Pu-237	9431	94237	NO
	Pu-238	9434	94238	NO
	Pu-239	9437	94239	YES
	Pu-240	9440	94240	YES
	Pu-241	9443	94241	YES
	Pu-242	9446	94242	YES
	Pu-243	9449	94243	YES
	Pu-244	9452	94244	NO
95	Am-241	9543	95241	YES
	Am-242	9546	95242	NO
	Am-242m	9547	95601	NO
	Am-243	9549	95243	YES
96	Cm-241	9628	96241	NO
	Cm-242	9631	96242	YES
	Cm-243	9634	96243	NO
	Cm-244	9637	96244	NO
	Cm-245	9640	96245	NO
	Cm-246	9643	96246	NO
	Cm-247	9646	96247	NO
	Cm-248	9649	96248	YES

TAB. 2.2

Background Cross Sections Values at which Bondarenko Factors Are Tabulated
in the VITENDF70.BOLIB Library.
All Nuclides Were Processed at Four Temperatures: 300 °K, 600 °K, 1000 °K and 2100 °K.

Nuclide	Background Cross Sections [barns]											Legendre Order
	1.E+10	1.E+6	1.E+5	1.E+4	1000.	300.	100.	50.	10.	1.	0.01	
H-1	1.E+10											7
H-2	1.E+10											7
H-3	1.E+10											7
He-3	1.E+10											7
He-4	1.E+10											7
Li-6	1.E+10											7
Li-7	1.E+10											7
Be-9	1.E+10											7
B-10	1.E+10											7
B-11	1.E+10				1000.		100.		10.	1.		7
C-nat	1.E+10											7
N-14	1.E+10				1000.		100.		10.	1.		7
N-15	1.E+10				1000.		100.		10.	1.		7
O-16	1.E+10				1000.		100.		10.	1.		7
O-17	1.E+10				1000.		100.		10.	1.		7
Fa-19	1.E+10				1000.		100.		10.	1.		7
Na-23	1.E+10				1000.	300.	100.	50.	10.	1.		7
Mg-24	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Mg-25	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Mg-26	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Al-27	1.E+10			1.E+4	1000.		100.	50.	10.	1.		7
Si-28	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Si-29	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Si-30	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
P-31	1.E+10			1.E+4	1000.		100.		10.	1.		7
S-32	1.E+10			1.E+4	1000.		100.		10.	1.		7
S-33	1.E+10			1.E+4	1000.		100.		10.	1.		7
S-34	1.E+10			1.E+4	1000.		100.		10.	1.		7
S-36	1.E+10			1.E+4	1000.		100.		10.	1.		7
Cl-35	1.E+10			1.E+4	1000.		100.		10.	1.		7
Cl-37	1.E+10			1.E+4	1000.		100.		10.	1.		7
K-39	1.E+10			1.E+4	1000.		100.		10.	1.		7
K-40	1.E+10			1.E+4	1000.		100.		10.	1.		7
K-41	1.E+10			1.E+4	1000.		100.		10.	1.		7
Ca-40	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Ca-42	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Ca-43	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Ca-44	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Ca-46	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Ca-48	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Ti-46	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Ti-47	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Ti-48	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Ti-49	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Ti-50	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
V-nat	1.E+10			1.E+4	1000.		100.		10.	1.		7
Cr-50	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Cr-52	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Cr-53	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Cr-54	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Mn-55	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Fe-54	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Fe-56	1.E+10		1.E+5	1.E+4	1000.		100.	50.	10.	1.	0.01	7
Fe-57	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Fe-58	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Co-59	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Ni-58	1.E+10		1.E+5	1.E+4	1000.		100.	50.	10.	1.		7
Ni-60	1.E+10		1.E+5	1.E+4	1000.		100.	50.	10.	1.		7
Ni-61	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7

TAB. 2.2 Continued

Background Cross Sections Values at which Bondarenko Factors Are Tabulated
in the VITENDF70.BOLIB Library.
All Nuclides Were Processed at Four Temperatures: 300 °K, 600 °K, 1000 °K and 2100 °K.

Nuclide	Background Cross Sections [barns]											Legendre
	1.E+10	1.E+6	1.E+5	1.E+4	1000.	300.	100.	50.	10.	1.	0.01	Order
Ni-62	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Ni-64	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Cu-63	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Cu-65	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		7
Ga-69	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Ga-71	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Y-89	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Zr-90	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Zr-91	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Zr-92	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Zr-94	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Zr-96	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Nb-93	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Mo-92	1.E+10		1.E+5	1.E+4	1000.		100.					5
Mo-94	1.E+10		1.E+5	1.E+4	1000.		100.					5
Mo-95	1.E+10		1.E+5	1.E+4	1000.		100.					5
Mo-96	1.E+10		1.E+5	1.E+4	1000.		100.					5
Mo-97	1.E+10		1.E+5	1.E+4	1000.		100.					5
Mo-98	1.E+10		1.E+5	1.E+4	1000.		100.					5
Mo-100	1.E+10		1.E+5	1.E+4	1000.		100.					5
Ag-107	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Ag-109	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Cd-106	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Cd-108	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Cd-110	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Cd-111	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Cd-112	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Cd-113	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Cd-114	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Cd-115m	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Cd-116	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
In-113	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
In-115	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Sn-112	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Sn-114	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Sn-115	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Sn-116	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Sn-117	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Sn-118	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Sn-119	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Sn-120	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Sn-122	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Sn-123	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Sn-124	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Sn-125	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Sn-126	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Ba-138	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Eu-151	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Eu-152	1.E+10	1.E+6	1.E+5	1.E+4	1000.		100.					5
Eu-153	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Eu-154	1.E+10	1.E+6	1.E+5	1.E+4	1000.		100.					5
Eu-155	1.E+10	1.E+6	1.E+5	1.E+4	1000.		100.					5
Gd-152	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Gd-154	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Gd-155	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Gd-156	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Gd-157	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Gd-158	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Gd-160	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5

TAB. 2.2 Continued

Background Cross Sections Values at which Bondarenko Factors Are Tabulated
in the VITENDF70.BOLIB Library.
All Nuclides Were Processed at Four Temperatures: 300 °K, 600 °K, 1000 °K and 2100 °K.

Nuclide	\multicolumn{11}{c}{Background Cross Sections [barns]}	Legendre Order
	1.E+10	1.E+6	1.E+5	1.E+4	1000.	300.	100.	50.	10.	1.	0.01	
Er-162	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Er-164	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Er-166	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Er-167	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Er-168	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Er-170	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Hf-174	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Hf-176	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Hf-177	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Hf-178	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Hf-179	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Hf-180	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Ta-181	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Ta-182	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
W-182	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
W-183	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
W-184	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
W-186	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Re-185	1.E+10	1.E+6	1.E+5	1.E+4	1000.		100.					5
Re-187	1.E+10	1.E+6	1.E+5	1.E+4	1000.		100.					5
Au-197	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Pb-204	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Pb-206	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Pb-207	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Pb-208	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Bi-209	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Th-230	1.E+10	1.E+6	1.E+5	1.E+4	1000.		100.					5
Th-232	1.E+10			1.E+4	1000.	300.	100.	50.	10.	1.		5
Pa-231	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Pa-233	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
U-232	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
U-233	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
U-234	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
U-235	1.E+10	1.E+6	1.E+5	1.E+4	1000.		100.	50.				5
U-236	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
U-237	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
U-238	1.E+10			1.E+4	1000.	300.	100.	50.	10.	1.		5
N-237	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Np-238	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Np-239	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Pu-236	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Pu-237	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Pu-238	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Pu-239	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Pu-240	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Pu-241	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Pu-242	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Pu-243	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Pu-244	1.E+10		1.E+5	1.E+4	1000.		100.		10.	1.		5
Am-241	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Am-242	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Am-242m	1.E+10		1.E+5	1.E+4	1000.							5
Am-243	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Cm-241	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Cm-242	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Cm-243	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Cm-244	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Cm-245	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Cm-246	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5

TAB. 2.2 Continued

Background Cross Sections Values at which Bondarenko Factors Are Tabulated
in the VITENDF70.BOLIB Library.
All Nuclides Were Processed at Four Temperatures: 300 °K, 600 °K, 1000 °K and 2100 °K.

Nuclide	Background Cross Sections [barns]											Legendre Order
	1.E+10	1.E+6	1.E+5	1.E+4	1000.	300.	100.	50.	10.	1.	0.01	
Cm-247	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5
Cm-248	1.E+10		1.E+5	1.E+4	1000.		100.	50.				5

TAB. 2.3

Processed Thermal Scattering Data in the VITENDF70.BOLIB Library.

Thermal scattering cross sections for the following bound nuclides were produced, through the THERMR module of NJOY, from the scattering matrices $S(\alpha,\beta)$ at various temperatures, included in the original ENDF/B-VII.0 thermal scattering law data file:

H-1 in H_2O (light water)
H-1 in CH_2 (polyethylene)
H-1 in ZrH   (zirconium hydride)
H-2 in D_2O (heavy water)
C   (graphite)
Be   (beryllium metal)

Nuclide	Temperature [°K]
H-1 in H_2O	293.6 350. 400. 450. 500. 550. 600. 650. 800.
H-1 in CH_2	296. 350.
H-1 in ZrH	296. 400. 500. 600. 700. 800. 1000. 1200.
H-2 in D_2O	293.6 350. 400. 450. 500. 550. 600. 650.
C	296. 400. 500. 600. 700. 800. 1000. 1200. 1600. 2000.
Be	296. 400. 500. 600. 700. 800. 1000. 1200.

2.3 - Energy Group Structure

The VITENDF70.BOLIB library adopts the same neutron and photon energy group structures as the VITAMIN-B6 /8/ library with 199 neutron energy groups (see TAB. 2.4) and 42 photon energy groups (see TAB. 2.5). The neutron and photon energy ranges are respectively included within 1.0E-05 eV and 1.9640E+07 eV for neutrons and within 1.0E+03 eV and 3.0E+07 eV for photons.

As reported in the VITAMIN-B6 library user's manual, this 199 neutron group structure was constructed as a compromise and improvement over the 174 neutron group structure used for the VITAMIN-E /59/ fine-group library, primarily conceived to treat fast neutron spectrum applications, and the 27 neutron group structure of the 27-neutron-group library, included in the SCALE /24/ system, suitable to treat criticality safety problems and out-of-core radiation shielding applications. The 27-neutron-group library has, in particular, a favourable neutron group discretization in the thermal neutron energy range whilst the resolution in the fast neutron energy range above 0.1 MeV results to be inadequate. Therefore the choice of the 199 group structure permits to treat not only fast neutron spectrum applications, through the proper neutron group structure at higher energies typical of VITAMIN-E, but also to treat physical systems with thermal neutron spectra, through the adequate group structure at lower energies of the 27-neutron-group library. Like the VITAMIN-B6 library, the VITENDF70.BOLIB thermal neutron energy range, i.e. the range of the neutron energy groups which include upscatter, contains 36 groups and has 5.043 eV as the uppermost boundary. In particular, the thermal neutron group energy limits are listed in TAB. 2.6. As underlined in the VITAMIN-B6 library user's manual, by combining the best features of the VITAMIN-E with the 27-group neutron energy grids, the best options were obtained for creating a problem-independent energy grid for a variety of reactor designs, including thermal (water or graphite-moderated) and fast reactor systems. Consequently, problem-dependent cross section libraries can be easily derived from VITENDF70.BOLIB, through the ENEA-Bologna 2007 Revision /19/ of the SCAMPI /18/ data processing system, without having to repeat the multi-group averaging directly from the ENDF/B-VII.0 /6/ data files.

The full VITENDF70.BOLIB library neutron energy group structure given in TAB. 2.4 is identical to the corresponding structure of the VITAMIN-B6 library. The 199 group energy limits are based on the 175 groups in VITAMIN-J /60/ (an OECD-NEA Data Bank library based on the VITAMIN-C /61/ and VITAMIN-E structures) with an expanded number of thermal groups as discussed above. At higher energies, the boundaries are almost identical with the earlier VITAMIN libraries, which consist of a basic 100-group-mesh of equal lethargy width plus numerous additional boundaries to resolve resonance minima that are important for radiation shielding calculations.

The full VITENDF70.BOLIB library photon energy group structure given in TAB. 2.5 is identical to the corresponding structure of the VITAMIN-B6 library. It is based on a combination of the 42 photon groups in the VITAMIN-J structure and the 18 group structure in the SCALE shielding library. The top energy group extends to 30 MeV, which allows proper representation of high energy gamma-rays from neutron capture at high energies. Although the cross section for capture at neutron energies between 20 and 30 MeV is small, such a reaction in some materials can produce gamma-rays with energies between 20 and 30 MeV, as reported in the VITAMIN-B6 library user's manual.

TAB. 2.4

Neutron Group Energy Boundaries for the VITENDF70.BOLIB Library.

Group	Upper Energy [eV]	Energy Width [eV]	Upper Lethargy	Lethargy Width
1	1.9640E+07	2.3080E+06	-6.7498E-01	0.1250
2	1.7332E+07	4.2700E+05	-5.4997E-01	0.0249
3	1.6905E+07	4.1800E+05	-5.2502E-01	0.0250
4	1.6487E+07	8.0400E+05	-4.9999E-01	0.0500
5	1.5683E+07	7.6500E+05	-4.4999E-01	0.0500
6	1.4918E+07	3.6800E+05	-3.9998E-01	0.0250
7	1.4550E+07	3.5900E+05	-3.7501E-01	0.0250
8	1.4191E+07	3.5100E+05	-3.5002E-01	0.0250
9	1.3840E+07	3.4100E+05	-3.2498E-01	0.0249
10	1.3499E+07	6.5900E+05	-3.0003E-01	0.0501
11	1.2840E+07	3.1700E+05	-2.4998E-01	0.0250
12	1.2523E+07	3.0900E+05	-2.2498E-01	0.0250
13	1.2214E+07	5.9600E+05	-2.0000E-01	0.0500
14	1.1618E+07	5.6600E+05	-1.4997E-01	0.0499
15	1.1052E+07	5.3900E+05	-1.0003E-01	0.0500
16	1.0513E+07	5.1300E+05	-5.0027E-02	0.0500
17	1.0000E+07	4.8770E+05	0.0000E+00	0.0500
18	9.5123E+06	4.6390E+05	4.9999E-02	0.0500
19	9.0484E+06	4.4130E+05	9.9997E-02	0.0500
20	8.6071E+06	4.1980E+05	1.5000E-01	0.0500
21	8.1873E+06	3.9930E+05	2.0000E-01	0.0500
22	7.7880E+06	3.7980E+05	2.5000E-01	0.0500
23	7.4082E+06	3.6130E+05	3.0000E-01	0.0500
24	7.0469E+06	3.4370E+05	3.5000E-01	0.0500
25	6.7032E+06	1.1080E+05	4.0000E-01	0.0167
26	6.5924E+06	2.1610E+05	4.1667E-01	0.0333
27	6.3763E+06	3.1100E+05	4.5000E-01	0.0500
28	6.0653E+06	2.9580E+05	5.0000E-01	0.0500
29	5.7695E+06	2.8140E+05	5.5000E-01	0.0500
30	5.4881E+06	2.6760E+05	6.0000E-01	0.0500
31	5.2205E+06	2.5460E+05	6.4999E-01	0.0500
32	4.9659E+06	2.4220E+05	6.9999E-01	0.0500
33	4.7237E+06	2.3040E+05	7.4999E-01	0.0500
34	4.4933E+06	4.2760E+05	8.0000E-01	0.1000
35	4.0657E+06	3.8690E+05	9.0000E-01	0.1000
36	3.6788E+06	3.5010E+05	1.0000E+00	0.1000
37	3.3287E+06	1.6230E+05	1.1000E+00	0.0500
38	3.1664E+06	1.5450E+05	1.1500E+00	0.0500
39	3.0119E+06	1.4680E+05	1.2000E+00	0.0500
40	2.8651E+06	1.3980E+05	1.2500E+00	0.0500
41	2.7253E+06	1.3290E+05	1.3000E+00	0.0500
42	2.5924E+06	1.2640E+05	1.3500E+00	0.0500
43	2.4660E+06	8.0800E+04	1.4000E+00	0.0333
44	2.3852E+06	1.9900E+04	1.4333E+00	0.0084
45	2.3653E+06	1.9600E+04	1.4417E+00	0.0083
46	2.3457E+06	3.8800E+04	1.4500E+00	0.0167
47	2.3069E+06	7.5600E+04	1.4667E+00	0.0333
48	2.2313E+06	1.0880E+05	1.5000E+00	0.0500
49	2.1225E+06	1.0350E+05	1.5500E+00	0.0500

TAB. 2.4 Continued

Neutron Group Energy Boundaries for the VITENDF70.BOLIB Library.

Group	Upper Energy [eV]	Energy Width [eV]	Upper Lethargy	Lethargy Width
50	2.0190E+06	9.8500E+04	1.6000E+00	0.0500
51	1.9205E+06	9.3700E+04	1.6500E+00	0.0500
52	1.8268E+06	8.9100E+04	1.7000E+00	0.0500
53	1.7377E+06	8.4700E+04	1.7500E+00	0.0500
54	1.6530E+06	8.0600E+04	1.8000E+00	0.0500
55	1.5724E+06	7.6700E+04	1.8500E+00	0.0500
56	1.4957E+06	7.3000E+04	1.9000E+00	0.0500
57	1.4227E+06	6.9300E+04	1.9500E+00	0.0500
58	1.3534E+06	6.6000E+04	2.0000E+00	0.0500
59	1.2874E+06	6.2800E+04	2.0500E+00	0.0500
60	1.2246E+06	5.9800E+04	2.1000E+00	0.0500
61	1.1648E+06	5.6800E+04	2.1500E+00	0.0500
62	1.1080E+06	1.0540E+05	2.2000E+00	0.1000
63	1.0026E+06	4.0960E+04	2.3000E+00	0.0417
64	9.6164E+05	5.4460E+04	2.3417E+00	0.0583
65	9.0718E+05	4.4240E+04	2.4000E+00	0.0500
66	8.6294E+05	4.2090E+04	2.4500E+00	0.0500
67	8.2085E+05	4.0030E+04	2.5000E+00	0.0500
68	7.8082E+05	3.8080E+04	2.5500E+00	0.0500
69	7.4274E+05	3.6230E+04	2.6000E+00	0.0500
70	7.0651E+05	3.4450E+04	2.6500E+00	0.0500
71	6.7206E+05	3.2780E+04	2.7000E+00	0.0500
72	6.3928E+05	3.1180E+04	2.7500E+00	0.0500
73	6.0810E+05	2.9660E+04	2.8000E+00	0.0500
74	5.7844E+05	2.8210E+04	2.8500E+00	0.0500
75	5.5023E+05	2.6830E+04	2.9000E+00	0.0500
76	5.2340E+05	2.5530E+04	2.9500E+00	0.0500
77	4.9787E+05	4.7380E+04	3.0000E+00	0.1000
78	4.5049E+05	4.2870E+04	3.1000E+00	0.1000
79	4.0762E+05	1.9880E+04	3.2000E+00	0.0500
80	3.8774E+05	1.8910E+04	3.2500E+00	0.0500
81	3.6883E+05	3.5100E+04	3.3000E+00	0.1000
82	3.3373E+05	3.1760E+04	3.4000E+00	0.1000
83	3.0197E+05	3.4800E+03	3.5000E+00	0.0116
84	2.9849E+05	1.2800E+03	3.5116E+00	0.0043
85	2.9721E+05	2.6900E+03	3.5159E+00	0.0091
86	2.9452E+05	7.2700E+03	3.5250E+00	0.0250
87	2.8725E+05	1.4010E+04	3.5500E+00	0.0500
88	2.7324E+05	2.6000E+04	3.6000E+00	0.1000
89	2.4724E+05	1.2060E+04	3.7000E+00	0.0500
90	2.3518E+05	1.1470E+04	3.7500E+00	0.0500
91	2.2371E+05	1.0910E+04	3.8000E+00	0.0500
92	2.1280E+05	1.0380E+04	3.8500E+00	0.0500
93	2.0242E+05	9.8700E+03	3.9000E+00	0.0500
94	1.9255E+05	9.3900E+03	3.9500E+00	0.0500
95	1.8316E+05	8.9400E+03	4.0000E+00	0.0500
96	1.7422E+05	8.4900E+03	4.0500E+00	0.0500
97	1.6573E+05	8.0900E+03	4.1000E+00	0.0500
98	1.5764E+05	7.6800E+03	4.1500E+00	0.0500

TAB. 2.4 Continued

Neutron Group Energy Boundaries for the VITENDF70.BOLIB Library.

Group	Upper Energy [eV]	Energy Width [eV]	Upper Lethargy	Lethargy Width
99	1.4996E+05	7.3200E+03	4.2000E+00	0.0500
100	1.4264E+05	6.9500E+03	4.2500E+00	0.0500
101	1.3569E+05	6.6200E+03	4.3000E+00	0.0500
102	1.2907E+05	6.3000E+03	4.3500E+00	0.0500
103	1.2277E+05	5.9800E+03	4.4000E+00	0.0500
104	1.1679E+05	5.7000E+03	4.4500E+00	0.0500
105	1.1109E+05	1.3053E+04	4.5000E+00	0.1250
106	9.8037E+04	1.1520E+04	4.6250E+00	0.1250
107	8.6517E+04	4.0140E+03	4.7500E+00	0.0475
108	8.2503E+04	3.0040E+03	4.7975E+00	0.0371
109	7.9499E+04	7.5010E+03	4.8346E+00	0.0991
110	7.1998E+04	4.6190E+03	4.9337E+00	0.0663
111	6.7379E+04	1.0817E+04	5.0000E+00	0.1750
112	5.6562E+04	4.0870E+03	5.1750E+00	0.0750
113	5.2475E+04	6.1660E+03	5.2500E+00	0.1250
114	4.6309E+04	5.4410E+03	5.3750E+00	0.1250
115	4.0868E+04	6.5610E+03	5.5000E+00	0.1750
116	3.4307E+04	2.4790E+03	5.6750E+00	0.0750
117	3.1828E+04	3.3270E+03	5.7500E+00	0.1104
118	2.8501E+04	1.5010E+03	5.8604E+00	0.0541
119	2.7000E+04	9.4200E+02	5.9145E+00	0.0355
120	2.6058E+04	1.2700E+03	5.9500E+00	0.0500
121	2.4788E+04	6.1200E+02	6.0000E+00	0.0250
122	2.4176E+04	5.9700E+02	6.0250E+00	0.0250
123	2.3579E+04	1.7040E+03	6.0500E+00	0.0750
124	2.1875E+04	2.5700E+03	6.1250E+00	0.1250
125	1.9305E+04	4.2710E+03	6.2500E+00	0.2500
126	1.5034E+04	3.3250E+03	6.5000E+00	0.2500
127	1.1709E+04	1.1140E+03	6.7500E+00	0.1000
128	1.0595E+04	1.4762E+03	6.8500E+00	0.1500
129	9.1188E+03	2.0171E+03	7.0000E+00	0.2500
130	7.1017E+03	1.5709E+03	7.2500E+00	0.2500
131	5.5308E+03	1.2234E+03	7.5000E+00	0.2500
132	4.3074E+03	6.0000E+02	7.7500E+00	0.1500
133	3.7074E+03	3.5280E+02	7.9000E+00	0.1000
134	3.3546E+03	3.1920E+02	8.0000E+00	0.1000
135	3.0354E+03	2.8890E+02	8.1000E+00	0.1000
136	2.7465E+03	1.3390E+02	8.2000E+00	0.0500
137	2.6126E+03	1.2740E+02	8.2500E+00	0.0500
138	2.4852E+03	2.3650E+02	8.3000E+00	0.1000
139	2.2487E+03	2.1400E+02	8.4000E+00	0.1000
140	2.0347E+03	4.5010E+02	8.5000E+00	0.2500
141	1.5846E+03	3.5050E+02	8.7500E+00	0.2500
142	1.2341E+03	2.7298E+02	9.0000E+00	0.2500
143	9.6112E+02	2.1260E+02	9.2500E+00	0.2500
144	7.4852E+02	1.6557E+02	9.5000E+00	0.2500
145	5.8295E+02	1.2895E+02	9.7500E+00	0.2500
146	4.5400E+02	1.0043E+02	1.0000E+01	0.2500
147	3.5357E+02	7.8210E+01	1.0250E+01	0.2500

TAB. 2.4 Continued

Neutron Group Energy Boundaries for the VITENDF70.BOLIB Library.

Group	Upper Energy [eV]	Energy Width [eV]	Upper Lethargy	Lethargy Width
148	2.7536E+02	6.0910E+01	1.0500E+01	0.2500
149	2.1445E+02	4.7430E+01	1.0750E+01	0.2500
150	1.6702E+02	3.6950E+01	1.1000E+01	0.2500
151	1.3007E+02	2.8770E+01	1.1250E+01	0.2500
152	1.0130E+02	2.2407E+01	1.1500E+01	0.2500
153	7.8893E+01	1.7451E+01	1.1750E+01	0.2500
154	6.1442E+01	1.3591E+01	1.2000E+01	0.2500
155	4.7851E+01	1.0585E+01	1.2250E+01	0.2500
156	3.7266E+01	8.2430E+00	1.2500E+01	0.2500
157	2.9023E+01	6.4200E+00	1.2750E+01	0.2500
158	2.2603E+01	4.9990E+00	1.3000E+01	0.2500
159	1.7604E+01	3.8940E+00	1.3250E+01	0.2500
160	1.3710E+01	3.0330E+00	1.3500E+01	0.2500
161	1.0677E+01	2.3617E+00	1.3750E+01	0.2500
162	8.3153E+00	1.8393E+00	1.4000E+01	0.2500
163	6.4760E+00	1.4325E+00	1.4250E+01	0.2500
164	5.0435E+00	1.1156E+00	1.4500E+01	0.2500
165	3.9279E+00	8.6890E-01	1.4750E+01	0.2500
166	3.0590E+00	6.7660E-01	1.5000E+01	0.2500
167	2.3824E+00	5.2700E-01	1.5250E+01	0.2500
168	1.8554E+00	4.1040E-01	1.5500E+01	0.2500
169	1.4450E+00	1.4500E-01	1.5750E+01	0.1060
170	1.3000E+00	1.7470E-01	1.5856E+01	0.1440
171	1.1253E+00	4.5300E-02	1.6000E+01	0.0410
172	1.0800E+00	4.0000E-02	1.6041E+01	0.0380
173	1.0400E+00	4.0000E-02	1.6079E+01	0.0390
174	1.0000E+00	1.2357E-01	1.6118E+01	0.1320
175	8.7643E-01	7.6430E-02	1.6250E+01	0.0910
176	8.0000E-01	1.1744E-01	1.6341E+01	0.1590
177	6.8256E-01	5.7500E-02	1.6500E+01	0.0880
178	6.2506E-01	9.3480E-02	1.6588E+01	0.1620
179	5.3158E-01	3.1580E-02	1.6750E+01	0.0610
180	5.0000E-01	8.6010E-02	1.6811E+01	0.1890
181	4.1399E-01	4.7190E-02	1.7000E+01	0.1210
182	3.6680E-01	4.1800E-02	1.7121E+01	0.1210
183	3.2500E-01	5.0000E-02	1.7242E+01	0.1670
184	2.7500E-01	5.0000E-02	1.7409E+01	0.2010
185	2.2500E-01	4.1000E-02	1.7610E+01	0.2010
186	1.8400E-01	3.4000E-02	1.7811E+01	0.2040
187	1.5000E-01	2.5000E-02	1.8015E+01	0.1830
188	1.2500E-01	2.5000E-02	1.8198E+01	0.2230
189	1.0000E-01	3.0000E-02	1.8421E+01	0.3560
190	7.0000E-02	2.0000E-02	1.8777E+01	0.3370
191	5.0000E-02	1.0000E-02	1.9114E+01	0.2230
192	4.0000E-02	1.0000E-02	1.9337E+01	0.2880
193	3.0000E-02	9.0000E-03	1.9625E+01	0.3560
194	2.1000E-02	6.5000E-03	1.9981E+01	0.3710
195	1.4500E-02	4.5000E-03	2.0352E+01	0.3710
196	1.0000E-02	5.0000E-03	2.0723E+01	0.6930

TAB. 2.4 Continued

Neutron Group Energy Boundaries for the VITENDF70.BOLIB Library.

Group	Upper Energy [eV]	Energy Width [eV]	Upper Lethargy	Lethargy Width
197	5.0000E-03	3.0000E-03	2.1416E+01	0.9170
198	2.0000E-03	1.5000E-03	2.2333E+01	1.3860
199	5.0000E-04	4.9000E-04	2.3719E+01	3.9120
	Lower Energy		Lower Lethargy	
	1.0000E-05		2.7631E+01	

TAB. 2.5

Photon Group Energy Boundaries for the VITENDF70.BOLIB Library.

Group	Upper Energy [eV]	Energy Width [eV]	Upper Lethargy	Lethargy Width
1	3.0000E+07	1.0000E+07	-1.0986E+00	0.4055
2	2.0000E+07	6.0000E+06	-6.9315E-01	0.3567
3	1.4000E+07	2.0000E+06	-3.3647E-01	0.1542
4	1.2000E+07	2.0000E+06	-1.8232E-01	0.1823
5	1.0000E+07	2.0000E+06	0.0000E+00	0.2231
6	8.0000E+06	5.0000E+05	2.2314E-01	0.0645
7	7.5000E+06	5.0000E+05	2.8768E-01	0.0690
8	7.0000E+06	5.0000E+05	3.5667E-01	0.0741
9	6.5000E+06	5.0000E+05	4.3078E-01	0.0800
10	6.0000E+06	5.0000E+05	5.1083E-01	0.0870
11	5.5000E+06	5.0000E+05	5.9784E-01	0.0953
12	5.0000E+06	5.0000E+05	6.9315E-01	0.1054
13	4.5000E+06	5.0000E+05	7.9851E-01	0.1178
14	4.0000E+06	5.0000E+05	9.1629E-01	0.1335
15	3.5000E+06	5.0000E+05	1.0498E+00	0.1542
16	3.0000E+06	5.0000E+05	1.2040E+00	0.1823
17	2.5000E+06	5.0000E+05	1.3863E+00	0.2231
18	2.0000E+06	3.4000E+05	1.6094E+00	0.1863
19	1.6600E+06	1.6000E+05	1.7958E+00	0.1014
20	1.5000E+06	1.6000E+05	1.8971E+00	0.1128
21	1.3400E+06	1.0000E+04	2.0099E+00	0.0075
22	1.3300E+06	3.3000E+05	2.0174E+00	0.2852
23	1.0000E+06	2.0000E+05	2.3026E+00	0.2231
24	8.0000E+05	1.0000E+05	2.5257E+00	0.1335
25	7.0000E+05	1.0000E+05	2.6593E+00	0.1542
26	6.0000E+05	8.8000E+04	2.8134E+00	0.1586
27	5.1200E+05	2.0000E+03	2.9720E+00	0.0039
28	5.1000E+05	6.0000E+04	2.9759E+00	0.1252
29	4.5000E+05	5.0000E+04	3.1011E+00	0.1178
30	4.0000E+05	1.0000E+05	3.2189E+00	0.2877
31	3.0000E+05	1.0000E+05	3.5066E+00	0.4055
32	2.0000E+05	5.0000E+04	3.9120E+00	0.2877
33	1.5000E+05	5.0000E+04	4.1997E+00	0.4055
34	1.0000E+05	2.5000E+04	4.6052E+00	0.2877
35	7.5000E+04	5.0000E+03	4.8929E+00	0.0690
36	7.0000E+04	1.0000E+04	4.9618E+00	0.1542
37	6.0000E+04	1.5000E+04	5.1160E+00	0.2877
38	4.5000E+04	5.0000E+03	5.4037E+00	0.1178
39	4.0000E+04	1.0000E+04	5.5215E+00	0.2877
40	3.0000E+04	1.0000E+04	5.8091E+00	0.4055
41	2.0000E+04	1.0000E+04	6.2146E+00	0.6931
42	1.0000E+04	9.0000E+03	6.9078E+00	2.3026
	Lower Energy		Lower Lethargy	
	1.0000E+03		9.2103E+00	

TAB. 2.6

VITENDF70.BOLIB Library Thermal Neutron Energy Range.

Group	Upper Energy [eV]	Lethargy Width		Group	Upper Energy [eV]	Lethargy Width
164	5.04350	0.250		182	0.36680	0.121
165	3.92790	0.250		183	0.32500	0.167
166	3.05900	0.250		184	0.27500	0.201
167	2.38240	0.250		185	0.22500	0.201
168	1.85540	0.250		186	0.18400	0.204
169	1.44500	0.106		187	0.15000	0.183
170	1.30000	0.144		188	0.12500	0.223
171	1.12530	0.041		189	0.10000	0.356
172	1.08000	0.038		190	0.07000	0.337
173	1.04000	0.039		191	0.05000	0.223
174	1.00000	0.132		192	0.04000	0.288
175	0.87643	0.091		193	0.03000	0.356
176	0.80000	0.159		194	0.02100	0.371
177	0.68256	0.088		195	0.01450	0.371
178	0.62506	0.162		196	0.01000	0.693
179	0.53158	0.061		197	0.00500	0.917
180	0.50000	0.189		198	0.00200	1.386
181	0.41399	0.121		199	0.00050	3.912
					Lower Energy 0.00001	

2.4 - Weighting Function

The neutron and photon weighting functions used to produce the VITENDF70.BOLIB /5/ library cross sections are the same as those employed in the generation of the VITAMIN-B6 /8/ cross sections.

The neutron weighting function is of the form typically chosen for fission reactor shielding problems, i.e. it consists of a smoothly varying combination of a Maxwellian thermal spectrum, a fission spectrum, and a "l/E" slowing down spectrum. This corresponds to the IWT=4 option in the GROUPR module of the NJOY /17/ system. The breakpoint energies for the 3-region spectrum are similar to those used in VITAMIN-C /61/. The breakpoint energy between the Maxwellian and l/E shapes is 0.125 eV. The fission temperature has been adjusted to better reflect the neutron spectrum in a thermal reactor ($\theta = 1.273$ MeV). The use of a large number of energy groups should make the exact functional form and energy break points less important compared to generating a broad-group library directly from ENDF/B data. The functional form of the weighting spectrum is given by the following:

Functional Form	Energy Limits	Groups
1. Maxwellian Thermal Spectrum (kT = 0.025 eV)		
$W_1(E) = C_1 E e^{-E/kT}$	1.0E-5 eV to 0.125 eV	188-199
2. "l/E" Slowing-Down Spectrum		
$W_2(E) = C_2/E$	0.125 eV to 820.8 keV	67-187
3. Fission Spectrum ($\theta = 1.273$ MeV)		
$W_3(E) = C_3 E^{1/2} e^{-E/\theta}$	820.8 keV to 20 MeV	1-66

A continuous weighting spectrum is achieved with the following constants: $C_1 = 9498.4$ eV^{-2}, $C_2 = 1.0$ and $C_3 = 2.5625$ MeV$^{-1.5}$. The neutron weighting function is shown in FIG. 2.1 and listed in TAB. 2.7 in a 199 group representation.

The photon weighting spectrum consists of a l/E spectrum with a "roll-off" of the spectrum at lower energies to represent photoelectric absorption and a similar drop-off of the spectrum at higher energies corresponding to the Q-value for neutron capture. This corresponds to the IWT=3 input option in the GAMINR module of the NJOY system. The photon weighting function is shown in FIG. 2.2 and listed in TAB. 2.8 in a 42 group representation.
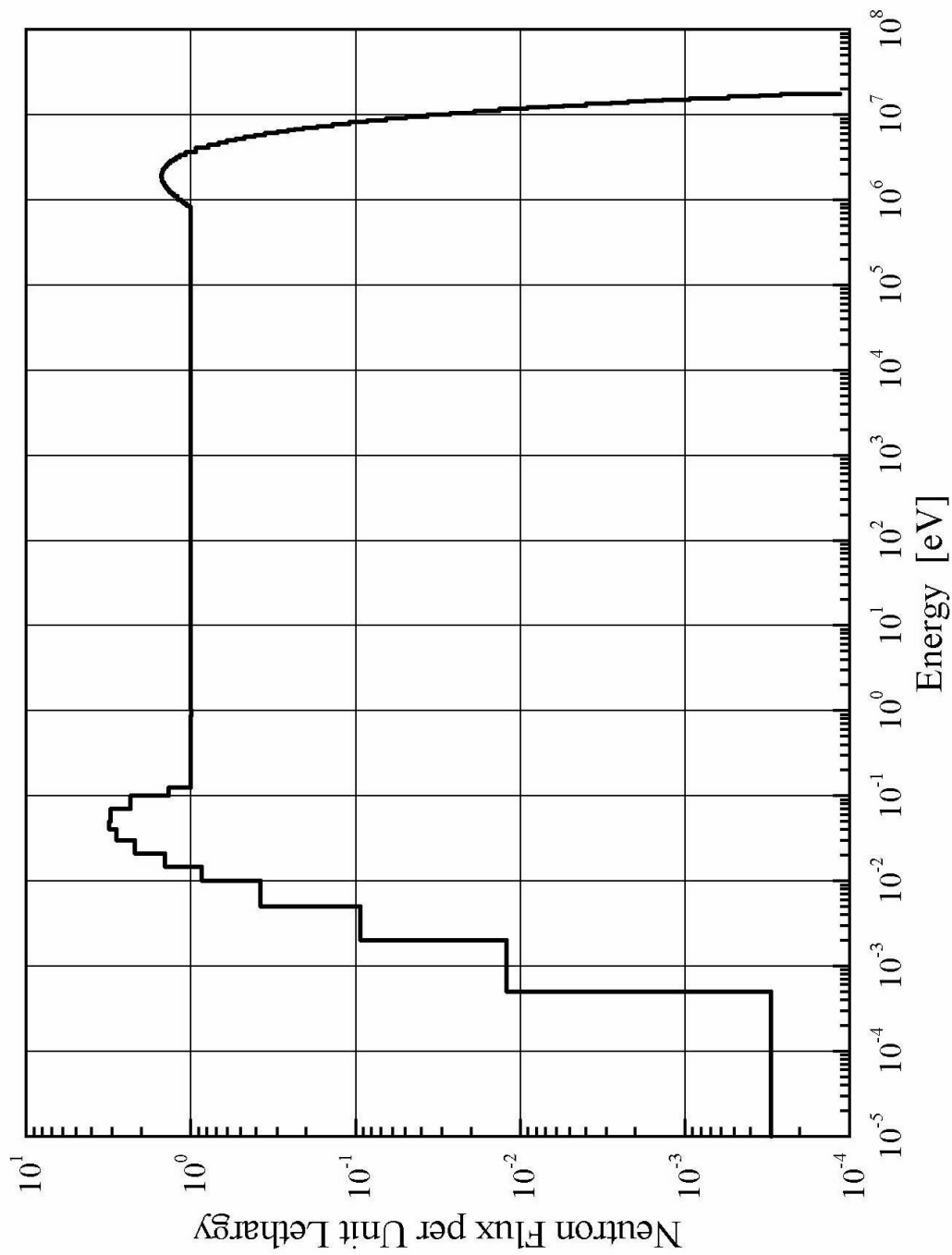
FIG. 2.1 199 Group Representation of Standard Weighting Spectrum Used to Create VITENDF70.BOLIB Neutron Cross Sections from ENDF/B-VII.0 Point-Wise Data.

TAB. 2.7

Neutron Energy Weighting Spectrum for the VITENDF70.BOLIB Library.

Group	Weight	Group	Weight	Group	Weight
1	1.423E-05	51	7.543E-02	101	5.002E-02
2	6.578E-06	52	7.516E-02	102	5.004E-02
3	8.861E-06	53	7.461E-02	103	4.994E-02
4	2.733E-05	54	7.389E-02	104	5.004E-02
5	4.690E-05	55	7.295E-02	105	1.250E-01
6	3.419E-05	56	7.181E-02	106	1.250E-01
7	4.380E-05	57	7.031E-02	107	4.751E-02
8	5.590E-05	58	6.888E-02	108	3.709E-02
9	7.036E-05	59	6.723E-02	109	9.911E-02
10	1.991E-04	60	6.552E-02	110	6.631E-02
11	1.368E-04	61	6.354E-02	111	1.750E-01
12	1.684E-04	62	1.211E-01	112	7.500E-02
13	4.576E-04	63	4.809E-02	113	1.250E-01
14	6.684E-04	64	6.474E-02	114	1.250E-01
15	9.574E-04	65	5.321E-02	115	1.750E-01
16	1.342E-03	66	5.108E-02	116	7.500E-02
17	1.843E-03	67	5.000E-02	117	1.104E-01
18	2 482E-03	68	5.000E-02	118	5.410E-02
19	3.285E-03	69	5.001E-02	119	3.551E-02
20	4.272E-03	70	4.999E-02	120	4.997E-02
21	5.465E-03	71	5.001E-02	121	2.500E-02
22	6.882E-03	72	5.000E-02	122	2.500E-02
23	8.540E-03	73	5.000E-02	123	7.501E-02
24	1.045E-02	74	5.000E-02	124	1.250E-01
25	3.951E-03	75	4.999E-02	125	2.501E-01
26	8.659E-03	76	5.001E-02	126	2.500E-01
27	1.503E-02	77	1.000E-01	127	9.998E-02
28	1.769E-02	78	1.000E-01	128	1.500E-01
29	2.058E-02	79	5.000E-02	129	2.500E-01
30	2.368E-02	80	5.000E-02	130	2.500E-01
31	2.697E-02	81	1.000E-01	131	2.500E-01
32	3.041E-02	82	1.000E-01	132	1.500E-01
33	3.397E-02	83	1.159E-02	133	1.000E-01
34	7.885E-02	84	4.297E-03	134	9.999E-02
35	9.338E-02	85	9.092E-03	135	1.000E-01
36	1.073E-01	86	2.499E-02	136	4.998E-02
37	5.848E-02	87	5.000E-02	137	4.999E-02
38	6.148E-02	88	9.999E-02	138	1.000E-01
39	6.413E-02	89	5.001E-02	139	1.000E-01
40	6.666E-02	90	5.000E-02	140	2.500E-01
41	6.879E-02	91	5.000E-02	141	2.500E-01
42	7.065E-02	92	5.001E-02	142	2.500E-01
43	4.797E-02	93	4.999E-02	143	2.500E-01
44	1.216E-02	94	5.000E-02	144	2.500E-01
45	1.212E-02	95	5.004E-02	145	2.500E-01
46	2.439E-02	96	4.996E-02	146	2.500E-01
47	4.909E-02	97	5.005E-02	147	2.500E-01
48	7.440E-02	98	4.995E-02	148	2.500E-01
49	7.503E-02	99	5.004E-02	149	2.500E-01
50	7.539E-02	100	4.995E-02	150	2.500E-01

TAB. 2.7 Continued

Neutron Energy Weighting Spectrum for the VITENDF70.BOLIB Library.

Group	Weight	Group	Weight	Group	Weight
151	2.500E-01	168	2.500E-01	185	2.012E-01
152	2.500E-01	169	1.057E-01	186	2.043E-01
153	2.500E-01	170	1.443E-01	187	1.823E-01
154	2.500E-01	171	4.109E-02	188	3.037E-01
155	2.500E-01	172	3.774E-02	189	8.282E-01
156	2.500E-01	173	3.922E-02	190	1.038E+00
157	2.500E-01	174	1.319E-01	191	7.060E-01
158	2.500E-01	175	9.125E-02	192	8.174E-01
159	2.500E-01	176	1.588E-01	193	7.820E-01
160	2.500E-01	177	8.800E-02	194	5.360E-01
161	2.500E-01	178	1.620E-01	195	3.194E-01
162	2.500E-01	179	6.125E-02	196	2.614E-01
163	2.500E-01	180	1.888E-01	197	8.601E-02
164	2.500E-01	181	1.210E-01	198	1.684E-02
165	2.500E-01	182	1.210E-01	199	1.171E-03
166	2.500E-01	183	1.671E-01		
167	2.500E-01	184	2.007E-01		
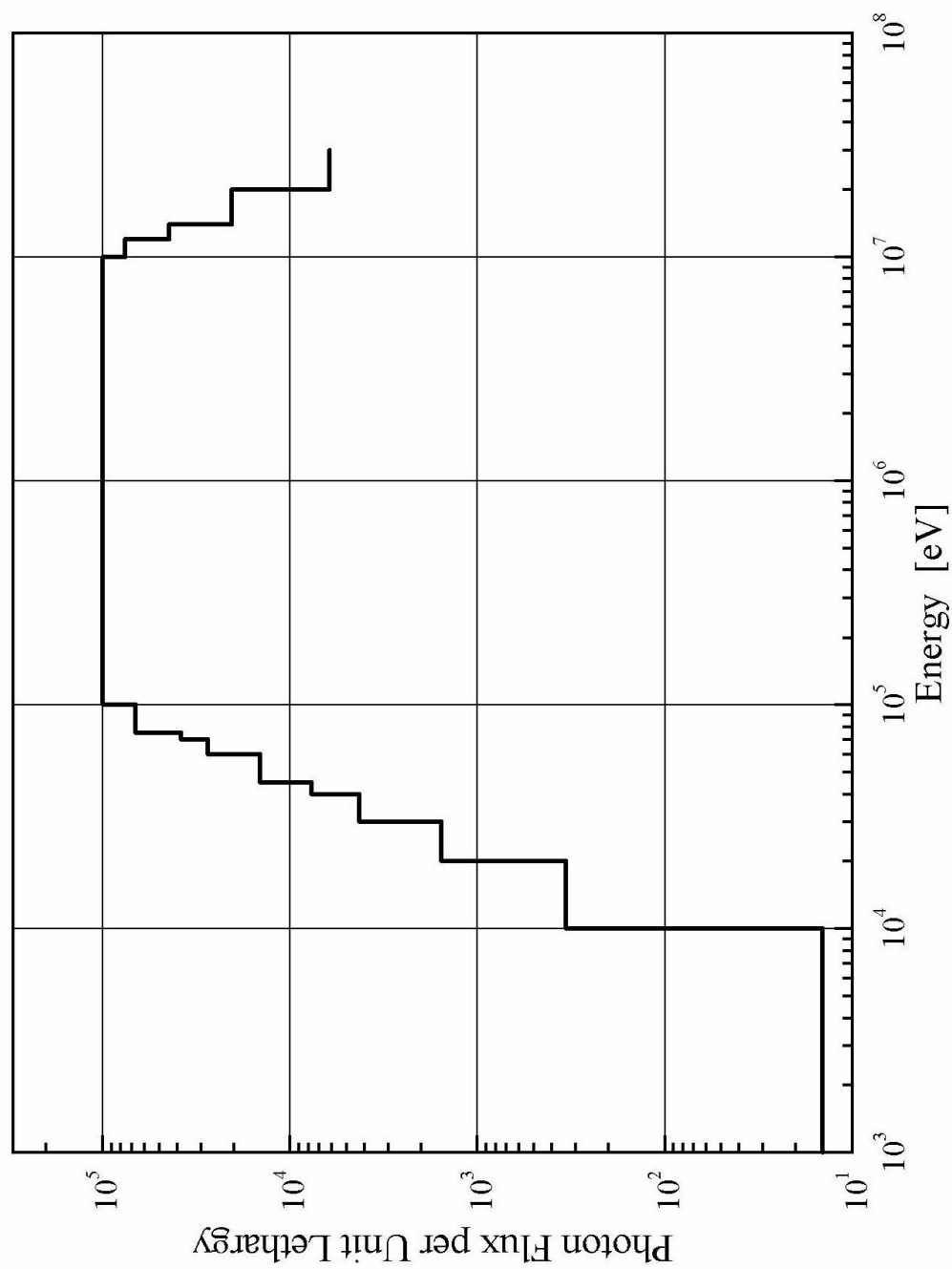
FIG. 2.2 42 Group Representation of Standard Weighting Spectrum Used to Create VITENDF70.BOLIB Photon Cross Sections from ENDF/B-VII.0 Point-Wise Data.

TAB. 2.8

Photon Energy Weighting Spectrum for the VITENDF70.BOLIB Library.

Group	Weight	Group	Weight	Group	Weight
1	2.498E+03	15	1.542E+04	29	1.178E+04
2	7.298E+03	16	1.824E+04	30	2.877E+04
3	6.824E+03	17	2.232E+04	31	4.055E+04
4	1.387E+04	18	1.864E+04	32	2.877E+04
5	2.232E+04	19	1.014E+04	33	4.055E+04
6	6.455E+03	20	1.128E+04	34	1.927E+04
7	6.901E+03	21	7.491E+02	35	2.629E+03
8	7.413E+03	22	2.852E+04	36	4.233E+03
9	8.006E+03	23	2.232E+04	37	4.163E+03
10	8.703E+03	24	1.335E+04	38	9.042E+02
11	9.534E+03	25	1.542E+04	39	1.233E+03
12	1.054E+04	26	1.586E+04	40	6.333E+02
13	1.178E+04	27	3.914E+02	41	2.333E+02
14	1.336E+04	28	1.252E+04	42	3.330E+01

2.5 - Legendre Order of Scattering

The order of scattering used for both neutrons and photons is P_7, for nuclides with Z=l through Z=29 (copper) and P_5 for the remainder of the nuclides. In particular, the previous values corresponding to L=ℓ-max, the maximum order of the Legendre polynomial (P_ℓ) expansion of the scattering cross section matrix, available for each nuclide of the library, are listed in TAB. 2.2.
Most calculations are likely to be done with P_3 scattering, but for some problems, e.g., when single scatter events dominate, higher orders may be required as stated in the VITAMIN-B6 /8/ library user's manual. For the same nuclide, an identical order of scattering for both neutrons and photons was adopted in the generation of the VITAMIN-B6, VITENDF70.BOLIB /5/, VITJEFF311.BOLIB /31/, MATJEFF31.BOLIB /43/, VITJEFF31.BOLIB /44/, MATJEF22.BOLIB /47/ and VITJEF22.BOLIB /48/ cross sections.

2.6 - Convergence Parameters

The following numerical values of the fractional error tolerances were chosen as input parameters in NJOY /17/ to generate the libraries VITENDF70.BOLIB /5/, VITJEFF311.BOLIB /31/, MATJEFF31.BOLIB /43/ and VITJEFF31.BOLIB /44/: 0.1% for resolved resonance reconstruction and for linearization in RECONR and 0.1% for thinning in BROADR. These data have reduced numerical values with respect to the 0.2% corresponding error tolerance values used to generate VITAMIN-B6 /8/, MATJEF22.BOLIB /47/ and VITJEF22.BOLIB /48/, chosen on the basis of the ORNL experience with the VITAMIN libraries and the experience of the NJOY users.

2.7 - Processing Codes and Procedures

The NJOY-99.259 /17/ nuclear data processing system and the ENEA-Bologna 2007 Revision /19/ of the SCAMPI /18/ nuclear data processing system were used on a Personal Computer (CPU INTEL Pentium III, 448 MB of RAM; FSF-g77 version 0.5.26 FORTRAN compiler for NJOY and f77 Absoft version 5.0 FORTRAN 77 compiler for SCAMPI) with the Linux Red Hat 7.1 operating system to generate the VITENDF70.BOLIB /5/ library. The following modules of NJOY were used to process neutron interaction (n-n), photon production (n-γ) and photon interaction (γ-γ) data into the GENDF format, from the ENDF/B-VII.0 incident neutron and photo-atomic data in ENDF-6 /56/ format. Specifically, the MODER, RECONR, BROADR, THERMR, HEATR, GASPR, PURR, GROUPR modules were used for the incident neutron data while the MODER, RECONR and GAMINR modules were used for the photo-atomic data. Then the ENEA-Bologna revised version of the SMILER module, contained in the ENEA-Bologna 2007 Revision of SCAMPI, was used to translate the fine-group data from the GENDF format into the AMPX master library format of VITENDF70.BOLIB. The RADE module of SCAMPI was used to check and screen the data for internal consistency and "sanity", i.e. the data values are physical and within expected bounds. Then the module AIM of SCAMPI was used to convert the master cross section libraries for the standard and bound nuclide cross section files from binary to BCD format.
A brief description of the function of the NJOY and SCAMPI modules is presented in TAB. 2.9. A schematic diagram illustrating the VITENDF70.BOLIB processing procedure to produce the standard nuclide cross section files is given in FIG. 2.3, while the procedure to produce the bound nuclide cross section files is reported in FIG. 2.4.

TAB. 2.9

Modules from the NJOY-99.259 and SCAMPI
Nuclear Data Processing Systems Used to Process VITENDF70.BOLIB.

NJOY-99.259 System

Module	Function
MODER	Converts between ENDF/B standard coded mode and the NJOY blocked binary mode.
RECONR	Reconstructs point-wise cross sections from ENDF/B resonance parameters and interpolation schemes.
BROADR	Doppler-broadens and thins point-wise cross sections.
THERMR	Produces cross sections and energy-to-energy matrices for free or bound scatterers in the thermal energy range.
HEATR	Generates heat production cross sections (KERMA factors) and damage-energy production.
GASPR	Computes gas production cross sections.
PURR	Computes probability tables and effective point-wise self-shielded cross sections in the unresolved energy range.
GROUPR	Generates self-shielded multi-group cross sections and group-to-group scattering and photon production matrices in GENDF format.
GAMINR	Computes multi-group photo-atomic cross sections, KERMA factors, group-to-group photon scattering matrices.

SCAMPI System (ENEA-Bologna 2007 Revision)

Module	Function
SMILER	Translates GENDF files produced by NJOY into AMPX master interface format.
RADE	Performs sanity and consistency tests on multi-group libraries.
AIM	Converts master cross section libraries from binary format to BCD (or vice-versa).

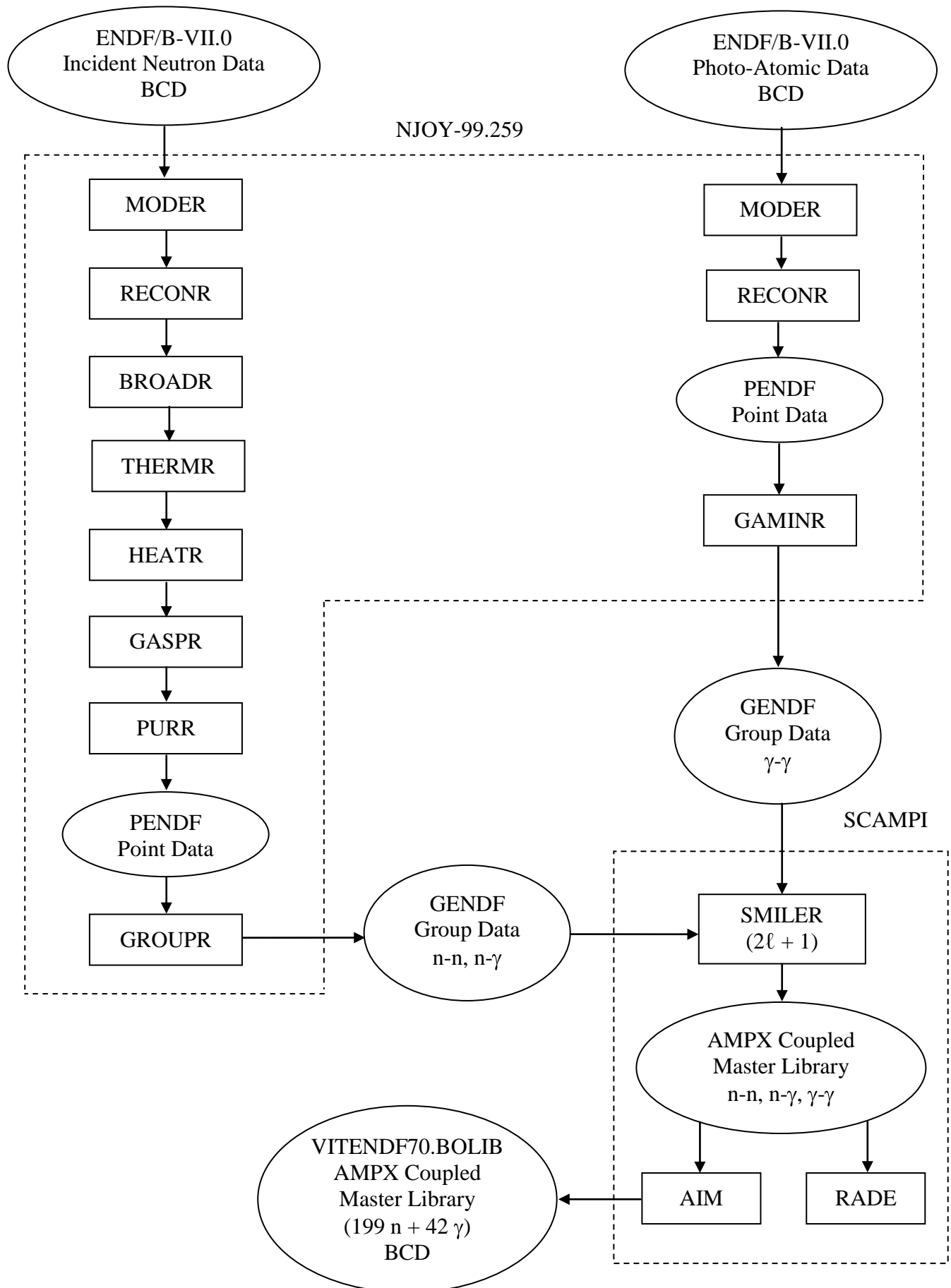FIG. 2.3 Procedure for Generating the VITENDF70.BOLIB Library in AMPX Format from ENDF/B-VII.0.

FIG. 2.4 Procedure for Generating the VITENDF70.BOLIB Bound Nuclides in AMPX Format from ENDF/B-VII.0.

2.8 - Response Functions

At present only the following "response" functions are included in tabulated form in the VITENDF70.BOLIB /5/ library package: neutron and photon group energy boundaries, neutron and photon group energy widths, neutron and photon group lethargy boundaries, neutron and photon group lethargy widths, total (prompt + delayed) neutron fission spectra (χ) for the U-235, U-238 and Pu-239 nuclides.

The total neutron fission spectra (χ) were obtained through the ENEA-Bologna 2007 Revision /19/ of the SCAMPI /18/ system which allows processing of delayed neutron spectra. In particular the BONAMI and NITAWL modules were firstly used to generate an AMPX working file for each fissile nuclide, containing infinitely dilute cross sections (background cross section $\sigma_0 = 1.0E+10$ barns) at the temperature of 300 °K. Then the fine-group χ-vectors were calculated by the ICE module with the KOPT=4 option (nuclide-dependent spectrum option) in the 1$$ array, i.e. using the neutron flux spectrum derived from the weighting functions corresponding to the IWT=4 option in GROUPR (see 2.4), always provided in each nuclide data file of an AMPX working library.

Then the normalization of the total neutron fission spectra (χ) to one neutron per fission was performed since the ICE module with the KOPT=4 option does not ensure a proper normalization (see /18/).

The resulting total neutron fission spectra for the U-235, U-238 and Pu-239 nuclides are reported in TAB. 2.10.

A 199 group representation of the U-235, U-238 and Pu-239 total neutron fission spectra are shown in FIG. 2.5.

TAB. 2.10

VITENDF70.BOLIB Library Total Neutron Fission Spectra for U-235, U-238 and Pu-239.

Group	Upper Energy [eV]	U-235 X	U-238 X	Pu-239 X
1	1.9640E+07	4.48487E-06	3.96812E-06	8.08940E-06
2	1.7332E+07	1.78036E-06	1.61066E-06	3.04165E-06
3	1.6905E+07	2.53075E-06	2.20345E-06	4.24039E-06
4	1.6487E+07	7.81808E-06	6.95344E-06	1.28221E-05
5	1.5683E+07	1.38041E-05	1.22533E-05	2.20075E-05
6	1.4918E+07	1.04142E-05	9.08987E-06	1.62518E-05
7	1.4550E+07	1.34914E-05	1.17857E-05	2.07938E-05
8	1.4191E+07	1.67525E-05	1.52045E-05	2.55743E-05
9	1.3840E+07	2.22662E-05	1.93326E-05	3.34426E-05
10	1.3499E+07	6.20470E-05	5.55362E-05	9.17669E-05
11	1.2840E+07	4.44299E-05	3.86555E-05	6.44415E-05
12	1.2523E+07	5.48109E-05	4.79983E-05	7.86931E-05
13	1.2214E+07	1.47656E-04	1.32016E-04	2.08738E-04
14	1.1618E+07	2.20844E-04	1.95609E-04	3.05587E-04
15	1.1052E+07	3.19935E-04	2.83836E-04	4.33535E-04
16	1.0513E+07	4.49962E-04	4.02938E-04	5.98666E-04
17	1.0000E+07	5.99349E-04	5.58856E-04	7.85270E-04
18	9.5123E+06	8.15877E-04	7.61601E-04	1.04929E-03
19	9.0484E+06	1.09028E-03	1.01902E-03	1.37736E-03
20	8.6071E+06	1.43116E-03	1.33946E-03	1.77749E-03
21	8.1873E+06	1.84627E-03	1.73067E-03	2.25656E-03
22	7.7880E+06	2.34208E-03	2.19927E-03	2.82002E-03
23	7.4082E+06	2.92318E-03	2.75017E-03	3.47132E-03
24	7.0469E+06	3.59131E-03	3.38571E-03	4.21079E-03
25	6.7032E+06	1.36042E-03	1.28433E-03	1.58274E-03
26	6.5924E+06	2.98418E-03	2.82034E-03	3.45219E-03
27	6.3763E+06	5.18167E-03	4.90604E-03	5.94099E-03
28	6.0653E+06	6.09432E-03	5.78236E-03	6.91861E-03
29	5.7695E+06	7.07797E-03	6.72959E-03	7.96166E-03
30	5.4881E+06	8.12031E-03	7.73577E-03	9.05525E-03
31	5.2205E+06	9.21823E-03	8.79774E-03	1.01951E-02
32	4.9659E+06	1.03591E-02	9.90336E-03	1.13661E-02
33	4.7237E+06	1.15317E-02	1.10417E-02	1.25555E-02
34	4.4933E+06	2.66534E-02	2.55783E-02	2.86918E-02
35	4.0657E+06	3.14418E-02	3.02593E-02	3.33648E-02
36	3.6788E+06	3.60420E-02	3.47873E-02	3.77335E-02
37	3.3287E+06	1.96141E-02	1.89743E-02	2.03429E-02
38	3.1664E+06	2.06016E-02	1.99625E-02	2.12414E-02
39	3.0119E+06	2.14605E-02	2.08363E-02	2.20036E-02
40	2.8651E+06	2.22674E-02	2.16613E-02	2.27134E-02
41	2.7253E+06	2.29212E-02	2.23406E-02	2.32705E-02
42	2.5924E+06	2.34670E-02	2.29205E-02	2.37220E-02
43	2.4660E+06	1.58833E-02	1.55410E-02	1.60031E-02
44	2.3852E+06	4.01958E-03	3.93646E-03	4.04369E-03
45	2.3653E+06	4.00129E-03	3.92024E-03	4.02276E-03
46	2.3457E+06	8.04429E-03	7.88599E-03	8.08050E-03
47	2.3069E+06	1.61487E-02	1.58471E-02	1.61953E-02
48	2.2313E+06	2.43563E-02	2.39476E-02	2.43645E-02
49	2.1225E+06	2.44031E-02	2.40480E-02	2.43438E-02

TAB. 2.10 Continued

VITENDF70.BOLIB Library Total Neutron Fission Spectra for U-235, U-238 and Pu-239.

Group	Upper Energy [eV]	U-235 X	U-238 X	Pu-239 X
50	2.0190E+06	2.43446E-02	2.40485E-02	2.42247E-02
51	1.9205E+06	2.41644E-02	2.39299E-02	2.39929E-02
52	1.8268E+06	2.38759E-02	2.36894E-02	2.36599E-02
53	1.7377E+06	2.34973E-02	2.33656E-02	2.32394E-02
54	1.6530E+06	2.30653E-02	2.29788E-02	2.27713E-02
55	1.5724E+06	2.25743E-02	2.25411E-02	2.22444E-02
56	1.4957E+06	2.20394E-02	2.20779E-02	2.16740E-02
57	1.4227E+06	2.14035E-02	2.14608E-02	2.10102E-02
58	1.3534E+06	2.07889E-02	2.08802E-02	2.03799E-02
59	1.2874E+06	2.01424E-02	2.02639E-02	1.97095E-02
60	1.2246E+06	1.95105E-02	1.96781E-02	1.90497E-02
61	1.1648E+06	1.88028E-02	1.89854E-02	1.83262E-02
62	1.1080E+06	3.54943E-02	3.59530E-02	3.45146E-02
63	1.0026E+06	1.39988E-02	1.42192E-02	1.35762E-02
64	9.6164E+05	1.87557E-02	1.90732E-02	1.81574E-02
65	9.0718E+05	1.53248E-02	1.55992E-02	1.48163E-02
66	8.6294E+05	1.46545E-02	1.49572E-02	1.41468E-02
67	8.2085E+05	1.39821E-02	1.42882E-02	1.34708E-02
68	7.8082E+05	1.33466E-02	1.36751E-02	1.28294E-02
69	7.4274E+05	1.27009E-02	1.30335E-02	1.21924E-02
70	7.0651E+05	1.20706E-02	1.24073E-02	1.15671E-02
71	6.7206E+05	1.14668E-02	1.18034E-02	1.09706E-02
72	6.3928E+05	1.08914E-02	1.12478E-02	1.03949E-02
73	6.0810E+05	1.03235E-02	1.06967E-02	9.84075E-03
74	5.7844E+05	9.76520E-03	1.01484E-02	9.29471E-03
75	5.5023E+05	9.22553E-03	9.60066E-03	8.76171E-03
76	5.2340E+05	8.72337E-03	9.11924E-03	8.26667E-03
77	4.9787E+05	1.59481E-02	1.67150E-02	1.50931E-02
78	4.5049E+05	1.40613E-02	1.47590E-02	1.32947E-02
79	4.0762E+05	6.40656E-03	6.74257E-03	6.04067E-03
80	3.8774E+05	6.02002E-03	6.35863E-03	5.66889E-03
81	3.6883E+05	1.08842E-02	1.15143E-02	1.02493E-02
82	3.3373E+05	9.52454E-03	1.00995E-02	8.95953E-03
83	3.0197E+05	1.02262E-03	1.08605E-03	9.61614E-04
84	2.9849E+05	3.75182E-04	3.98718E-04	3.52722E-04
85	2.9721E+05	7.86566E-04	8.36010E-04	7.39444E-04
86	2.9452E+05	2.11344E-03	2.24788E-03	1.98634E-03
87	2.8725E+05	4.02252E-03	4.28495E-03	3.77777E-03
88	2.7324E+05	7.26811E-03	7.74599E-03	6.82012E-03
89	2.4724E+05	3.27331E-03	3.49017E-03	3.06853E-03
90	2.3518E+05	3.05036E-03	3.25667E-03	2.85961E-03
91	2.2371E+05	2.84389E-03	3.03769E-03	2.66514E-03
92	2.1280E+05	2.64960E-03	2.83138E-03	2.48324E-03
93	2.0242E+05	2.46704E-03	2.64182E-03	2.31239E-03
94	1.9255E+05	2.30370E-03	2.47644E-03	2.15560E-03
95	1.8316E+05	2.15113E-03	2.31750E-03	2.00998E-03
96	1.7422E+05	2.00153E-03	2.15774E-03	1.86855E-03
97	1.6573E+05	1.86740E-03	2.01254E-03	1.74253E-03
98	1.5764E+05	1.73463E-03	1.86757E-03	1.61862E-03

TAB. 2.10 Continued

VITENDF70.BOLIB Library Total Neutron Fission Spectra for U-235, U-238 and Pu-239.

Group	Upper Energy [eV]	U-235 X	U-238 X	Pu-239 X
99	1.4996E+05	1.61776E-03	1.74049E-03	1.50952E-03
100	1.4264E+05	1.50372E-03	1.61996E-03	1.40221E-03
101	1.3569E+05	1.40184E-03	1.51244E-03	1.30638E-03
102	1.2907E+05	1.30530E-03	1.41228E-03	1.21556E-03
103	1.2277E+05	1.21089E-03	1.31028E-03	1.12779E-03
104	1.1679E+05	1.12782E-03	1.22064E-03	1.05046E-03
105	1.1109E+05	2.48174E-03	2.68071E-03	2.31195E-03
106	9.8037E+04	2.07246E-03	2.24412E-03	1.92774E-03
107	8.6517E+04	6.93576E-04	7.53783E-04	6.44999E-04
108	8.2503E+04	5.09006E-04	5.53796E-04	4.73142E-04
109	7.9499E+04	1.23035E-03	1.34208E-03	1.14311E-03
110	7.1998E+04	7.27543E-04	7.87821E-04	6.76125E-04
111	6.7379E+04	1.60973E-03	1.73204E-03	1.49606E-03
112	5.6562E+04	5.72664E-04	6.15820E-04	5.31977E-04
113	5.2475E+04	8.24983E-04	8.91338E-04	7.65589E-04
114	4.6309E+04	6.86850E-04	7.45611E-04	6.36411E-04
115	4.0868E+04	7.71330E-04	8.38078E-04	7.14334E-04
116	3.4307E+04	2.74573E-04	2.99190E-04	2.53852E-04
117	3.1828E+04	3.54298E-04	3.87604E-04	3.26313E-04
118	2.8501E+04	1.54479E-04	1.69722E-04	1.41650E-04
119	2.7000E+04	9.49637E-05	1.04482E-04	8.69987E-05
120	2.6058E+04	1.25548E-04	1.38323E-04	1.14918E-04
121	2.4788E+04	5.94622E-05	6.55945E-05	5.43843E-05
122	2.4176E+04	5.73442E-05	6.33113E-05	5.24194E-05
123	2.3579E+04	1.60003E-04	1.76954E-04	1.46105E-04
124	2.1875E+04	2.31266E-04	2.57489E-04	2.10222E-04
125	1.9305E+04	3.56542E-04	4.03479E-04	3.20453E-04
126	1.5034E+04	2.48177E-04	2.84193E-04	2.21343E-04
127	1.1709E+04	7.67522E-05	8.87156E-05	6.80352E-05
128	1.0595E+04	9.28095E-05	1.04500E-04	8.39487E-05
129	9.1188E+03	1.12799E-04	1.24855E-04	1.03421E-04
130	7.1017E+03	7.82429E-05	8.75073E-05	7.13475E-05
131	5.5308E+03	5.43256E-05	6.14618E-05	4.92361E-05
132	4.3074E+03	2.42788E-05	2.77521E-05	2.18829E-05
133	3.7074E+03	1.34838E-05	1.55175E-05	1.21085E-05
134	3.3546E+03	1.16632E-05	1.34976E-05	1.04417E-05
135	3.0354E+03	1.00942E-05	1.17498E-05	9.00811E-06
136	2.7465E+03	4.52318E-06	5.28900E-06	4.02638E-06
137	2.6126E+03	4.20916E-06	4.93689E-06	3.74048E-06
138	2.4852E+03	7.56126E-06	8.90970E-06	6.70189E-06
139	2.2487E+03	6.54719E-06	7.76442E-06	5.78204E-06
140	2.0347E+03	1.27876E-05	1.53378E-05	1.12198E-05
141	1.5846E+03	8.94093E-06	1.09171E-05	7.76302E-06
142	1.2341E+03	6.26443E-06	7.79748E-06	5.37656E-06
143	9.6112E+02	4.39888E-06	5.58861E-06	3.72765E-06
144	7.4852E+02	3.09549E-06	4.01912E-06	2.58669E-06
145	5.8295E+02	2.18380E-06	2.90106E-06	1.79711E-06
146	4.5400E+02	1.54473E-06	2.10179E-06	1.25015E-06
147	3.5357E+02	1.09569E-06	1.52832E-06	8.70805E-07

TAB. 2.10 Continued

VITENDF70.BOLIB Library Total Neutron Fission Spectra for U-235, U-238 and Pu-239.

Group	Upper Energy [eV]	U-235 X	U-238 X	Pu-239 X
148	2.7536E+02	7.79591E-07	1.11567E-06	6.07561E-07
149	2.1445E+02	5.56391E-07	8.17495E-07	4.24569E-07
150	1.6702E+02	3.98609E-07	6.01613E-07	2.97383E-07
151	1.3007E+02	2.86418E-07	4.44212E-07	2.08611E-07
152	1.0130E+02	2.06642E-07	3.29365E-07	1.46736E-07
153	7.8893E+01	1.49629E-07	2.45099E-07	1.03450E-07
154	6.1442E+01	1.08761E-07	1.83045E-07	7.31235E-08
155	4.7851E+01	7.93633E-08	1.37177E-07	5.18335E-08
156	3.7266E+01	5.81324E-08	1.03129E-07	3.68484E-08
157	2.9023E+01	4.27526E-08	7.77828E-08	2.62821E-08
158	2.2603E+01	3.15557E-08	5.88235E-08	1.88039E-08
159	1.7604E+01	2.33884E-08	4.46235E-08	1.35056E-08
160	1.3710E+01	1.73974E-08	3.39341E-08	9.73426E-09
161	1.0677E+01	1.28576E-08	2.57326E-08	6.91964E-09
162	8.3153E+00	9.32006E-09	1.93467E-08	4.72477E-09
163	6.4760E+00	6.82789E-09	1.46367E-08	3.26711E-09
164	5.0435E+00	5.05610E-09	1.11373E-08	2.29406E-09
165	3.9279E+00	3.77951E-09	8.51593E-09	1.63493E-09
166	3.0590E+00	2.84692E-09	6.53510E-09	1.18103E-09
167	2.3824E+00	2.15915E-09	5.03183E-09	8.64046E-10
168	1.8554E+00	1.64607E-09	3.88314E-09	6.39000E-10
169	1.4450E+00	5.74175E-10	1.36455E-09	2.18677E-10
170	1.3000E+00	6.86647E-10	1.63892E-09	2.58550E-10
171	1.1253E+00	1.77133E-10	4.23951E-10	6.61649E-11
172	1.0800E+00	1.56095E-10	3.74040E-10	5.81233E-11
173	1.0400E+00	1.55801E-10	3.73750E-10	5.78416E-11
174	1.0000E+00	4.79449E-10	1.15297E-09	1.76907E-10
175	8.7643E-01	2.95142E-10	7.11677E-10	1.08074E-10
176	8.0000E-01	4.51413E-10	1.09150E-09	1.64059E-10
177	6.8256E-01	2.20092E-10	5.33446E-10	7.94394E-11
178	6.2506E-01	3.56516E-10	8.65988E-10	1.27905E-10
179	5.3158E-01	1.20077E-10	2.92169E-10	4.28619E-11
180	5.0000E-01	3.26108E-10	7.94870E-10	1.15846E-10
181	4.1399E-01	1.78344E-10	4.35521E-10	6.30065E-11
182	3.6680E-01	1.57631E-10	3.85441E-10	5.54824E-11
183	3.2500E-01	1.88132E-10	4.60640E-10	6.59624E-11
184	2.7500E-01	1.87673E-10	4.60189E-10	6.55222E-11
185	2.2500E-01	1.53549E-10	3.76999E-10	5.33996E-11
186	1.8400E-01	1.27099E-10	3.12384E-10	4.40581E-11
187	1.5000E-01	9.33193E-11	2.29148E-10	3.22658E-11
188	1.2500E-01	9.32044E-11	2.29009E-10	3.21557E-11
189	1.0000E-01	1.11694E-10	2.74792E-10	3.84416E-11
190	7.0000E-02	7.43700E-11	1.83174E-10	2.55395E-11
191	5.0000E-02	3.71573E-11	9.15869E-11	1.27432E-11
192	4.0000E-02	3.71390E-11	9.15868E-11	1.27257E-11
193	3.0000E-02	3.34095E-11	8.24281E-11	1.14381E-11
194	2.1000E-02	2.41198E-11	5.95314E-11	8.25198E-12
195	1.4500E-02	1.66938E-11	4.12140E-11	5.70857E-12
196	1.0000E-02	1.85443E-11	4.57934E-11	6.33868E-12

TAB. 2.10 Continued

VITENDF70.BOLIB Library Total Neutron Fission Spectra for U-235, U-238 and Pu-239.

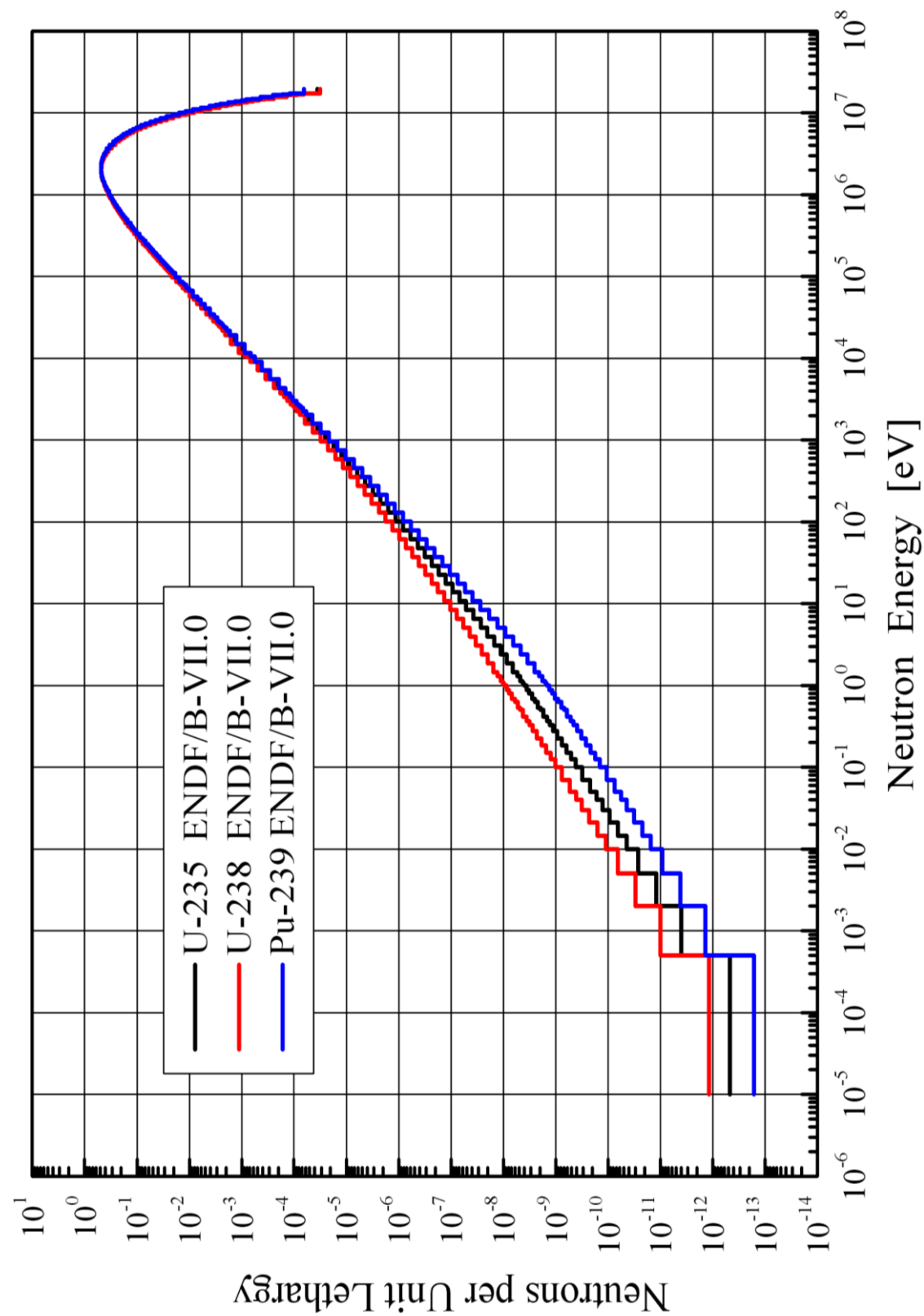| Group	Upper Energy [eV]	U-235 X	U-238 X	Pu-239 X
197	5.0000E-03	1.11244E-11	2.74760E-11	3.80110E-12
198	2.0000E-03	5.56158E-12	1.37380E-11	1.89996E-12
199	5.0000E-04	1.85377E-12	4.57934E-12	6.33231E-13
	Lower Energy 1.0000E-05			

FIG. 2.5 199 Group Representation of the Total (Prompt + Delayed) Neutron Fission Spectrum for the U-235, U-238 and Pu-239 Processed Files Included in the VITENDF70.BOLIB Library. Spectra Averaged on Incident Neutron Energies and Normalized to 1 Neutron per Fission.

2.9 - Library Validation

The VITENDF70.BOLIB /5/ library was extensively tested on about 80, thermal, intermediate and fast neutron spectrum criticality safety benchmark experiments, taking the compositional and geometrical data for the transport calculations exclusively from the ICSBEP handbook (2004 Edition) /62/ of benchmark specifications.

The k-effective (k_{eff}) results obtained with the VITENDF70.BOLIB library were compared with the results previously obtained in ENEA-Bologna with the VITJEFF311.BOLIB /31/ library.

The ENEA-Bologna 2007 Revision /19/ of the SCAMPI /18/ system was used to prepare, through the BONAMI module, the working libraries of neutron self-shielded cross sections for the one-dimensional (1D) and two-dimensional (2D) transport calculations, respectively performed with the XSDRNPM 1D discrete ordinates transport code of the SCAMPI system and the DORT 2D discrete ordinates transport code, included in the DOORS-3.2 /35/ system.

Following the denominations reported in the ICSBEP handbook, the k-effective (k_{eff}) results obtained in the 1D and 2D transport calculations with the VITENDF70.BOLIB library refer to HEU (Highly Enriched Uranium) benchmark experiments (TAB. 2.11), to IEU (Intermediate Enrichement Uranium) benchmark experiments (TAB. 2.12), to LEU (Low Enriched Uranium) benchmark experiments (TAB. 2.13), to a MIX (MIXed Plutonium-Uranium) benchmark experiment (TAB. 2.14), to U-233 benchmark experiments (TAB. 2.15) and to Pu-239 benchmark experiments (TAB. 2.16). The following acronyms are used in the cited tables: NU (natural uranium), DU (depleted uranium) and WC (wolfram carbide or tungsten carbide).

The k-effective (k_{eff}) results obtained with the VITENDF70.BOLIB library are compared in the cited tables with the corresponding results obtained with the VITJEFF311.BOLIB library.

It is underlined that in the ICSBEP Handbook it is considered usually reasonable the assumption of a normal probability distribution for the k_{eff} values of the "experimental or benchmark-model" configurations. Consequently the k_{eff} values reported in TABs. 2.11÷2.16 are the mean values of the assumed normal distributions and their associated uncertainties correspond to one standard deviation (± sigma) which represents a 68% confidence level.

The one-dimensional transport calculations in spherical geometry were performed with the XSDRNPM code, module of the SCAMPI system, in the P_5-S_{16} approximation: P_5 corresponds to L=ℓ-max, the maximum order of the Legendre polynomial (P_ℓ) expansion of the scattering cross section matrices and S_{16} represents the order of the flux angular discretization. The DORT 2D discrete ordinates code, included in the DOORS-3.2 system, was exclusively employed in the simulation of the IEU-MET-FAST-007 (BIG TEN) (see TAB. 2.12) fast neutron spectrum benchmark experiment with a 2D (R,Z) cylindrical geometry calculation in the P_3-S_8 approximation.

It is noted that the most part of the results obtained with the VITENDF70.BOLIB library on the U-233 benchmark experiments fall within the experimental or benchmark-model k-effective errors whereas the corresponding results obtained with the VITJEFF311.BOLIB library are always outside the cited experimental errors.

At present no testing of the VITENDF70.BOLIB library was performed on integral shielding benchmark experiments.

TAB. 2.11

VITENDF70.BOLIB and VITJEFF311.BOLIB Criticality Calculation Results
for Highly Enriched Uranium (HEU) Benchmark Experiments.

ICSBEP Handbook Benchmark Name	Reflector	Experimental or Benchmark-Model $k_{eff} \pm \Delta k_{eff}$ (pcm)	Geometry/ P_L - S_N	VITENDF70 k_{eff}	VITJEFF311 k_{eff}
Thermal Neutron Spectrum					
HEU-SOL-THERM-009-001	H2O	$0.99900 \pm (430)$	1D Sph./ P5-S16	1.00378	1.00189
HEU-SOL-THERM-009-002	H2O	$1.00000 \pm (390)$	1D Sph./ P5-S16	1.00425	1.00236
HEU-SOL-THERM-009-003	H2O	$1.00000 \pm (360)$	1D Sph./ P5-S16	1.00342	1.00166
HEU-SOL-THERM-009-004	H2O	$0.99860 \pm (350)$	1D Sph./ P5-S16	0.99737	0.99591
HEU-SOL-THERM-010-001	H2O	$1.00000 \pm (290)$	1D Sph./ P5-S16	1.00192	1.00123
HEU-SOL-THERM-010-002	H2O	$1.00000 \pm (290)$	1D Sph./ P5-S16	1.00255	1.00185
HEU-SOL-THERM-010-003	H2O	$1.00000 \pm (290)$	1D Sph./ P5-S16	1.00123	1.00044
HEU-SOL-THERM-010-004	H2O	$0.99920 \pm (290)$	1D Sph./ P5-S16	0.99989	0.99906
HEU-SOL-THERM-011-001	H2O	$1.00000 \pm (230)$	1D Sph./ P5-S16	1.00517	1.00523
HEU-SOL-THERM-011-002	H2O	$1.00000 \pm (230)$	1D Sph./ P5-S16	1.00129	1.00137
HEU-SOL-THERM-012-001	H2O	$0.99990 \pm (580)$	1D Sph./ P5-S16	1.00071	1.00111
HEU-SOL-THERM-013-001 (ORNL-1)	--	$1.00120 \pm (260)$	1D Sph./ P5-S16	0.99832	0.99892
HEU-SOL-THERM-013-002 (ORNL-2)	--	$1.00070 \pm (360)$	1D Sph./ P5-S16	0.99749	0.99797
HEU-SOL-THERM-013-003 (ORNL-3)	--	$1.00090 \pm (360)$	1D Sph./ P5-S16	0.99406	0.99442
HEU-SOL-THERM-013-004 (ORNL-4)	--	$1.00030 \pm (360)$	1D Sph./ P5-S16	0.99568	0.99599
HEU-SOL-THERM-032 (ORNL-10)	--	$1.00150 \pm (260)$	1D Sph./ P5-S16	0.99893	0.99878
Intermediate Neutron Spectrum					
ICSBEP Handbook Benchmark Name	Reflector	Experimental or Benchmark-Model $k_{inf} \pm \Delta k_{inf}$ (pcm)	Geometry/ P_L - S_N	VITENDF70 k_{inf}	VITJEFF311 k_{inf}
HEU-COMP-INTER-004	--	$1.00000 \pm (400)$	Inf. Homogeneous/ P5-S16	1.00970	1.00807

TAB. 2.11 Continued

VITENDF70.BOLIB and VITJEFF311.BOLIB Criticality Calculation Results
for Highly Enriched Uranium (HEU) Benchmark Experiments.

ICSBEP Handbook Benchmark Name	Reflector	Experimental or Benchmark-Model $k_{eff} \pm \Delta k_{eff}$(pcm)	Geometry/ P_L - S_N	VITENDF70 k_{eff}	VITJEFF311 k_{eff}
		Fast Neutron Spectrum			
HEU-MET-FAST-001-001 (GODIVA)	--	$1.00000 \pm (100)$	1D Sph./ P5-S16	0.99954	0.99533
HEU-MET-FAST-002-001 (TOPSY)	NU	$1.00000 \pm (300)$	1D Sph./ P5-S16	1.00266	1.00095
HEU-MET-FAST-003-001	NU (2in)	$1.00000 \pm (500)$	1D Sph./ P5-S16	0.99576	0.99373
HEU-MET-FAST-003-002	NU (3in)	$1.00000 \pm (500)$	1D Sph./ P5-S16	0.99544	0.99345
HEU-MET-FAST-003-003	NU (4in)	$1.00000 \pm (500)$	1D Sph./ P5-S16	1.00027	0.99825
HEU-MET-FAST-003-004	NU (5in)	$1.00000 \pm (300)$	1D Sph./ P5-S16	0.99843	0.99643
HEU-MET-FAST-003-005	NU (7in)	$1.00000 \pm (300)$	1D Sph./ P5-S16	1.00251	1.00071
HEU-MET-FAST-003-006	NU (8in)	$1.00000 \pm (300)$	1D Sph./ P5-S16	1.00262	1.00092
HEU-MET-FAST-003-007	NU (11in)	$1.00000 \pm (300)$	1D Sph./ P5-S16	1.00299	1.00148
HEU-MET-FAST-003-008	WC (1.9in)	$1.00000 \pm (500)$	1D Sph./ P5-S16	1.00962	1.00146
HEU-MET-FAST-003-009	WC (2.9in)	$1.00000 \pm (500)$	1D Sph./ P5-S16	1.01093	1.00364
HEU-MET-FAST-003-010	WC (4.5in)	$1.00000 \pm (500)$	1D Sph./ P5-S16	1.01379	1.00897
HEU-MET-FAST-003-011	WC (6.5in)	$1.00000 \pm (500)$	1D Sph./ P5-S16	1.01733	1.01416
HEU-MET-FAST-003-012	Ni (8in)	$1.00000 \pm (500)$	1D Sph./ P5-S16	1.01600	1.00922
HEU-MET-FAST-022-001	Al	$1.00000 \pm (190)$	1D Sph./ P5-S16	0.99941	0.99527
HEU-MET-FAST-027-001	Pb	$1.00000 \pm (250)$	1D Sph./ P5-S16	1.00019	1.00052
HEU-MET-FAST-028-001 (FLATTOP-25)	NU	$1.00000 \pm (300)$	1D Sph./ P5-S16	1.00397	1.00218
HEU-MET-FAST-057-001	Pb	$1.00000 \pm (200)$	1D Sph./ P5-S16	0.98979	0.99425
HEU-MET-FAST-057-002	Pb	$1.00000 \pm (230)$	1D Sph./ P5-S16	0.99788	1.00122

TAB. 2.12

VITENDF70.BOLIB and VITJEFF311.BOLIB Criticality Calculation Results
for Intermediate Enrichment Uranium (IEU) Benchmark Experiments.

ICSBEP Handbook Benchmark Name	Reflector	Experimental or Benchmark-Model $k_{eff} \pm \Delta k_{eff}$ (pcm)	Geometry/ P_L - S_N	VITENDF70 k_{eff}	VITJEFF311 k_{eff}
Fast Neutron Spectrum					
IEU-MET-FAST-007 (BIG TEN) Simplified Model	DU	$1.00450 \pm (70)$	2D Cyl./ P3-S8	1.00416	0.99714
IEU-MET-FAST-007 (BIG TEN) Two Zone Model	DU	$0.99480 \pm (130)$	2D Cyl./ P3-S8	0.99705	0.99039

TAB. 2.13

VITENDF70.BOLIB and VITJEFF311.BOLIB Criticality Calculation Results
for Low Enriched Uranium (LEU) Benchmark Experiments.

ICSBEP Handbook Benchmark Name	Reflector	Experimental or Benchmark-Model $k_{eff} \pm \Delta k_{eff}$ (pcm)	Geometry/ P_L - S_N	VITENDF70 k_{eff}	VITJEFF311 k_{eff}
Thermal Neutron Spectrum					
LEU-SOL-THERM-003-003	--	$0.99950 \pm (420)$	1D Sph./ P5-S16	1.00163	1.00358
LEU-SOL-THERM-003-006	--	$0.99990 \pm (490)$	1D Sph./ P5-S16	0.99869	1.00088
LEU-SOL-THERM-003-009	--	$0.99960 \pm (520)$	1D Sph./ P5-S16	0.99757	0.99942

TAB. 2.14

VITENDF70.BOLIB and VITJEFF311.BOLIB Criticality Calculation Results for Mixed Plutonium-Uranium (MIX) Benchmark Experiments.

ICSBEP Handbook Benchmark Name	Reflector	Experimental or Benchmark-Model $k_{eff} \pm \Delta k_{eff}$(pcm)	Geometry/ P_L - S_N	VITENDF70 k_{eff}	VITJEFF311 k_{eff}
Fast Neutron Spectrum					
MIX-MET-FAST-001-001	--	1.00000 ± (160)	1D Sph./ P5-S16	0.99923	0.99740

TAB. 2.15

VITENDF70.BOLIB and VITJEFF311.BOLIB Criticality Calculation Results for U-233 Benchmark Experiments.

ICSBEP Handbook Benchmark Name	Reflector	Experimental or Benchmark-Model $k_{eff} \pm \Delta k_{eff}$(pcm)	Geometry/ P_L - S_N	VITENDF70 k_{eff}	VITJEFF311 k_{eff}
Fast Neutron Spectrum					
U233-MET-FAST-001-001 (JEZEBEL-233)	--	1.00000 ± (100)	1D Sph./.P5-S16	1.00065	1.00383
U233-MET-FAST-002-001	HEU	1.00000 ± (100)	1D Sph./.P5-S16	1.00020	1.00257
U233-MET-FAST-002-002	HEU	1.00000 ± (110)	1D Sph./.P5-S16	1.00174	1.00342
U233-MET-FAST-003-001	NU	1.00000 ± (100)	1D Sph./.P5-S16	1.00087	1.00575
U233-MET-FAST-003-002	NU	1.00000 ± (100)	1D Sph./.P5-S16	1.00184	1.00719
U233-MET-FAST-006-001 (FLATTOP-23)	NU	1.00000 ± (140)	1D Sph./.P5-S16	1.00138	1.00665

TAB. 2.16

VITENDF70.BOLIB and VITJEFF311.BOLIB Criticality Calculation Results
for Plutonium Benchmark Experiments.

ICSBEP Handbook Benchmark Name	Reflector	Experimental or Benchmark-Model $k_{eff} \pm \Delta k_{eff}$(pcm)	Geometry/ P_L - S_N	VITENDF70 k_{eff}	VITJEFF311 k_{eff}
		Thermal Neutron Spectrum			
PU-SOL-THERM-006-001	H2O	$1.00000 \pm (350)$	1D Sph./ P5-S16	1.00095	0.99716
PU-SOL-THERM-006-002	H2O	$1.00000 \pm (350)$	1D Sph./ P5-S16	1.00223	0.99843
PU-SOL-THERM-006-003	H2O	$1.00000 \pm (350)$	1D Sph./ P5-S16	1.00181	0.99800
PU-SOL-THERM-011-001	--	$1.00000 \pm (520)$	1D Sph./ P5-S16	1.01017	1.00721
PU-SOL-THERM-011-002	--	$1.00000 \pm (520)$	1D Sph./ P5-S16	1.01489	1.01190
PU-SOL-THERM-011-003	--	$1.00000 \pm (520)$	1D Sph./ P5-S16	1.01697	1.01397
PU-SOL-THERM-011-004	--	$1.00000 \pm (520)$	1D Sph./ P5-S16	1.00951	1.00656
PU-SOL-THERM-011-005 (PNL-5R)	--	$1.00000 \pm (520)$	1D Sph./ P5-S16	1.00659	1.00364
PU-SOL-THERM-011-006 (PNL-3R)	--	$1.00000 \pm (520)$	1D Sph./ P5-S16	0.99453	0.99165
PU-SOL-THERM-011-007	--	$1.00000 \pm (520)$	1D Sph./ P5-S16	1.00052	0.99760
PU-SOL-THERM-011-008	--	$1.00000 \pm (520)$	1D Sph./ P5-S16	0.99712	0.99421
PU-SOL-THERM-011-009	--	$1.00000 \pm (520)$	1D Sph./ P5-S16	0.99376	0.99088
PU-SOL-THERM-011-010	--	$1.00000 \pm (520)$	1D Sph./ P5-S16	1.00378	1.00083
PU-SOL-THERM-011-011 (PNL-4R)	--	$1.00000 \pm (520)$	1D Sph./ P5-S16	1.00043	0.99752
PU-SOL-THERM-011-012	--	$1.00000 \pm (520)$	1D Sph./ P5-S16	0.99993	0.99702
PU-SOL-THERM-021-007 (PNL-1)	--	$1.00000 \pm (320)$	1D Sph./ P5-S16	1.00699	1.00412
PU-SOL-THERM-021-008 (PNL-2)	--	$1.00000 \pm (650)$	1D Sph./ P5-S16	1.00424	1.00226
PU-SOL-THERM-021-009	--	$1.00000 \pm (320)$	1D Sph./ P5-S16	1.00748	1.00461
		Intermediate Neutron Spectrum			
ICSBEP Handbook Benchmark Name	Reflector	Experimental or Benchmark-Model $k_{eff} \pm \Delta k_{eff}$ (pcm)	Geometry/ P_L - S_N	VITENDF70 k_{eff}	VITJFF311 k_{eff}
PU-COMP-INTER-001	--	$1.00000 \pm (1100)$	Inf. Homogeneous/ P5-S16	1.00517	1.00110

TAB. 2.16 Continued

VITENDF70.BOLIB and VITJEFF311.BOLIB Criticality Calculation Results
for Plutonium Benchmark Experiments.

ICSBEP Handbook Benchmark Name	Reflector	Experimental or Benchmark-Model $k_{eff} \pm \Delta k_{eff}$ (pcm)	Geometry/ P_L - S_N	VITENDF70 k_{eff}	VITJEFF311 k_{eff}
Fast Neutron Spectrum					
PU-MET-FAST-001-001 (JEZEBEL)	--	1.00000 ± (200)	1D Sph./ P5-S16	0.99924	0.99889
PU-MET-FAST-002-001 (JEZEBEL-240)	--	1.00000 ± (200)	1D Sph./ P5-S16	0.99966	1.00277
PU-MET-FAST-006-001 (FLATTOP-PU)	NU	1.00000 ± (300)	1D Sph./ P5-S16	1.00163	1.00305
PU-MET-FAST-008-001 (THOR)	Th	1.00000 ± (60)	1D Sph./ P5-S16	0.99848	1.00124
PU-MET-FAST-009-001	Al	1.00000 ± (270)	1D Sph./ P5-S16	1.00492	1.00401
PU-MET-FAST-010-001	NU	1.00000 ± (180)	1D Sph./ P5-S16	0.99971	1.00091
PU-MET-FAST-011-001	H2O	1.00000 ± (100)	1D Sph./ P5-S16	1.00262	0.99943
PU-MET-FAST-018-001	Be	1.00000 ± (300)	1D Sph./ P5-S16	0.99705	1.00200
PU-MET-FAST-023-001	Graphite	1.00000 ± (230)	1D Sph./ P5-S16	1.00008	0.99850
PU-MET-FAST-024-001	Polyethylene	1.00000 ± (200)	1D Sph./ P5-S16	1.00192	0.99948
PU-MET-FAST-030-001	Graphite	1.00000 ± (210)	1D Sph./ P5-S16	1.00377	1.00395
PU-MET-FAST-031-001	Polyethylene	1.00000 ± (210)	1D Sph./ P5-S16	1.00530	1.00385

3 - BUGENDF70.BOLIB BROAD-GROUP LIBRARY SPECIFICATIONS

The ENEA-Bologna Nuclear Data Group generated the BUGENDF70.BOLIB broad-group coupled neutron/photon working cross section library in FIDO-ANISN /4/ format, based on the US ENDF/B-VII.0 /6/ evaluated nuclear data library. This BUGLE-type library has features similar to the ORNL DLC-0185/BUGLE-96 /8/ broad-group working library and the same neutron and photon energy group structures (47 n + 20 γ) with the 47 neutron energy groups covering the energy range 1.0E-05 eV - 1.7332E+07 eV and the 20 photon groups included within the energy range 1.0E+04 - 1.4E+07 eV.

As previously performed at ORNL, where BUGLE-96 was obtained through proper cross section collapsing with the ORNL SCAMPI /18/ nuclear data processing system from the ORNL DLC-0184/VITAMIN-B6 /8/ (ENDF/B-VI.3 /9/) fine-group (199 n + 42 γ) library, the ENEA-Bologna BUGENDF70.BOLIB library was generated with the same methodology and reactor models previously used to produce the BUGLE-96 library. It was obtained through cross section collapsing from the recently produced ENEA-Bologna VITENDF70.BOLIB /5/ fine-group library (see Chapter 2), based on ENDF/B-VII.0 data and characterized by the same neutron and photon energy group structure as VITAMIN-B6.

In order to perform this task an updating was required to the original SCAMPI, developed at ORNL from the AMPX-77 /23/ system and already employed to generate BUGLE-96. In particular, the so called "ENEA-Bologna 2007 Revision of SCAMPI" /19/ was developed (see 1.3) and released to the OECD-NEA Data Bank and ORNL-RSICC.

3.1 - Name

The present problem-dependent broad-group working cross section library, derived through proper cross section collapsing of the ENEA-Bologna VITENDF70.BOLIB /5/ fine-group mother library, is designated as BUGENDF70.BOLIB.
"BUG" suggests that the main features of the library are similar to those of the BUGLE-96 /8/ library, generated at ORNL. The "ENDF70" designation in both ENEA-Bologna libraries conveniently reflects the origin of the evaluated data, i.e. the US ENDF/B-VII.0 /6/ evaluated nuclear data library (see 1.2). Finally, "BOLIB" means BOlogna LIBrary and so it is indicative of the place of production of the library.

3.2 - Materials, Legendre Order of Scattering and Energy Group Structure

The BUGENDF70.BOLIB library contains all the 183 nuclides, based on the the US ENDF/B-VII.0 /6/ evaluated nuclear data library, available in the VITENDF70.BOLIB /5/ fine-group mother library, as listed in TAB. 2.1. Some nuclides appear several times due to the inclusion of different resonance self-shielding and energy weighting options for key fuel and structural materials.

The Legendre order of scattering of the cross sections contained in BUGENDF70.BOLIB is the same as available in the VITENDF70.BOLIB library: P_7 for both neutrons and photons for nuclides with Z=l through Z=29 (copper) and P_5 for the remainder of the nuclides (see TAB. 2.2).

The BUGENDF70.BOLIB library, as previously reported, has the same neutron and photon energy group structures (47 neutron groups + 20 photon groups) as the ORNL BUGLE-96 /8/ library with 47 neutron groups covering the energy range 1.0E-05 eV - 1.7332E+07 eV and 20 photon groups included within the energy range 1.0E+04 eV - 1.4E+07 eV. The energy boundaries for the 47 neutron groups are given in TAB. 3.1 along with the corresponding VITENDF70.BOLIB group numbers which were collapsed to form the BUGENDF70.BOLIB groups. Similarly, the 20 photon group structure is given in TAB. 3.2.

With respect to the neutron energy range covered by the VITENDF70.BOLIB library, it is underlined that in the BUGENDF70.BOLIB neutron energy group structure (see TAB. 3.1), the contribution of the highest neutron energy group (neutron fine-group No.1 in the energy range 1.7332E+07 eV - 1.9640E+07 eV; see TAB. 2.4) of the VITENDF70.BOLIB neutron group structure is omitted. Concerning the photon energy range covered by the VITENDF70.BOLIB library, the contributions of the two highest photon energy groups (photon fine-group No.1 in the energy range 2.0E+07 eV - 3.0E+07 eV and No. 2 in the energy range 1.4E+07 eV - 2.0E+07 eV; see TAB. 2.5) and that of the lowest photon energy group (photon fine-group No. 42 in the energy range 1.0E+03 eV - 1.0E+04 eV; see TAB. 2.5) of the VITENDF70.BOLIB photon group structure are omitted in the BUGENDF70.BOLIB photon energy group structure (see TAB. 3.2).

The BUGENDF70.BOLIB thermal neutron energy range below 5.0435 eV, i.e. the energy range containing the neutron groups which include upscatter in the BUGENDF70T.BOLIB version of the library (see also 3.5), has only five neutron groups (see TAB. 3.1) with respect to the 36 neutron groups of the VITENDF70.BOLIB library in the same energy range (see TAB. 2.4).

TAB. 3.1

Neutron Group Energy Boundaries for the BUGENDF70.BOLIB Library.

Broad Group	Upper Energy [eV]	Upper Lethargy	VITENDF70 Groups
1	1.7332E+07	-5.4997E-01	2-7
2	1.4191E+07	-3.5002E-01	8-12
3	1.2214E+07	-2.0000E-01	13-16
4	1.0000E+07	0.0000E+00	17-19
5	8.6071E+06	1.5000E-01	20-22
6	7.4082E+06	3.0000E-01	23-27
7	6.0653E+06	5.0000E-01	28-31
8	4.9659E+06	7.0000E-01	32-35
9	3.6788E+06	1.0000E+00	36-38
10	3.0119E+06	1.2000E+00	39-40
11	2.7253E+06	1.3000E+00	41-42
12	2.4660E+06	1.4000E+00	43-44
13	2.3653E+06	1.4417E+00	45
14	2.3457E+06	1.4500E+00	46-47
15	2.2313E+06	1.5000E+00	48-50
16	1.9205E+06	1.6500E+00	51-53
17	1.6530E+06	1.8000E+00	54-57
18	1.3534E+06	2.0000E+00	58-62
19	1.0026E+06	2.3000E+00	63-66
20	8.2085E+05	2.5000E+00	67-68
21	7.4274E+05	2.6000E+00	69-72
22	6.0810E+05	2.8000E+00	73-76
23	4.9787E+05	3.0000E+00	77-80
24	3.6883E+05	3.3000E+00	81-84
25	2.9721E+05	3.5159E+00	85-94
26	1.8316E+05	4.0000E+00	95-104
27	1.1109E+05	4.5000E+00	105-110
28	6.7379E+04	5.0000E+00	111-114
29	4.0868E+04	5.5000E+00	115-116
30	3.1828E+04	5.7500E+00	117-119
31	2.6058E+04	5.9500E+00	120-121
32	2.4176E+04	6.0250E+00	122-123
33	2.1875E+04	6.1250E+00	124-125
34	1.5034E+04	6.5000E+00	126-129
35	7.1017E+03	7.2500E+00	130-133
36	3.3546E+03	8.0000E+00	134-140
37	1.5846E+03	8.7500E+00	141-145
38	4.5400E+02	1.0000E+01	146-148
39	2.1445E+02	1.0750E+01	149-151
40	1.0130E+02	1.1500E+01	152-155
41	3.7266E+01	1.2500E+01	156-160
42	1.0677E+01	1.3750E+01	161-163
43	5.0435E+00	1.4500E+01	164-167
44	1.8554E+00	1.5500E+01	168-174
45	8.7643E-01	1.6250E+01	175-180
46	4.1399E-01	1.7000E+01	181-188
47	1.0000E-01	1.8421E+01	189-199
	1.0000E-05	2.7631E+01	

TAB. 3.2

Photon Group Energy Boundaries for the BUGENDF70.BOLIB Library.

Broad Group	Upper Energy [eV]	VITENDF70 Groups
1	1.4000E+07	3-4
2	1.0000E+07	5
3	8.0000E+06	6-7
4	7.0000E+06	8-9
5	6.0000E+06	10-11
6	5.0000E+06	12-13
7	4.0000E+06	14-15
8	3.0000E+06	16-17
9	2.0000E+06	18-19
10	1.5000E+06	20-22
11	1.0000E+06	23
12	8.0000E+05	24
13	7.0000E+05	25
14	6.0000E+05	26-29
15	4.0000E+05	30-31
16	2.0000E+05	32-33
17	1.0000E+05	34-36
18	6.0000E+04	37-39
19	3.0000E+04	40
20	2.0000E+04	41
	1.0000E+04	

3.3 - Self-Shielding, Weighting Spectra and Collapsing

As reported in the BUGLE-96 /8/ library user's manual, the accuracy of the results from a radiation transport calculation which uses broad-group cross section data can be significantly affected by the energy-dependent weighting spectrum used to collapse the data from point-wise or fine-group data. It is important to use a weighting spectrum which is sufficiently prototypical to provide the desired accuracy. In general, a broad-group library is useful only for the range of problems represented by the specific weighting functions. For BUGENDF70.BOLIB, this range includes in-vessel and reactor cavity analyses for light-water-cooled reactors (PWR and BWR). For other applications, the validity of the BUGENDF70.BOLIB data will need to be explicitly demonstrated. Even for LWR applications, it is important that the proper data sets be used for specific regions of the reactor geometry to insure sufficiently accurate results.

The cross section sets contained in the BUGENDF70.BOLIB library were produced in two phases. The first phase was dedicated 1) to properly self-shield the cross sections and 2) to calculate problem-dependent BWR- and PWR-specific neutron/photon weighting spectra. In the second phase, infinitely dilute (not self-shielded) and self-shielded cross section sets, derived from the VITENDF70.BOLIB /5/ fine-group library, were collapsed to generate the BUGENDF70.BOLIB broad-group cross section sets, using the neutron/photon weighting spectra, pre-calculated in the first phase of the data processing. It is underlined that the BUGENDF70.BOLIB broad-group cross sections were generated from the VITENDF70.BOLIB fine-group cross sections with the same methodology and reactor models previously used to produce the ORNL BUGLE-96 library. All the compositional, geometrical and temperature data needed for the cross section self-shielding, the BWR and PWR neutron/photon weighting spectra calculation methodology and the cross section collapsing procedure were consistently taken from the BUGLE-96 library user's manual, respectively from the corresponding ORNL data processing inputs which generated BUGLE-96.

In particular in the first phase, five different neutron/photon weighting spectra in the VITENDF70.BOLIB neutron and photon energy group structures were calculated with a one-dimensional transport code in order to permit, in the second phase, problem-dependent cross section collapsing from the VITENDF70.BOLIB neutron and photon fine-group energy structures into the BUGENDF70.BOLIB neutron and photon broad-group energy structures.

In order to determine the BWR- and PWR-specific neutron/photon weighting spectra, one-dimensional fixed source transport calculations were performed introducing the same compositional/geometrical reactor models used to obtain BUGLE-96: the former representing a typical BWR plant and the latter representing a typical PWR plant. These reactor models, which correspond exactly to those described in the BUGLE-96 library user's manual, are shown in FIG. 3.1. The atomic densities for the various reactor regions, used in the transport calculations to determine the neutron and photon weighting spectra, are given in TAB. 3.3.

The atomic densities used in the transport calculations (see in particular 3.4 for the calculation details) were always directly taken from the transport code input examples reported in the APPENDIX A of the BUGLE-96 library user's manual. On the other hand in certain cases, e.g., for the magnesium, silicon, potassium, calcium and zirconium isotopes, contained in

VITENDF70.BOLIB and involved in the transport calculations, it was necessary to determine their atomic densities which are not available in the BUGLE-96 library user's manual. In fact these isotopes are constituents of the natural elements included as corresponding data files in the VITAMIN-B6 /8/ library since the relative component isotopes were not available in the ENDF/B-VI.3 /9/ evaluated data library. The atomic densities of these isotopes were calculated on the basis of the natural isotopic abundances (see TAB. 3.3), taken from the BNL-NNDC database /63/.

The 1D fixed source transport calculations were performed using as volumetric source spectra the same fine-group neutron source spectrum data set (see APPENDIX A, TAB. A.2 and TAB. A.3) for both types of reactors. These data were taken from the BUGLE-96 library user's manual /8/ (see APPENDIX A, TAB. A.6 and TAB. A.7).

Neutron and photon flux spectra from five representative spatial locations within the previously cited BWR and PWR models (see FIG. 3.1) were then selected corresponding to: 1) off-center in the BWR core region (spatial mesh interval #57 in the BWR model), 2) off-center in the PWR core region (spatial mesh interval #37 in the PWR model), 3) the downcomer region (spatial mesh interval #69 in the PWR model), 4) within the pressure vessel at a depth of one-fourth of the total thickness (spatial mesh interval #82 in the PWR model) and 5) within the concrete shield surrounding the reactor pressure vessel (spatial mesh interval #106 in the PWR model).
The midpoints of the one-dimensional spatial mesh intervals, where the selected neutron and photon flux spectra were calculated, are located at the following distances, respectively from the BWR and PWR core centers (see FIG. 3.1): at about 217.50 cm in the interval #57 in the BWR core region, at about 140.58 cm in the interval #37 in the PWR core region, at about 203.52 cm in the interval #69 in the PWR downcomer, at about 226.28 cm in the interval #82 at a depth of one-fourth of the total thickness of the PWR pressure vessel and at about 270.83 cm in the interval #106 within the concrete shield surrounding the PWR pressure vessel.
The calculated neutron flux spectra are compared graphically in FIG. 3.2 and listed in TAB. 3.4 while the corresponding photon flux spectra are compared in FIG. 3.3 and reported in TAB. 3.5.

The cross sections used in the transport calculations, dedicated to the determination of the five different neutron/photon weighting spectra, were previously separately self-shielded using the BWR and PWR compositional/geometrical models and the specific operating temperatures of the nuclides involved in the calculations. In particular, the proper cross section self-shielding factors were determined on the basis of the specific background cross sections and temperatures of the single nuclides constituting the various material mixtures of the reactor model spatial regions.

In the BWR and PWR core regions, the cross sections of the nuclides involved were self-shielded using the corresponding fuel-clad-moderator pin models. The key compositional, geometrical and temperature parameters for the BWR and PWR pin cell models are given in TAB. 3.6.

In the steel regions, the cross sections of the nuclides that are constituents of steels were separately self-shielded for carbon steel (A533-B), used for the BWR and PWR pressure vessels, and for stainless steel (SS-304), employed for the BWR and PWR core barrels, using

the atomic densities given in TAB. 3.7. As in the case of the BUGLE-96 library, the cross sections of the constituents of carbon steel and stainless steel were self-shielded at the temperature of 600 °K.

It is underlined that, as performed for BUGLE-96, the self-shielding of the cross sections of the nuclides constituting steel was performed following an approach that contained only steel instead of a 50%-50% mixture of water and stainless steel, which was the approach previously followed for similar ORNL working cross section libraries. The present choice using only steel constituents, adopted originally for BUGLE-96, permitted to obtain a significant improvement in the agreement of the results coming from one-dimensional transport calculations using ENDF/B-VI.3 nuclear data and the dosimeter experimental data relative to the ANO-1 (Arkansas Nuclear One) commercial nuclear power reactor, as reported in the BUGLE-96 library user's manual.

The cross section self-shielding for the nuclides of the iron-water mixture representing the PWR downcomer was calculated at the temperature of 590 °K.

Finally, concerning the nuclides constituting the concrete shields, their cross section self-shielding was performed at the temperature of 300 °K.

Although it was decided to use, as much as possible, the input data given in the BUGLE-96 library user's manual, it seems proper to make a few remarks.

The atomic densities of the concrete constituents used in the cross section self-shielding and spectrum calculations, except that of carbon, present different values in TAB. 3.3 (spectrum calculations) and in TAB. 3.7 (self-shielding calculations). The different numerical values reported were respectively taken from the BUGLE-96 /8/ library user's manual (respectively from page 31, TAB. 3.3 and from page A.8 of APPENDIX A, TAB. A.5), although both concrete compositions are labelled with "Type 04".

The atomic densities of the iron isotopes used in the cross section self-shielding calculations for the carbon and stainless steel components (see APPENDIX A, TAB. A.1) were calculated with a different set of natural atomic abundances with respect to the set of atomic abundances of the iron isotopes reported in TAB. 3.3. In particular, in this case, the following atomic abundances were used to determine the atomic densities of the iron isotopes: 5.8% for Fe-54, 91.8% for Fe-56, 2.1% for Fe-57 (identical to the numerical value reported in TAB. 3.3) and 0.3% for Fe-58.

Silicon is present in the carbon and stainless steel material compositions (see TAB. 3.3) used in the spectrum calculations whilst it is not included, as steel constituent, in the corresponding material compositions employed in the cross section self-shielding calculations.

The second phase of the data processing was dedicated, as previously reported, to the cross section collapsing of infinitely dilute (not-self-shielded) and self-shielded cross sections, using the neutron/photon weighting spectra, pre-calculated in the first phase of the data processing.

Concerning the self-shielded cross sections, six sets of collapsed and self-shielded cross sections were generated using the five pre-calculated neutron/photon weighting spectra: in particular there is 1) a set of cross sections for the PWR core materials collapsed with the PWR core flux spectra using a fuel-clad-moderator pin model, 2) a set of cross sections for an iron-water mixture of structural and coolant materials collapsed with the PWR downcomer

flux spectra, 3) a set of cross sections for the carbon steel pressure vessel materials collapsed with the PWR flux spectra at the one-quarter thickness (T) position in the pressure vessel (1/4 T PV), 4) a set of cross sections for the biological shield concrete materials collapsed with the PWR flux spectra in the concrete biological shield, 5) a set of cross sections for the carbon and stainless steel materials collapsed with the PWR pressure vessel flux spectra at the one quarter thickness (T) position in the pressure vessel (1/4 T PV) and, finally, 6) a set of cross sections for the BWR core materials collapsed with the BWR core flux spectra using a fuel-clad-moderator pin model.

Finally, a cross section set of collapsed and infinitely dilute (not self-shielded) cross sections was generated for all the 183 materials contained in the VITENDF70.BOLIB fine-group library. In particular the VITENDF70.BOLIB fine-group cross sections, processed at the infinite dilution background cross section ($\sigma_0 = 1.0E+10$ barns) and at the temperature of 300 °K, were collapsed using the neutron/photon weighting spectrum, pre-calculated in the concrete PWR biological shield during the first phase of the data processing.

FIG. 3.1

One-Dimensional PWR and BWR Radial Geometry and Composition Models[a] Adopted to
Calculate the Specific Flux Spectra for Collapsing the VITENDF70.BOLIB Library
Fine-Group Cross Sections into the BUGENDF70.BOLIB Broad-Group Library.
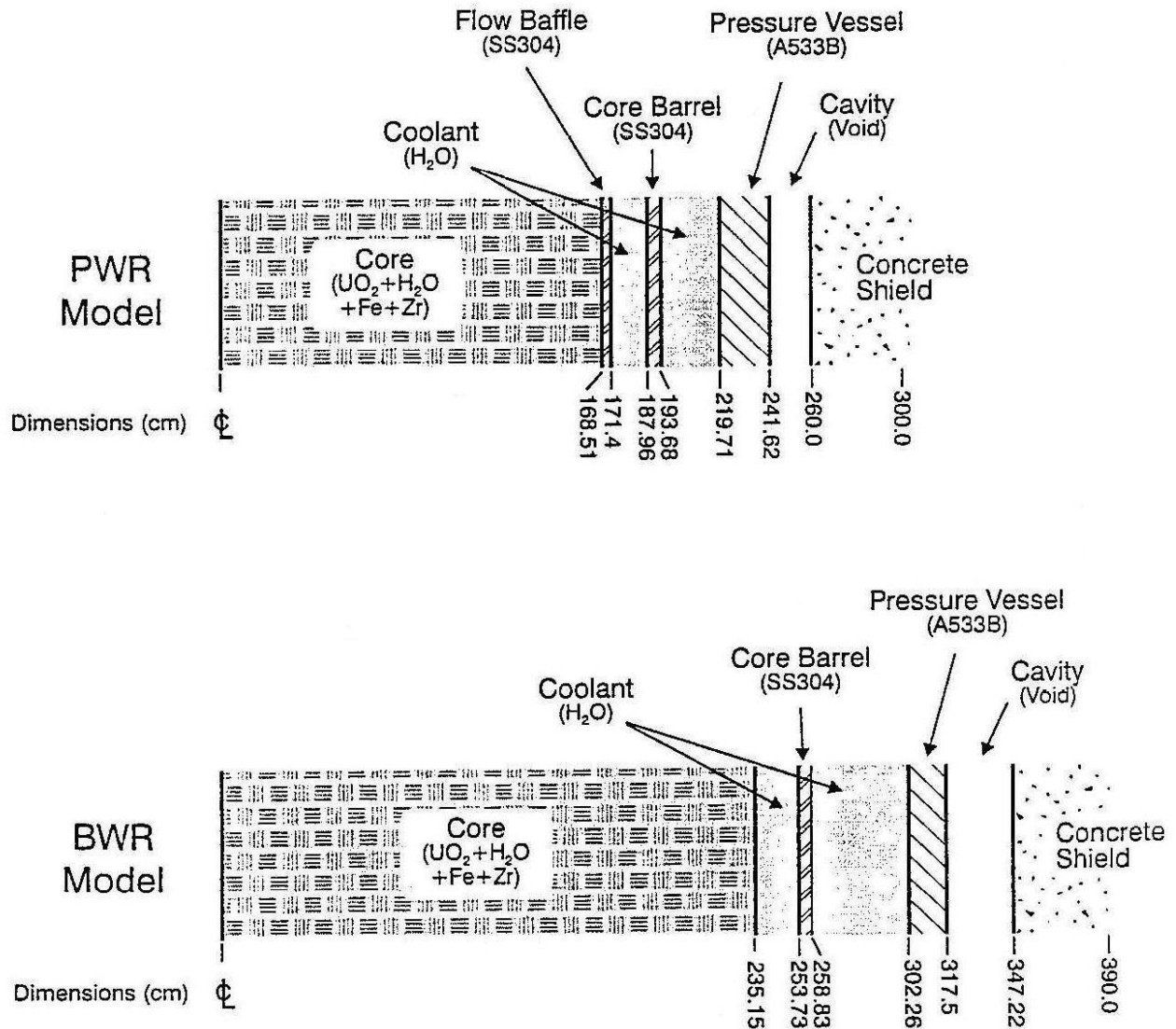
([a]) Figure taken from reference /8/.

TAB. 3.3

Atomic Densities[a] and Natural Isotopic Abundances Used in BWR and PWR Models
for Spectrum Calculations.

	HOMOGENEOUS CORES		COOLANT	
	BWR	**PWR**	**BWR**	**PWR**
Hydrogen	1.5354-2[b]	2.768-2	4.950-2	4.714-2
Oxygen	7.6770-3	1.384-2	2.475-2	2.357-2
Boron-10	0.0	2.466-6	0.0	4.200-6
Zirconium	5.7645-3	4.257-3		
Iron	2.0030-5	1.444-5		
U-235	1.2125-4	1.903-4		
U-238	5.3220-3	6.515-3		
Fuel Oxygen	1.0884-2	1.343-2		

	STEELS			CONCRETE	
	SS-304	**A533-B**			**Type 04**
Carbon	2.37-4	9.81-4	Hydrogen		7.77-3
Silicon	8.93-4	3.71-4	Carbon		1.15-4
Chromium	1.74-2	1.27-4	Oxygen		4.38-2
Manganese	1.52-3	1.12-3	Sodium		1.05-3
Iron	5.83-2	8.19-2	Magnesium		1.48-4
Nickel	8.55-3	4.44-4	Aluminum		2.39-3
			Silicon		1.58-2
			Potassium		6.93-4
			Calcium		2.92-3
			Iron		3.13-4

NATURAL ISOTOPIC ABUNDANCES (%)

Mg-24	78.99	Ca-40	96.941	Fe-54	5.90	Zr-90	51.45
Mg-25	10.00	Ca-42	0.647	Fe-56	91.72	Zr-91	11.22
Mg-26	11.01	Ca-43	0.135	Fe-57	2.10	Zr-92	17.15
		Ca-44	2.086	Fe-58	0.28	Zr-94	17.38
Si-28	92.23	Ca-46	0.004			Zr-96	2.80
Si-29	4.67	Ca-48	0.187	Ni-58	68.27		
Si-30	3.10			Ni-60	26.10		
		Cr-50	4.345	Ni-61	1.13		
		Cr-52	83.790	Ni-62	3.59		
K-39	93.2581	Cr-53	9.500	Ni-64	0.91		
K-40	0.0117	Cr-54	2.365				
K-41	6.7302						

(a) In units of $[\text{Atoms} \cdot b^{-1} \cdot cm^{-1}]$.

(b) Read as 1.535×10^{-2}.

FIG. 3.2

Comparison of Five BWR- and PWR-Specific Neutron Flux Spectra Calculated with
VITENDF70.BOLIB and Used to Generate BUGENDF70.BOLIB.
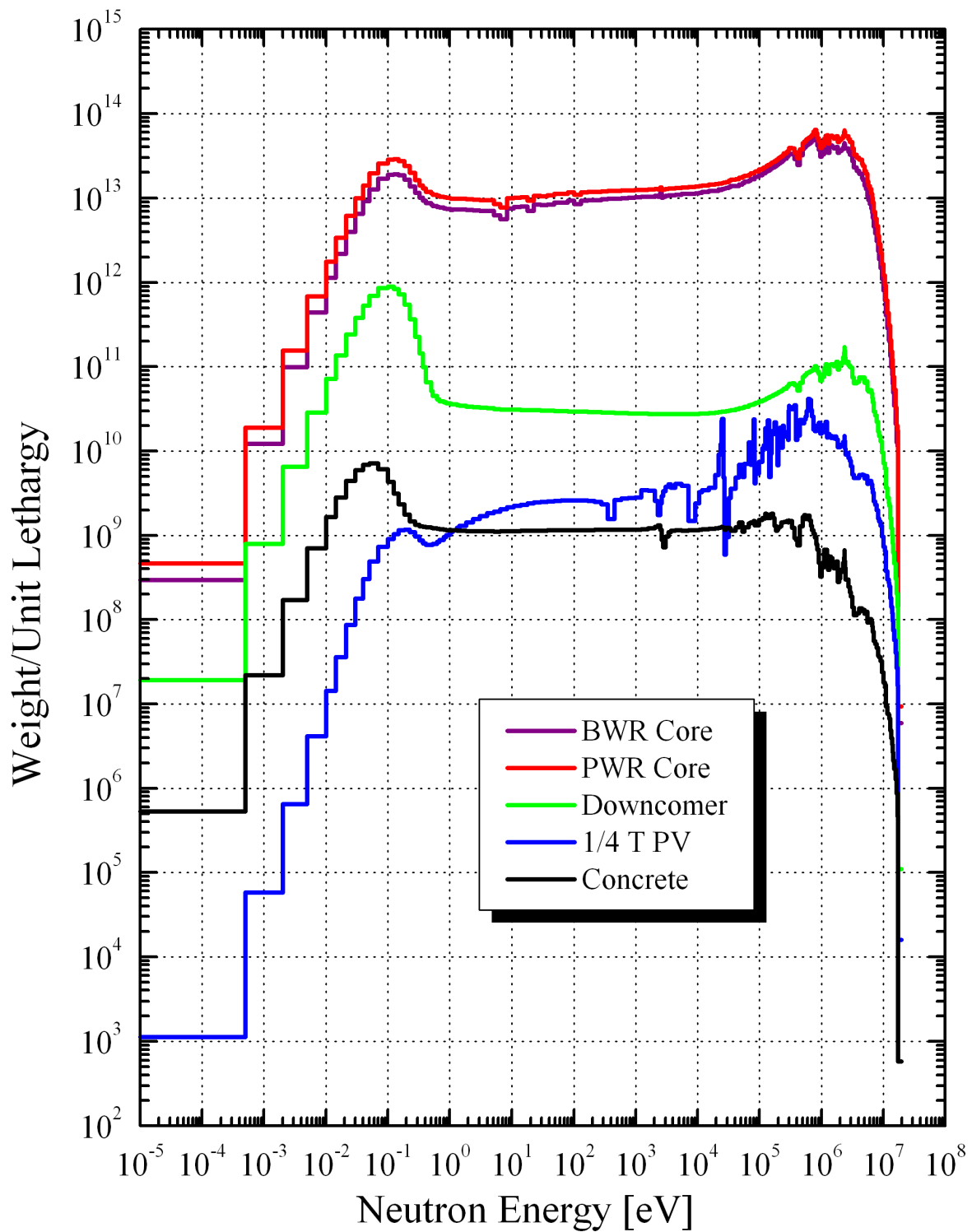
TAB. 3.4

Neutron Weighting Spectra from BWR/PWR Models for BUGENDF70.BOLIB.

Fine Group	BWR Core (Int#57)[a]	PWR Core (Int#37)	Downcomer (Int#69)	1/4T PV (Int#82)	Concrete (Int#106)
1	7.4025E+05	1.1650E+06	1.3681E+04	1.9829E+03	7.2935E+01
2	2.8344E+08	4.3732E+08	3.4919E+06	4.1242E+05	1.1586E+04
3	3.1895E+08	4.9820E+08	5.1200E+06	6.7457E+05	2.1681E+04
4	1.0540E+09	1.6352E+09	1.5493E+07	2.0141E+06	6.3887E+04
5	1.7353E+09	2.6794E+09	2.3249E+07	2.8672E+06	8.4498E+04
6	1.2326E+09	1.8982E+09	1.5689E+07	1.8794E+06	5.3272E+04
7	1.5787E+09	2.4352E+09	2.0222E+07	2.4034E+06	6.6498E+04
8	1.9934E+09	3.0749E+09	2.5329E+07	2.9897E+06	8.2439E+04
9	2.4647E+09	3.7946E+09	3.0381E+07	3.5418E+06	9.7269E+04
10	6.7907E+09	1.0424E+10	7.9651E+07	9.1051E+06	2.4442E+05
11	5.0385E+09	7.7162E+09	5.6097E+07	6.2370E+06	1.6663E+05
12	5.2022E+09	7.9780E+09	6.0622E+07	6.8338E+06	1.8527E+05
13	1.4365E+10	2.1795E+10	1.4048E+08	1.4959E+07	3.8123E+05
14	2.0358E+10	3.0758E+10	1.8456E+08	1.8735E+07	4.6472E+05
15	3.0729E+10	4.6948E+10	3.1251E+08	3.2336E+07	8.4425E+05
16	4.0256E+10	6.0828E+10	3.5875E+08	3.6853E+07	9.0218E+05
17	5.5066E+10	8.3265E+10	4.8258E+08	4.8687E+07	1.1949E+06
18	7.4769E+10	1.1330E+11	6.5470E+08	6.4793E+07	1.6253E+06
19	9.4934E+10	1.4287E+11	7.5753E+08	7.3318E+07	1.7966E+06
20	1.1927E+11	1.7846E+11	8.5884E+08	7.9709E+07	1.8592E+06
21	1.5511E+11	2.3299E+11	1.1213E+09	1.0024E+08	2.3023E+06
22	1.9220E+11	2.8797E+11	1.3209E+09	1.1607E+08	2.5888E+06
23	2.3228E+11	3.4643E+11	1.4707E+09	1.2590E+08	2.7770E+06
24	2.9495E+11	4.4205E+11	1.9059E+09	1.6005E+08	3.5921E+06
25	1.1263E+11	1.6888E+11	7.1923E+08	5.9867E+07	1.3883E+06
26	2.5109E+11	3.7728E+11	1.6246E+09	1.3483E+08	3.1291E+06
27	4.2900E+11	6.4242E+11	2.6630E+09	2.2099E+08	5.1044E+06
28	4.7310E+11	7.0154E+11	2.5633E+09	2.0711E+08	4.6616E+06
29	5.3208E+11	7.8528E+11	2.6366E+09	2.0468E+08	4.6376E+06
30	6.4626E+11	9.6078E+11	3.3740E+09	2.5206E+08	6.2127E+06
31	7.0482E+11	1.0394E+12	3.2420E+09	2.3668E+08	5.8666E+06
32	8.1178E+11	1.2003E+12	3.7464E+09	2.6199E+08	6.5265E+06
33	8.5524E+11	1.2552E+12	3.6537E+09	2.6398E+08	6.4087E+06
34	1.9380E+12	2.8285E+12	7.4621E+09	5.2357E+08	1.3612E+07
35	2.0777E+12	2.9845E+12	6.6285E+09	4.8303E+08	1.1927E+07
36	2.2946E+12	3.2600E+12	6.4338E+09	4.7984E+08	1.1327E+07
37	1.3610E+12	1.9340E+12	3.7556E+09	2.6389E+08	7.0245E+06
38	1.6108E+12	2.3018E+12	4.8742E+09	3.2914E+08	9.4467E+06
39	1.6242E+12	2.3025E+12	4.9815E+09	3.8294E+08	1.0614E+07
40	1.7505E+12	2.4736E+12	5.2546E+09	4.1381E+08	1.2350E+07
41	1.9177E+12	2.7026E+12	5.7523E+09	4.7559E+08	1.4992E+07
42	1.9309E+12	2.7058E+12	5.6134E+09	4.5358E+08	1.5056E+07
43	1.3379E+12	1.8736E+12	4.2366E+09	3.5808E+08	1.3768E+07
44	3.6298E+11	5.1074E+11	1.3052E+09	1.1056E+08	4.2041E+06
45	3.7055E+11	5.2120E+11	1.4330E+09	1.2531E+08	5.5247E+06
46	7.2864E+11	1.0174E+12	2.4624E+09	2.0268E+08	9.0900E+06
47	1.3276E+12	1.8267E+12	4.3032E+09	4.3066E+08	1.5808E+07
48	1.8676E+12	2.5477E+12	5.6674E+09	5.9945E+08	1.9770E+07
49	1.8159E+12	2.4620E+12	5.3213E+09	5.7117E+08	1.8312E+07

TAB. 3.4 Continued

Neutron Weighting Spectra from BWR/PWR Models for BUGENDF70.BOLIB.

Fine Group	BWR Core (Int#57)	PWR Core (Int#37)	Downcomer (Int#69)	1/4T PV (Int#82)	Concrete (Int#106)
50	1.7712E+12	2.3882E+12	5.1294E+09	5.5517E+08	1.5470E+07
51	1.7046E+12	2.2821E+12	4.6683E+09	5.5953E+08	1.4093E+07
52	2.0163E+12	2.6983E+12	5.7639E+09	6.8998E+08	2.4411E+07
53	1.9684E+12	2.6035E+12	5.4552E+09	7.4713E+08	2.2057E+07
54	1.9173E+12	2.5228E+12	5.1200E+09	7.0477E+08	2.0303E+07
55	2.0023E+12	2.6265E+12	5.3461E+09	7.4303E+08	2.4541E+07
56	2.0994E+12	2.7362E+12	5.3784E+09	7.1658E+08	2.7119E+07
57	2.0558E+12	2.6585E+12	5.2690E+09	7.7057E+08	2.8179E+07
58	1.7185E+12	2.2083E+12	4.1096E+09	7.9652E+08	1.9622E+07
59	2.1403E+12	2.7533E+12	5.2114E+09	7.2592E+08	2.8771E+07
60	2.1484E+12	2.7329E+12	5.3577E+09	1.1106E+09	3.4043E+07
61	1.9986E+12	2.5266E+12	4.8505E+09	1.0250E+09	3.3681E+07
62	3.2928E+12	4.1553E+12	7.5622E+09	1.6827E+09	4.1824E+07
63	1.2885E+12	1.6233E+12	2.8558E+09	6.7115E+08	1.3501E+07
64	2.1579E+12	2.6957E+12	4.7982E+09	1.2668E+09	2.7517E+07
65	2.0821E+12	2.5685E+12	4.4274E+09	8.9787E+08	3.3896E+07
66	2.4020E+12	2.9288E+12	4.8970E+09	9.9559E+08	3.9284E+07
67	2.6645E+12	3.2113E+12	5.0958E+09	8.6835E+08	4.7241E+07
68	2.5535E+12	3.0399E+12	4.7796E+09	8.9582E+08	5.2384E+07
69	2.3812E+12	2.8179E+12	4.5917E+09	1.2883E+09	5.7828E+07
70	2.2965E+12	2.7103E+12	4.4619E+09	1.5289E+09	6.4702E+07
71	2.2354E+12	2.6305E+12	4.4037E+09	2.0346E+09	7.5256E+07
72	2.1748E+12	2.5534E+12	4.2915E+09	2.0964E+09	8.5677E+07
73	2.1148E+12	2.4791E+12	4.2134E+09	1.4932E+09	8.6573E+07
74	2.0546E+12	2.4056E+12	4.0731E+09	1.2887E+09	7.8635E+07
75	1.9945E+12	2.3332E+12	3.9408E+09	1.1903E+09	8.6643E+07
76	1.9159E+12	2.2412E+12	3.7898E+09	1.0762E+09	8.2136E+07
77	3.1947E+12	3.7665E+12	6.4588E+09	2.3416E+09	1.2779E+08
78	2.4517E+12	2.9366E+12	5.1998E+09	1.5393E+09	7.5982E+07
79	1.4389E+12	1.7255E+12	3.0302E+09	6.7941E+08	4.4420E+07
80	1.4435E+12	1.7202E+12	3.0189E+09	1.7453E+09	4.6671E+07
81	3.2825E+12	3.8576E+12	6.4294E+09	3.1689E+09	1.2326E+08
82	3.3472E+12	3.8868E+12	6.3418E+09	3.3188E+09	1.4833E+08
83	3.7436E+11	4.3218E+11	7.0686E+08	4.0089E+08	1.7291E+07
84	1.3765E+11	1.5890E+11	2.6038E+08	1.3103E+08	6.4723E+06
85	2.8939E+11	3.3410E+11	5.4772E+08	2.1106E+08	1.3308E+07
86	7.8517E+11	9.0673E+11	1.4859E+09	3.4030E+08	3.5818E+07
87	1.5310E+12	1.7670E+12	2.8986E+09	7.3878E+08	7.4432E+07
88	2.9125E+12	3.3579E+12	5.5531E+09	1.6753E+09	1.4511E+08
89	1.3910E+12	1.6025E+12	2.6728E+09	9.9618E+08	7.1569E+07
90	1.3458E+12	1.5528E+12	2.6098E+09	5.5194E+08	7.1090E+07
91	1.3094E+12	1.5124E+12	2.5595E+09	9.4950E+08	6.3929E+07
92	1.2787E+12	1.4781E+12	2.5128E+09	5.6124E+08	6.2548E+07
93	1.2463E+12	1.4420E+12	2.4627E+09	3.3807E+08	5.9170E+07
94	1.2123E+12	1.4033E+12	2.4088E+09	7.8682E+08	5.6633E+07
95	1.1828E+12	1.3699E+12	2.3635E+09	1.0948E+09	6.5219E+07
96	1.1522E+12	1.3356E+12	2.3120E+09	8.7232E+08	8.0572E+07
97	1.1274E+12	1.3078E+12	2.2737E+09	7.9015E+08	9.1186E+07
98	1.0995E+12	1.2768E+12	2.2307E+09	4.6712E+08	8.8392E+07

TAB. 3.4 Continued

Neutron Weighting Spectra from BWR/PWR Models for BUGENDF70.BOLIB.

Fine Group	BWR Core (Int#57)	PWR Core (Int#37)	Downcomer (Int#69)	1/4T PV (Int#82)	Concrete (Int#106)
99	1.0762E+12	1.2506E+12	2.1904E+09	2.4586E+08	8.0999E+07
100	1.0484E+12	1.2186E+12	2.1435E+09	1.1577E+09	8.7814E+07
101	1.0256E+12	1.1927E+12	2.1063E+09	5.3322E+08	8.0647E+07
102	1.0036E+12	1.1679E+12	2.0703E+09	9.3379E+08	9.1300E+07
103	9.8044E+11	1.1418E+12	2.0304E+09	5.8425E+08	7.9655E+07
104	9.6292E+11	1.1223E+12	2.0014E+09	5.4552E+08	8.0802E+07
105	2.3257E+12	2.7150E+12	4.8712E+09	9.3612E+08	1.9055E+08
106	2.2150E+12	2.5934E+12	4.6988E+09	6.8193E+08	1.6574E+08
107	8.2918E+11	9.6431E+11	1.7429E+09	1.9615E+08	6.2988E+07
108	6.1907E+11	7.3185E+11	1.3524E+09	8.8029E+08	5.5013E+07
109	1.6596E+12	1.9364E+12	3.5290E+09	6.8452E+08	1.4179E+08
110	1.0548E+12	1.2482E+12	2.3137E+09	8.4874E+08	9.2748E+07
111	2.7528E+12	3.2174E+12	5.9301E+09	1.2309E+09	2.1758E+08
112	1.1311E+12	1.3292E+12	2.4706E+09	3.7243E+08	8.3527E+07
113	1.8220E+12	2.1529E+12	4.0400E+09	8.5631E+08	1.7870E+08
114	1.7431E+12	2.0797E+12	3.9484E+09	6.1176E+08	1.5677E+08
115	2.4629E+12	2.8747E+12	5.3922E+09	6.2555E+08	1.9862E+08
116	1.0157E+12	1.1949E+12	2.2672E+09	2.0745E+08	9.1546E+07
117	1.4688E+12	1.7299E+12	3.2920E+09	1.0548E+08	1.3643E+08
118	7.0453E+11	8.3382E+11	1.5980E+09	3.2175E+07	6.3577E+07
119	4.6046E+11	5.4382E+11	1.0518E+09	1.7269E+08	4.3907E+07
120	6.4522E+11	7.6127E+11	1.4763E+09	1.2049E+09	6.2143E+07
121	3.1605E+11	3.7628E+11	7.3564E+08	5.7570E+08	3.1617E+07
122	3.1412E+11	3.7486E+11	7.3123E+08	3.0336E+08	3.1203E+07
123	9.5770E+11	1.1290E+12	2.1815E+09	6.8175E+08	9.2597E+07
124	1.5343E+12	1.8373E+12	3.5964E+09	6.4926E+08	1.5176E+08
125	3.0795E+12	3.6361E+12	7.0884E+09	9.1839E+08	2.9877E+08
126	2.9748E+12	3.5358E+12	7.0057E+09	8.5943E+08	2.9525E+08
127	1.1683E+12	1.3928E+12	2.7841E+09	3.2580E+08	1.1708E+08
128	1.7235E+12	2.0641E+12	4.1556E+09	3.6134E+08	1.7262E+08
129	2.8168E+12	3.3900E+12	6.8869E+09	3.7098E+08	2.8972E+08
130	2.8211E+12	3.3656E+12	6.8714E+09	9.7704E+08	2.8522E+08
131	2.7564E+12	3.3118E+12	6.8639E+09	1.0207E+09	2.8834E+08
132	1.6357E+12	1.9700E+12	4.1202E+09	5.5460E+08	1.6989E+08
133	1.0920E+12	1.3134E+12	2.7502E+09	4.0239E+08	1.0925E+08
134	1.0751E+12	1.3002E+12	2.7518E+09	3.7764E+08	9.7316E+07
135	1.0480E+12	1.2835E+12	2.7537E+09	3.3342E+08	7.2631E+07
136	5.0889E+11	6.3085E+11	1.3763E+09	1.4598E+08	4.4271E+07
137	5.8229E+11	6.7608E+11	1.3764E+09	1.1990E+08	5.8126E+07
138	1.0474E+12	1.2724E+12	2.7545E+09	1.7415E+08	1.3097E+08
139	1.0786E+12	1.2995E+12	2.7647E+09	2.5730E+08	1.2594E+08
140	2.6055E+12	3.1734E+12	6.9350E+09	8.5098E+08	3.0503E+08
141	2.5966E+12	3.1630E+12	6.9602E+09	8.6365E+08	2.9589E+08
142	2.5920E+12	3.1596E+12	6.9873E+09	7.0385E+08	2.9220E+08
143	2.5431E+12	3.1156E+12	7.0265E+09	6.9478E+08	2.9132E+08
144	2.5105E+12	3.0858E+12	7.0657E+09	6.8514E+08	2.9110E+08
145	2.4829E+12	3.0622E+12	7.1059E+09	6.5879E+08	2.9104E+08
146	2.4619E+12	3.0460E+12	7.1387E+09	3.9369E+08	2.9048E+08
147	2.4376E+12	3.0268E+12	7.1951E+09	5.8702E+08	2.9046E+08

TAB. 3.4 Continued

Neutron Weighting Spectra from BWR/PWR Models for BUGENDF70.BOLIB.

Fine Group	BWR Core (Int#57)	PWR Core (Int#37)	Downcomer (Int#69)	1/4T PV (Int#82)	Concrete (Int#106)
148	2.3888E+12	2.9775E+12	7.2380E+09	6.2524E+08	2.9031E+08
149	2.3793E+12	2.9701E+12	7.2795E+09	6.4008E+08	2.9003E+08
150	2.3388E+12	2.9344E+12	7.3247E+09	6.4699E+08	2.8979E+08
151	2.1458E+12	2.7419E+12	7.3664E+09	6.4964E+08	2.8931E+08
152	2.3391E+12	2.9344E+12	7.4102E+09	6.4970E+08	2.8881E+08
153	2.2210E+12	2.8141E+12	7.4538E+09	6.4740E+08	2.8822E+08
154	2.2189E+12	2.8154E+12	7.4973E+09	6.4278E+08	2.8756E+08
155	2.1194E+12	2.7234E+12	7.5407E+09	6.3575E+08	2.8681E+08
156	2.0757E+12	2.6612E+12	7.5829E+09	6.2611E+08	2.8596E+08
157	2.0970E+12	2.6979E+12	7.6255E+09	6.1385E+08	2.8505E+08
158	1.7727E+12	2.3430E+12	7.6657E+09	5.9864E+08	2.8398E+08
159	1.9864E+12	2.5867E+12	7.7081E+09	5.8077E+08	2.8292E+08
160	1.9629E+12	2.5544E+12	7.7495E+09	5.5997E+08	2.8175E+08
161	1.9056E+12	2.4937E+12	7.7879E+09	5.3615E+08	2.8039E+08
162	1.4209E+12	1.9508E+12	7.8266E+09	5.0967E+08	2.7896E+08
163	1.5632E+12	2.1429E+12	7.8646E+09	4.8065E+08	2.7741E+08
164	1.7667E+12	2.3686E+12	8.0790E+09	4.5846E+08	2.8195E+08
165	1.7654E+12	2.3648E+12	8.1582E+09	4.2600E+08	2.7939E+08
166	1.7985E+12	2.4083E+12	8.2906E+09	3.9176E+08	2.8043E+08
167	1.8091E+12	2.4260E+12	8.4443E+09	3.5763E+08	2.8162E+08
168	1.8363E+12	2.4657E+12	8.6332E+09	3.2496E+08	2.8338E+08
169	7.8247E+11	1.0523E+12	3.7184E+09	1.2998E+08	1.2052E+08
170	1.0577E+12	1.4259E+12	5.1534E+09	1.6670E+08	1.6536E+08
171	2.9892E+11	4.0369E+11	1.4849E+09	4.5682E+07	4.7270E+07
172	2.7673E+11	3.7384E+11	1.3719E+09	4.1031E+07	4.3518E+07
173	2.8985E+11	3.9160E+11	1.4338E+09	4.1570E+07	4.5302E+07
174	9.8816E+11	1.3354E+12	4.8897E+09	1.3292E+08	1.5316E+08
175	6.9429E+11	9.3886E+11	3.4553E+09	8.6494E+07	1.0679E+08
176	1.2307E+12	1.6660E+12	6.2108E+09	1.4024E+08	1.8786E+08
177	6.9636E+11	9.4391E+11	3.6098E+09	7.3289E+07	1.0538E+08
178	1.3164E+12	1.7878E+12	7.2811E+09	1.2808E+08	1.9670E+08
179	5.1247E+11	6.9767E+11	3.1396E+09	4.7224E+07	7.5258E+07
180	1.6524E+12	2.2605E+12	1.2479E+10	1.4556E+08	2.3486E+08
181	1.1415E+12	1.5758E+12	1.2025E+10	9.7862E+07	1.5284E+08
182	1.2376E+12	1.7255E+12	1.7533E+10	1.0592E+08	1.5547E+08
183	1.9273E+12	2.7244E+12	3.7689E+10	1.6285E+08	2.2264E+08
184	2.7615E+12	3.9718E+12	7.3416E+10	2.1894E+08	2.9323E+08
185	3.2933E+12	4.8142E+12	1.0991E+11	2.3383E+08	3.5464E+08
186	3.7588E+12	5.5695E+12	1.4708E+11	2.3378E+08	4.7690E+08
187	3.4997E+12	5.2436E+12	1.5298E+11	1.8988E+08	5.7698E+08
188	4.2030E+12	6.3482E+12	1.9970E+11	2.0264E+08	9.6933E+08
189	5.9818E+12	9.0880E+12	3.0709E+11	2.6093E+08	2.1738E+09
190	4.2990E+12	6.5668E+12	2.3551E+11	1.6614E+08	2.4183E+09
191	2.0674E+12	3.1727E+12	1.1820E+11	6.8152E+07	1.5432E+09
192	1.8770E+12	2.8910E+12	1.1067E+11	5.1387E+07	1.7120E+09
193	1.4298E+12	2.2092E+12	8.6786E+10	3.0769E+07	1.5764E+09
194	8.1302E+11	1.2596E+12	5.0573E+10	1.3344E+07	1.0479E+09
195	4.2302E+11	6.5686E+11	2.6799E+10	5.3192E+06	6.1017E+08
196	3.0798E+11	4.7920E+11	1.9815E+10	2.8752E+06	4.8914E+08

TAB. 3.4 Continued

Neutron Weighting Spectra from BWR/PWR Models for BUGENDF70.BOLIB.

Fine Group	BWR Core (Int#57)	PWR Core (Int#37)	Downcomer (Int#69)	1/4T PV (Int#82)	Concrete (Int#106)
197	9.1598E+10	1.4284E+11	5.9694E+09	5.9519E+05	1.5778E+08
198	1.6924E+10	2.6443E+10	1.1073E+09	8.0640E+04	3.0367E+07
199	1.1616E+09	1.8146E+09	7.5093E+07	4.4219E+03	2.0797E+06

(ª) Spatial mesh interval number from BWR or PWR model.

FIG. 3.3

Comparison of Five BWR- and PWR-Specific Photon Flux Spectra Calculated with
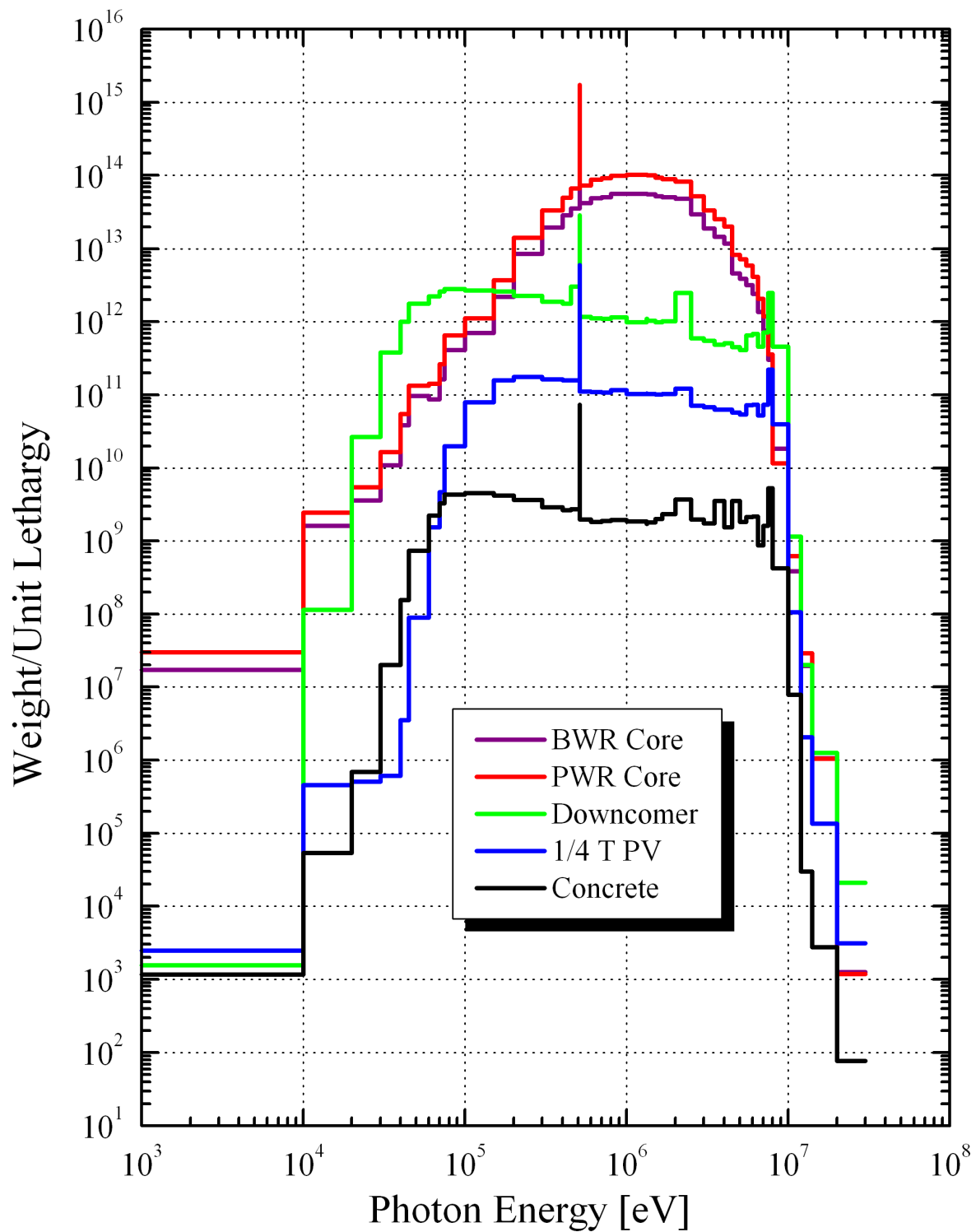VITENDF70.BOLIB and Used to Generate BUGENDF70.BOLIB.

TAB. 3.5

Photon Weighting Spectra from BWR/PWR Models for BUGENDF70.BOLIB.

Fine Group	BWR Core (Int#57)[a]	PWR Core (Int#37)	Downcomer (Int#69)	1/4T PV (Int#82)	Concrete (Int#106)
1	5.0831E+02	4.7579E+02	8.4498E+03	1.2571E+03	3.1569E+01
2	3.7694E+05	3.7200E+05	4.4996E+05	4.8458E+04	9.7961E+02
3	3.0268E+06	4.4810E+06	3.0968E+06	3.1593E+05	4.5927E+03
4	7.0618E+07	1.1422E+08	2.1045E+08	1.9307E+07	1.4377E+06
5	4.0943E+09	2.5728E+09	1.0245E+11	8.8636E+09	9.4421E+07
6	1.9559E+10	2.3127E+10	1.6051E+11	1.4383E+10	3.4197E+08
7	5.2925E+10	8.2212E+10	4.9206E+10	5.0388E+09	1.1117E+08
8	1.0202E+11	1.5424E+11	3.4018E+10	3.8711E+09	6.4852E+07
9	1.9498E+11	3.3225E+11	5.4957E+10	5.8388E+09	1.7398E+08
10	2.7683E+11	5.1668E+11	5.7308E+10	6.3157E+09	1.8695E+08
11	3.6997E+11	6.8588E+11	3.9351E+10	5.1544E+09	1.7388E+08
12	4.8631E+11	8.7940E+11	4.7738E+10	6.0964E+09	3.7584E+08
13	1.3960E+12	2.3874E+12	5.9884E+10	7.4714E+09	1.8088E+08
14	1.9476E+12	3.4046E+12	6.5392E+10	8.4386E+09	4.7368E+08
15	2.9568E+12	5.2194E+12	8.5624E+10	1.0517E+10	2.7088E+08
16	5.4447E+12	9.5553E+12	1.0798E+11	1.3083E+10	3.5783E+08
17	1.0914E+13	1.8747E+13	5.5500E+11	2.7052E+10	8.2641E+08
18	9.5225E+12	1.6400E+13	1.9002E+11	1.9322E+10	4.3624E+08
19	5.3084E+12	9.3930E+12	1.0054E+11	1.0354E+10	2.0389E+08
20	6.3114E+12	1.1238E+13	1.1785E+11	1.1682E+10	2.0200E+08
21	4.2783E+11	7.6895E+11	8.2900E+09	7.8064E+08	1.2838E+07
22	1.6108E+13	2.8830E+13	2.8454E+11	2.9384E+10	5.2851E+08
23	1.2663E+13	2.2108E+13	2.5647E+11	2.6136E+10	4.3557E+08
24	6.8405E+12	1.2188E+13	1.4764E+11	1.4399E+10	2.5304E+08
25	7.6465E+12	1.3453E+13	1.7390E+11	1.6908E+10	2.8238E+08
26	6.7136E+12	1.1628E+13	1.8504E+11	1.7764E+10	3.1078E+08
27	3.9479E+12	6.7828E+12	1.1279E+11	2.3433E+10	2.8651E+08
28	4.5021E+12	8.3713E+12	3.8318E+11	1.9838E+10	3.4318E+08
29	3.3960E+12	5.8816E+12	2.0955E+11	1.8771E+10	3.1205E+08
30	5.6062E+12	9.6218E+12	5.4689E+11	4.7279E+10	8.3220E+08
31	3.4419E+12	5.7994E+12	9.2405E+11	7.1730E+10	1.4764E+09
32	6.3753E+11	1.0640E+12	7.5135E+11	4.5429E+10	1.2139E+09
33	2.8459E+11	4.5112E+11	1.0959E+12	3.2370E+10	1.8432E+09
34	1.1787E+11	1.8856E+11	8.1626E+11	5.6941E+09	1.2420E+09
35	1.1243E+10	1.8074E+10	1.8049E+11	3.2265E+08	2.2553E+08
36	1.3505E+10	2.1843E+10	3.4414E+11	2.3976E+08	3.4320E+08
37	2.8056E+10	3.8614E+10	5.1195E+11	2.5808E+07	2.1135E+08
38	4.5300E+09	6.4259E+09	1.1805E+11	4.1726E+05	1.8241E+07
39	3.1269E+09	4.7548E+09	1.0912E+11	1.7638E+05	5.7579E+06
40	1.4622E+09	2.2175E+09	1.0788E+10	2.0522E+05	2.7980E+05
41	1.1260E+09	1.6856E+09	7.9699E+07	3.1495E+05	3.7504E+04
42	3.9691E+07	6.8888E+07	3.5751E+03	5.6930E+03	2.6790E+03

(a) Spatial mesh interval number from BWR or PWR model.

TAB. 3.6

Key Parameters for BWR and PWR Pin Cells in Self-Shielding Calculations.

	BWR	PWR
Inner radius clad [cm]	0.53213	0.41783
Outer radius clad [cm]	0.6134	0.47498
Outer radius cell [cm]	0.9174	0.71079
Region temperature [°K]		
Pellet	921	921
Clad	672	672
Moderator	551	583
Pellet nuclear density [Atoms·b^{-1}·cm^{-1}]		
U-235	4.959-4[a]	6.325-4
U-238	2.177-2	2.166-2
Oxygen	4.455-2	4.465-2
Moderator density [Atoms·b^{-1}·cm^{-1}]		
Hydrogen	2.475-2	4.714-2
Oxygen	1.2375-2	2.357-2
Boron-10	0.0	4.200-6
Zircaloy-4 density [Atoms·b^{-1}·cm^{-1}]		
Cromium	7.64-5	7.64-5
Iron	1.45-4	1.45-4
Nickel	8.77-4	8.77-4
Zirconium	4.27-2	4.27-2

([a]) Read as 4.959×10^{-4}.

TAB. 3.7

Atomic Densities[a] Used for Steel and Concrete Constituents in Self-Shielding Calculations.

	STEELS			**CONCRETE**	
	Stainless Steel	**Carbon Steel**			**Type 04**
Carbon	2.37-4[b]	9.81-4	Hydrogen		8.60-3
Chromium	1.74-2	1.27-4	Carbon		1.15-4
Manganese	1.52-3	1.12-3	Oxygen		4.33-2
Iron	5.83-2	8.19-2	Sodium		9.64-4
Nickel	8.55-3	4.44-4	Magnesium		1.24-4
			Aluminum		1.74-4
			Silicon		1.66-3
			Potassium		4.60-4
			Calcium		1.50-3
			Iron		3.45-4

(a) In units of $[Atoms \cdot b^{-1} \cdot cm^{-1}]$.
(b) Read as 2.37×10^{-4}.

3.4 - Processing Codes and Procedures

All the computational tools used to self-shield, temperature correct, and collapse the VITENDF70.BOLIB /5/ fine-group cross sections into the BUGENDF70.BOLIB broad-group cross sections were modules of the ENEA-Bologna 2007 Revision /19/ of the SCAMPI /18/ nuclear data processing system which was used on a Personal Computer (CPU INTEL Pentium III, 448 MB of RAM; f77 Absoft version 5.0 FORTRAN 77 compiler) with the Linux Red Hat 7.1 operating system. The names and a brief description of the primary SCAMPI modules used are given in TAB. 3.8.

The first part of the data processing procedure, dedicated to the generation of the BUGENDF70.BOLIB library, is given in FIG. 3.4 which diagrams the sequence of steps needed to select and self-shield specific sets of cross sections and to perform the BWR and PWR one-dimensional transport calculations addressed to obtain five neutron/photon flux weighting spectra, employed in the second part of the data processing procedure for a proper cross section collapsing. The complete input listings are given in APPENDIX A, in TAB. A.1, for the cross section self-shielding calculation and in TAB. A.2 and TAB. A.3, respectively for the calculation of the BWR and PWR neutron/photon flux weighting spectra.

Starting from the VITENDF70.BOLIB binary AMPX master library, it was necessary to perform one-dimensional transport calculations with the XSDRNPM module of SCAMPI, using proper fine-group working cross sections self-shielded with the BONAMI module, to obtain BWR- and PWR-specific neutron/photon flux weighting spectra to collapse the fine-group cross sections of the VITENDF70.BOLIB mother library. In particular, five different fine-group neutron/photon flux weighting spectra in the VITENDF70.BOLIB (or VITAMIN-B6 /8/) neutron and photon energy group structures were calculated (see also 3.3) with two XSDRNPM fixed source transport calculations simulating the simplified in-vessel and ex-vessel one-dimensional radial geometries and compositions at the reactor core midplanes, typical of the BWR and PWR models. The same fine-group neutron source spectrum data set (see APPENDIX A, TAB. A.2 and TAB. A.3) was introduced in both BWR and PWR spectrum calculations in the 31** volumetric source spectra array of XSDRNPM. These data, taken from the BUGLE-96 /8/ library user's manual (see APPENDIX A, TAB. A.6 and TAB. A.7), are normalized to the unity. The XNF normalization factor numerical value, in the 5** array of XSDRNPM, permits respectively to obtain the BWR and the PWR total neutron source (XNF = 7.46419E+17 in the BWR case and XNF = 7.2676E+17 in the PWR case). The calculation of these BWR- and PWR-specific neutron/photon flux weighting spectra permitted the problem-dependent collapsing of the VITENDF70.BOLIB cross sections in the second part of the data processing procedure. The complete detailed listings of the 199 group values of the neutron flux weighting spectra is reported in TAB. 3.4 and graphically represented in FIG. 3.2 while the corresponding 42 group photon spectra are respectively inserted in TAB. 3.5 and shown in FIG. 3.3.

The second part of the data processing procedure is diagrammed in FIG. 3.5. The starting point of this sequence is represented by the AMPX master library of fine-group self-shielded cross sections, previously calculated during the first part of the data processing procedure. Then the MALOCS module was used to perform the collapsing of the self-shielded cross sections from the VITENDF70.BOLIB neutron and photon fine-group energy structures into the respective broad-group energy structures of the BUGENDF70.BOLIB (or BUGLE-96 /8/)

library. The complete input listings for the six MALOCS cases are given in APPENDIX A, in TABs. A.4÷A.9.

In particular the five neutron/photon flux weighting spectra, previously calculated with the XSDRNPM module in the first part of the data processing procedure, were used in the MALOCS module to obtain six cross section sets of self-shielded cross sections which were included in the BUGENDF70.BOLIB library.

The ANISN /4/ code methodology was selected in the MALOCS module for removal of the upscatter transfer matrix in the thermal energy range.

As reported in the BUGLE-96 library user's manual, removing upscatter is done for purposes of economy, since it eliminates the need to perform outer (source) iterations during flux convergence. However, removing the upscatter terms does require a non-physical adjustment to the cross sections, since in reality, low-energy neutrons can increase their kinetic energy by scattering off low-Z nuclei such as hydrogen. For most shielding problems, this is an acceptable approximation because the leading shielding issues are frequently dominated by the transport of higher-energy neutrons and an accurate transport of the thermal neutrons is relatively unimportant.

The approach used for removing the upscatter terms in BUGENDF70.BOLIB and BUGLE-96 is the same as implemented in the ANISN code. Conceptually, the upscatter between two groups is set to zero and the downscatter is reduced by an equivalent amount to preserve the net transfer reaction rate between the two groups. The within-group scatter terms of both groups are increased by a corresponding amount to preserve the total scatter reaction rate.

While this yields an acceptable solution in most circumstances, it can cause the generation of negative downscatter terms if the upscatter is greater than the downscatter between two groups. This highlights the importance of using a particular cross section set only for the type of application for which it was intended.

A final collapsing sequence within the second part of the data processing procedure is diagrammed in FIG. 3.6 and its complete input listing is given in APPENDIX A, in TAB. A.10.

In particular, starting from the VITENDF70.BOLIB binary AMPX master library, a set of infinitely dilute cross sections at 300 °K, for the whole set of 183 materials contained in the VITENDF70.BOLIB fine-group library, were collapsed with the MALOCS module introducing in input the neutron/photon weighting spectrum for concrete, previously calculated during the first part of the data processing procedure. A set of infinitely dilute (not self-shielded) broad-group cross sections was generated and included in the BUGENDF70.BOLIB library. These data can be used for general purposes and for cases where the problem-specific self-shielded/weighted data sets are not appropriate.

In both the collapsing procedures, for self-shielded (see FIG. 3.5) and infinitely dilute (not self-shielded) (see FIG. 3.6) cross sections, the NITAWL and ALPO modules were used for format conversion, respectively from the AMPX master to the AMPX working format and from the AMPX working to the FIDO-ANISN /4/ format.

The complete BUGENDF70.BOLIB library actually contains three distinct types of data: 1) the infinitely dilute (not self-shielded) cross section set for all the 183 materials contained in the VITENDF70.BOLIB library, 2) six sets of self-shielded and energy weighted cross

sections for nuclides specific to BWR and PWR models, and 3) a set of response functions and KERMA factors.

TAB. 3.8

Modules of the SCAMPI Nuclear Data Processing System (ENEA-Bologna 2007 Revision)
Used to Produce the BUGENDF70.BOLIB Library.

Module	Function
AIM	Converts master libraries from binary-to-BCD (or vice-versa).
ALPO	Produces ANISN library from AMPX working library format.
AJAX	Merges and deletes nuclides from master libraries.
BONAMI	Performs interpolation on Bondarenko factors to self-shield reaction cross sections.
MALOCS	Collapses energy group structure of master library.
NITAWL	Converts master library to working library format.
XSDRNPM	Performs a one-dimensional discrete ordinates or diffusion theory calculation using cross sections in an AMPX working library. Performs spatial cross section weighting.

FIG. 3.4 Procedure for Calculating BWR- and PWR-Specific Flux Spectra
Using the VITENDF70.BOLIB Library.

FIG. 3.5 Procedure for Generating the BUGENDF70.BOLIB Library Self-Shielded and
Collapsed Cross Sections Using BWR- and PWR-Specific Flux Spectra.

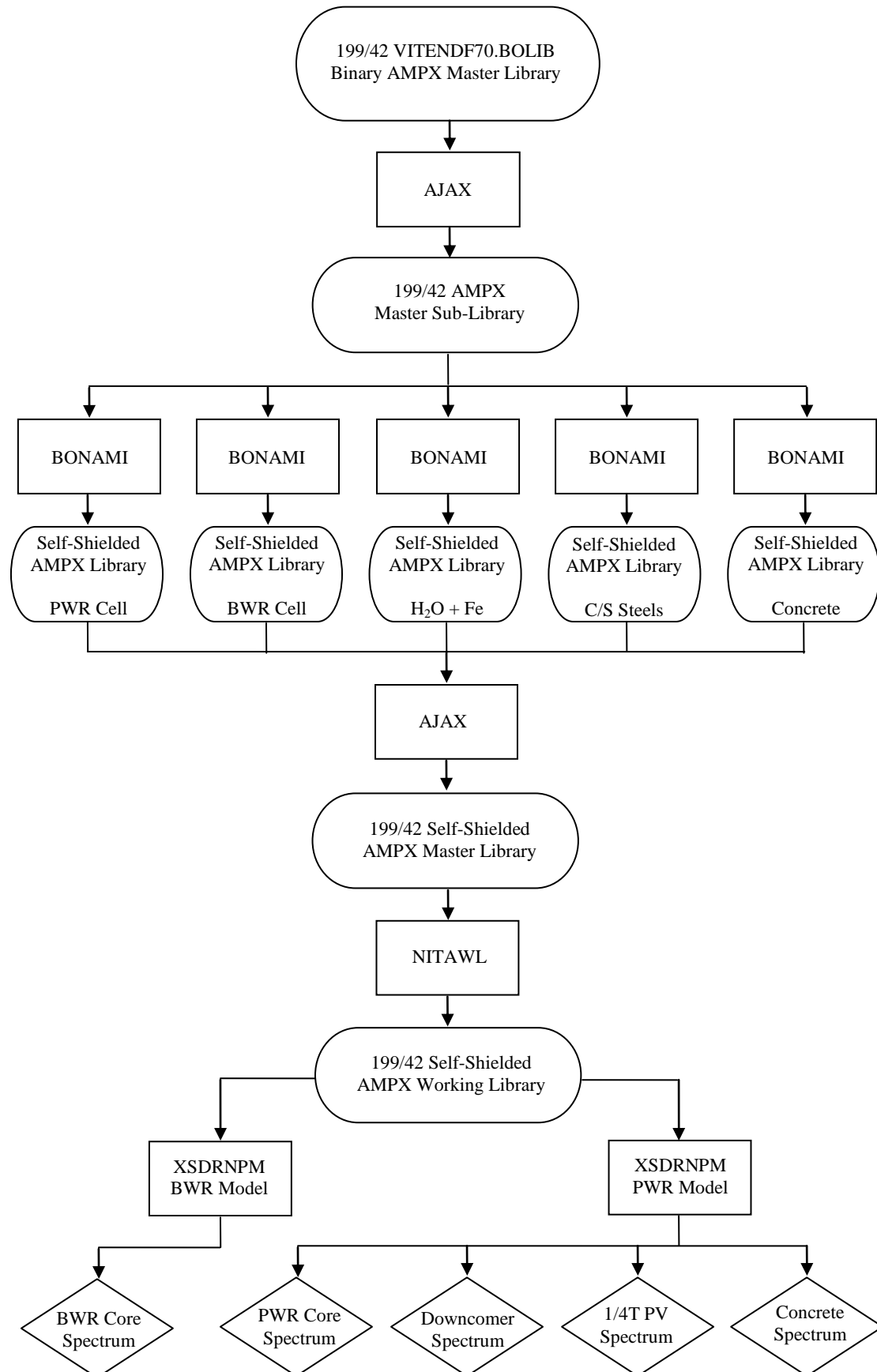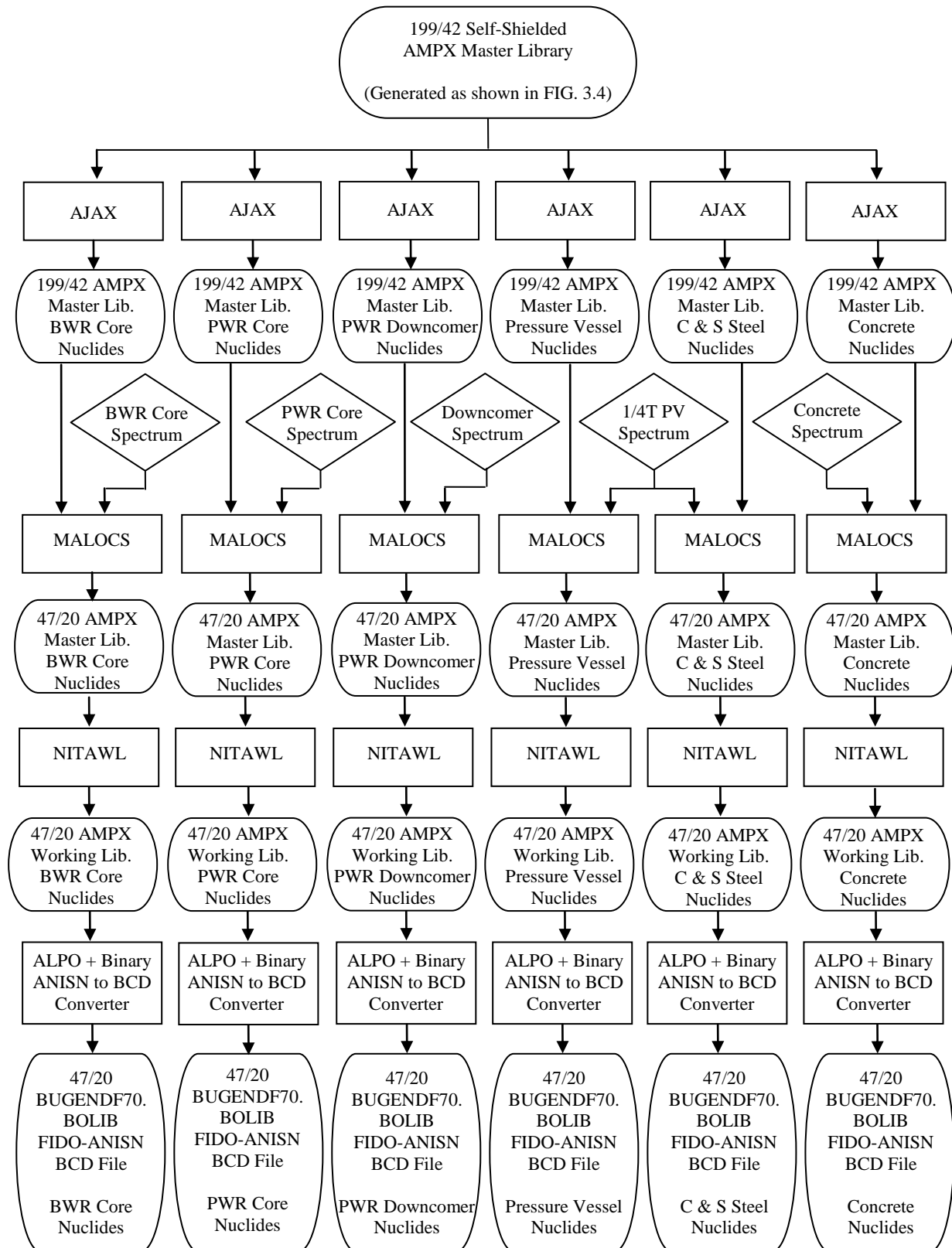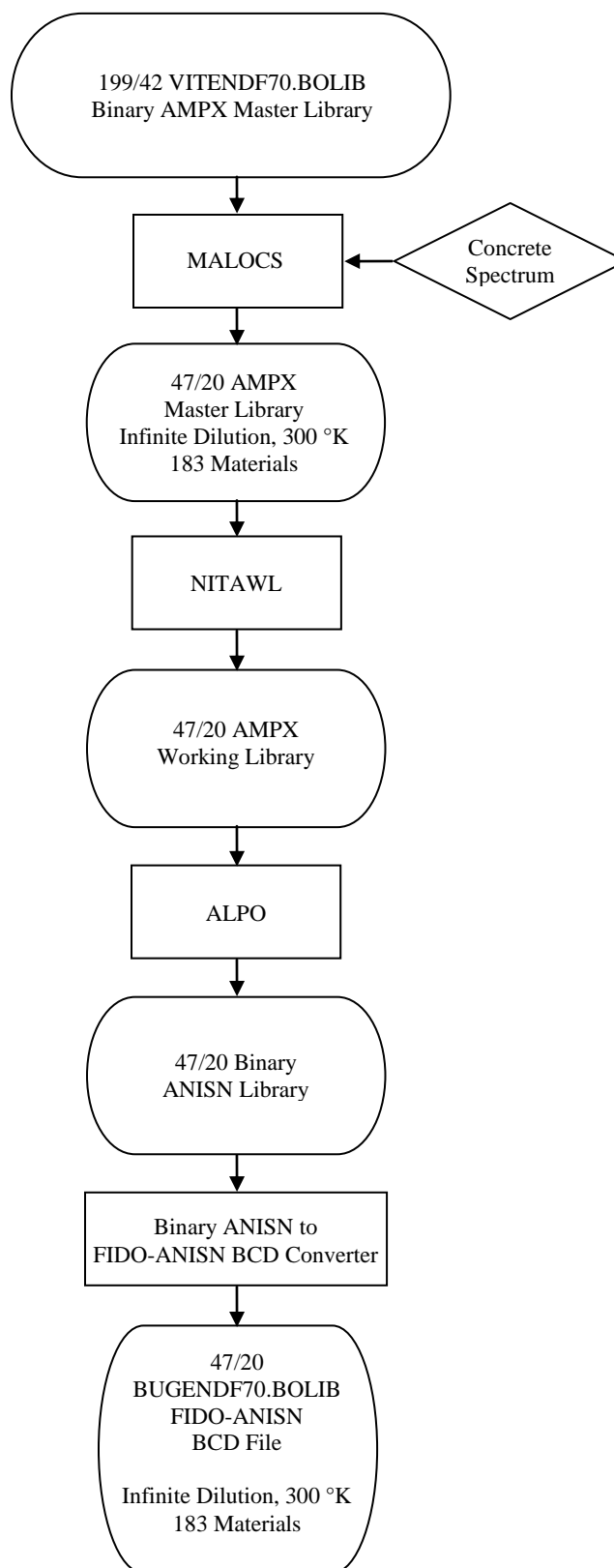FIG. 3.6 Procedure for Generating the BUGENDF70.BOLIB Library Infinitely Dilute and
Collapsed Cross Sections at 300 °K Using Concrete Flux Spectrum.

3.5 - Library Format and Content

The BUGENDF70.BOLIB library package consists of two major parts:

BUGENDF70.BOLIB   - including the self-shielded and infinitely dilute cross section sets;

BUGENDF70T.BOLIB - including the same cross sections as BUGENDF70.BOLIB with the thermal neutron upscattering cross sections retained in the thermal neutron energy region below 5.0435 eV.

The BUGENDF70.BOLIB broad-group data library package is available in ANISN /4/ card image format for direct use, e.g., in the ORNL deterministic radiation transport codes ANISN (1D), DORT (2D) and TORT (3D) included in the DOORS /35/ system or in the ORNL Monte Carlo code MORSE (3D) /58/.

For BUGENDF70.BOLIB, the Carlson table width is 67 (total number of 47 neutron + 20 photon groups) and the table length is 70 (total number of groups plus 3). For BUGENDF70T.BOLIB, the Carlson table width is 67 (total number of groups) and the table length is 74 (total number of groups plus 3 plus 4 upscatter groups).

In the BUGENDF70.BOLIB library, the cross section table positions 1 through 4 are defined as:

1. absorption cross section (σ_a);
2. fission cross section times the number of neutrons produced per fission ($\nu\sigma_f$);
3. total cross section (σ_t);
4. within-group scattering cross section ($\sigma_{g \to g}$).

These positions are followed by the standard downscatter transfer matrix. As mentioned in 3.4, the upscatter transfer matrix was removed using the ANISN methodology option in the MALOCS module of the ENEA-Bologna 2007 Revision /19/ of the SCAMPI /18/ system for the BUGENDF70.BOLIB library.
In the case of the BUGENDF70T.BOLIB library, the upscatter cross sections for the four lower energy neutron groups in each cross section matrix occupy the cross section table positions from 4 to 7 and consequently, in this case, the within-group scattering cross section ($\sigma_{g \to g}$) takes up the position 8.

Each cross section matrix (67×70 cross section data in BUGENDF70.BOLIB without upscatter and 67×74 in BUGENDF70T.BOLIB with upscatter) is preceded by a title card containing four integer parameters and an alpha/numeric descriptor. The integer parameters (4I6 format) include: 1) the number of columns in the cross section table (total number of energy groups), 2) the number of rows in the table (normally equals the number of columns plus 3), 3) a control code which is generally not used and 4) a unique identification (ANISN identifier or ANISN-ID) number. The 36-character descriptor field (36H format) can contain any pertinent information, but usually includes the order of the Legendre polynomial (P_ℓ) expansion of the scattering cross section matrix, the specific isotope or element and special processing treatments such as self-shielding or energy weighting in the specific BWR or PWR spatial region.

On the first step, the collapsing procedure outlined in FIG. 3.5 generated the self-shielded cross sections without thermal upscattering data for the BUGENDF70.BOLIB library. The collapsing procedure was then repeated for the six MALOCS cases to produce self-shielded cross sections with upscattering data. On this second step a new library with upscattering data, designated as BUGENDF70T.BOLIB, was created. For example, the input data listing referring to the processing job for collapsing PWR concrete nuclides with upscattering cross sections is given in APPENDIX A, TAB. A.11. The difference between the production of the collapsed data for BUGENDF70.BOLIB and BUGENDF70T.BOLIB depends on a single parameter (IOPT7) of the 3$$ array in the MALOCS module input: IOPT7=0 (no upscatter) for BUGENDF70.BOLIB or IOPT7=2 (with upscatter, according to the "ANISN" scheme) for BUGENDF70T.BOLIB.

Listings of the library contents for BUGENDF70.BOLIB are given in the TABs. 3.9 and 3.10. TAB. 3.9 lists all the 183 materials included in the "standard weighted" library, i.e. the collection of materials which were processed as infinitely dilute and collapsed using the neutron and photon flux spectra for concrete. TAB. 3.10 lists the "special weighted" library, i.e. the collection of materials which were processed with specific self-shielding compositions and collapsed using the BWR and PWR neutron and photon flux spectra. As in the case of the corresponding BUGLE-96 special weighted library, the self-shielded cross sections of the carbon steel constituents in the "Carbon & Stainless Steel" cross section set are the same as those included in the pressure vessel cross section set, designated as "PWR 1/4 T in Pressure Vessel".

In both TABs. 3.9 and 3.10, the range of the ANISN identifiers (ANISN-IDs) for each nuclide are for the P_0 through P_L components, where L=ℓ-max is either 5 or 7 (see 3.2).

TAB. 3.9

Nuclides in the BUGENDF70.BOLIB Library which are Infinitely Dilute
and Weighted with a Concrete Flux Spectrum.

Entry	Nuclide	ANISN-ID	Entry	Nuclide	ANISN-ID
1	Ag-107	1-6	49	Er-162	333-338
2	Ag-109	7-12	50	Er-164	339-344
3	Al-27	13-20	51	Er-166	345-350
4	Am-241	21-26	52	Er-167	351-356
5	Am-242	27-32	53	Er-168	357-362
6	Am-242m	33-38	54	Er-170	363-368
7	Am-243	39-44	55	Eu-151	369-374
8	Au-197	45-50	56	Eu-152	375-380
9	B-10	51-58	57	Eu-153	381-386
10	B-11	59-66	58	Eu-154	387-392
11	Ba-138	67-72	59	Eu-155	393-398
12	Be-9	73-80	60	F-19	399-406
13	Be-9 (Thermal)	81-88	61	Fe-54	407-414
14	Bi-209	89-94	62	Fe-56	415-422
15	C-nat	95-102	63	Fe-57	423-430
16	C-nat(Graphite)	103-110	64	Fe-58	431-438
17	Ca-40	111-118	65	Ga-69	439-444
18	Ca-42	119-126	66	Ga-71	445-450
19	Ca-43	127-134	67	Gd-152	451-456
20	Ca-44	135-142	68	Gd-154	457-462
21	Ca-46	143-150	69	Gd-155	463-468
22	Ca-48	151-158	70	Gd-156	469-474
23	Cd-106	159-164	71	Gd-157	475-480
24	Cd-108	165-170	72	Gd-158	481-486
25	Cd-110	171-176	73	Gd-160	487-492
26	Cd-111	177-182	74	H-1(H_2O)	493-500
27	Cd-112	183-188	75	H-1(CH_2)	501-508
28	Cd-113	189-194	76	H-1(ZrH)	509-516
29	Cd-114	195-200	77	H-2(D_2O)	517-524
30	Cd-115m	201-206	78	H-3	525-532
31	Cd-116	207-212	79	He-3	533-540
32	Cl-35	213-220	80	He-4	541-548
33	Cl-37	221-228	81	Hf-174	549-554
34	Cm-241	229-234	82	Hf-176	555-560
35	Cm-242	235-240	83	Hf-177	561-566
36	Cm-243	241-246	84	Hf-178	567-572
37	Cm-244	247-252	85	Hf-179	573-578
38	Cm-245	253-258	86	Hf-180	579-584
39	Cm-246	259-264	87	In-113	585-590
40	Cm-247	265-270	88	In-115	591-596
41	Cm-248	271-276	89	K-39	597-604
42	Co-59	277-284	90	K-40	605-612
43	Cr-50	285-292	91	K-41	613-620
44	Cr-52	293-300	92	Li-6	621-628
45	Cr-53	301-308	93	Li-7	629-636
46	Cr-54	309-316	94	Mg-24	637-644
47	Cu-63	317-324	95	Mg-25	645-652
48	Cu-65	325-332	96	Mg-26	653-660

TAB. 3.9 Continued

Nuclides in the BUGENDF70.BOLIB Library which are Infinitely Dilute
and Weighted with a Concrete Flux Spectrum.

Entry	Nuclide	ANISN-ID	Entry	Nuclide	ANISN-ID
97	Mn-55	661-668	141	Si-28	957-964
98	Mo-92	669-674	142	Si-29	965-972
99	Mo-94	675-680	143	Si-30	973-980
100	Mo-95	681-686	144	Sn-112	981-986
101	Mo-96	687-692	145	Sn-114	987-992
102	Mo-97	693-698	146	Sn-115	993-998
103	Mo-98	699-704	147	Sn-116	999-1004
104	Mo-100	705-710	148	Sn-117	1005-1010
105	N-14	711-718	149	Sn-118	1011-1016
106	N-15	719-726	150	Sn-119	1017-1022
107	Na-23	727-734	151	Sn-120	1023-1028
108	Nb-93	735-740	152	Sn-122	1029-1034
109	Ni-58	741-748	153	Sn-123	1035-1040
110	Ni-60	749-756	154	Sn-124	1041-1046
111	Ni-61	757-764	155	Sn-125	1047-1052
112	Ni-62	765-772	156	Sn-126	1053-1058
113	Ni-64	773-780	157	Ta-181	1059-1064
114	Np-237	781-786	158	Ta-182	1065-1070
115	Np-238	787-792	159	Th-230	1071-1076
116	Np-239	793-798	160	Th-232	1077-1082
117	O-16	799-806	161	Ti-46	1083-1090
118	O-17	807-814	162	Ti-47	1091-1098
119	P-31	815-822	163	Ti-48	1099-1106
120	Pa-231	823-828	164	Ti-49	1107-1114
121	Pa-233	829-834	165	Ti-50	1115-1122
122	Pb-204	835-840	166	U-232	1123-1128
123	Pb-206	841-846	167	U-233	1129-1134
124	Pb-207	847-852	168	U-234	1135-1140
125	Pb-208	853-858	169	U-235	1141-1146
126	Pu-236	859-864	170	U-236	1147-1152
127	Pu-237	865-870	171	U-237	1153-1158
128	Pu-238	871-876	172	U-238	1159-1164
129	Pu-239	877-882	173	V	1165-1172
130	Pu-240	883-888	174	W-182	1173-1178
131	Pu-241	889-894	175	W-183	1179-1184
132	Pu-242	895-900	176	W-184	1185-1190
133	Pu-243	901-906	177	W-186	1191-1196
134	Pu-244	907-912	178	Y-89	1197-1202
135	Re-185	913-918	179	Zr-90	1203-1208
136	Re-187	919-924	180	Zr-91	1209-1214
137	S-32	925-932	181	Zr-92	1215-1220
138	S-33	933-940	182	Zr-94	1221-1226
139	S-34	941-948	183	Zr-96	1227-1232
140	S-36	949-956			

TAB. 3.10

Nuclides in the BUGENDF70.BOLIB Library which are Self-Shielded and Weighted
with BWR- and PWR-Specific Flux Spectra.

Nuclide	ANISN-ID	Description
B-10	2001-2008	PWR Core Coolant
Cr-50	2009-2016	PWR Core Clad
Cr-52	2017-2024	PWR Core Clad
Cr-53	2025-2032	PWR Core Clad
Cr-54	2033-2040	PWR Core Clad
Fe-54	2041-2048	PWR Core Clad
Fe-56	2049-2056	PWR Core Clad
Fe-57	2057-2064	PWR Core Clad
Fe-58	2065-2072	PWR Core Clad
H-1(H_2O)	2073-2080	PWR Core Coolant
Ni-58	2081-2088	PWR Core Clad
Ni-60	2089-2096	PWR Core Clad
Ni-61	2097-2104	PWR Core Clad
Ni-62	2105-2112	PWR Core Clad
Ni-64	2113-2120	PWR Core Clad
O-16	2121-2128	PWR Core Coolant
O-16	2129-2136	PWR Core Fuel
U-235	2137-2142	PWR Core Fuel
U-238	2143-2148	PWR Core Fuel
Zr-90	2149-2154	PWR Core Clad
Zr-91	2155-2160	PWR Core Clad
Zr-92	2161-2166	PWR Core Clad
Zr-94	2167-2172	PWR Core Clad
Zr-96	2173-2178	PWR Core Clad
H-1(H_2O)	3001-3008	PWR Downcomer
O-16	3009-3016	PWR Downcomer
C-nat	3017-3024	PWR Downcomer
Cr-50	3025-3032	PWR Downcomer
Cr-52	3033-3040	PWR Downcomer
Cr-53	3041-3048	PWR Downcomer
Cr-54	3049-3056	PWR Downcomer
Fe-54	3057-3064	PWR Downcomer
Fe-56	3065-3072	PWR Downcomer
Fe-57	3073-3080	PWR Downcomer
Fe-58	3081-3088	PWR Downcomer
Mn-55	3089-3096	PWR Downcomer
Ni-58	3097-3104	PWR Downcomer
Ni-60	3105-3112	PWR Downcomer
Ni-61	3113-3120	PWR Downcomer
Ni-62	3121-3128	PWR Downcomer
Ni-64	3129-3136	PWR Downcomer

TAB. 3.10 Continued

Nuclides in the BUGENDF70.BOLIB Library which are Self-Shielded and Weighted
with BWR- and PWR-Specific Flux Spectra.

Nuclide	ANISN-ID	Description
C-nat	4001-4008	PWR 1/4 T in Pressure Vessel
Cr-50	4009-4016	PWR 1/4 T in Pressure Vessel
Cr-52	4017-4024	PWR 1/4 T in Pressure Vessel
Cr-53	4025-4032	PWR 1/4 T in Pressure Vessel
Cr-54	4033-4040	PWR 1/4 T in Pressure Vessel
Fe-54	4041-4048	PWR 1/4 T in Pressure Vessel
Fe-56	4049-4056	PWR 1/4 T in Pressure Vessel
Fe-57	4057-4064	PWR 1/4 T in Pressure Vessel
Fe-58	4065-4072	PWR 1/4 T in Pressure Vessel
Mn-55	4073-4080	PWR 1/4 T in Pressure Vessel
Ni-58	4081-4088	PWR 1/4 T in Pressure Vessel
Ni-60	4089-4096	PWR 1/4 T in Pressure Vessel
Ni-61	4097-4104	PWR 1/4 T in Pressure Vessel
Ni-62	4105-4112	PWR 1/4 T in Pressure Vessel
Ni-64	4113-4120	PWR 1/4 T in Pressure Vessel
Al-27	5001-5008	Concrete Type 04
C-nat	5009-5016	Concrete Type 04
Ca-40	5017-5024	Concrete Type 04
Ca-42	5025-5032	Concrete Type 04
Ca-43	5033-5040	Concrete Type 04
Ca-44	5041-5048	Concrete Type 04
Ca-46	5049-5056	Concrete Type 04
Ca-48	5057-5064	Concrete Type 04
Fe-54	5065-5072	Concrete Type 04
Fe-56	5073-5080	Concrete Type 04
Fe-57	5081-5088	Concrete Type 04
Fe-58	5089-5096	Concrete Type 04
H-1(H_2O)	5097-5104	Concrete Type 04
K-39	5105-5112	Concrete Type 04
K-40	5113-5120	Concrete Type 04
K-41	5121-5128	Concrete Type 04
Mg-24	5129-5136	Concrete Type 04
Mg-25	5137-5144	Concrete Type 04
Mg-26	5145-5152	Concrete Type 04
Na-23	5153-5160	Concrete Type 04
O-16	5161-5168	Concrete Type 04
Si-28	5169-5176	Concrete Type 04
Si-29	5177-5184	Concrete Type 04
Si-30	5185-5192	Concrete Type 04

TAB. 3.10 Continued

Nuclides in the BUGENDF70.BOLIB Library which are Self-Shielded and Weighted
with BWR- and PWR-Specific Flux Spectra.

Nuclide	ANISN-ID	Description
C-nat	6001-6008	Carbon Steel
C-nat	6009-6016	Stainless Steel
Cr-50	6017-6024	Carbon Steel
Cr-50	6025-6032	Stainless Steel
Cr-52	6033-6040	Carbon Steel
Cr-52	6041-6048	Stainless Steel
Cr-53	6049-6056	Carbon Steel
Cr-53	6057-6064	Stainless Steel
Cr-54	6065-6072	Carbon Steel
Cr-54	6073-6080	Stainless Steel
Fe-54	6081-6088	Carbon Steel
Fe-54	6089-6096	Stainless Steel
Fe-56	6097-6104	Carbon Steel
Fe-56	6105-6112	Stainless Steel
Fe-57	6113-6120	Carbon Steel
Fe-57	6121-6128	Stainless Steel
Fe-58	6129-6136	Carbon Steel
Fe-58	6137-6144	Stainless Steel
Mn-55	6145-6152	Carbon Steel
Mn-55	6153-6160	Stainless Steel
Ni-58	6161-6168	Carbon Steel
Ni-58	6169-6176	Stainless Steel
Ni-60	6177-6184	Carbon Steel
Ni-60	6185-6192	Stainless Steel
Ni-61	6193-6200	Carbon Steel
Ni-61	6201-6208	Stainless Steel
Ni-62	6209-6216	Carbon Steel
Ni-62	6217-6224	Stainless Steel
Ni-64	6225-6232	Carbon Steel
Ni-64	6233-6240	Stainless Steel
Fe-54	7001-7008	BWR Core Clad
Fe-56	7009-7016	BWR Core Clad
Fe-57	7017-7024	BWR Core Clad
Fe-58	7025-7032	BWR Core Clad
H-1(H_2O)	7033-7040	BWR Core Coolant
O-16	7041-7048	BWR Core Coolant
O-16	7049-7056	BWR Core Fuel
U-235	7057-7062	BWR Core Fuel
U-238	7063-7068	BWR Core Fuel
Zr-90	7069-7074	BWR Core Clad
Zr-91	7075-7080	BWR Core Clad
Zr-92	7081-7086	BWR Core Clad
Zr-94	7087-7092	BWR Core Clad
Zr-96	7093-7098	BWR Core Clad

3.6 - Response Functions

The package of the BUGENDF70.BOLIB library contains many response functions in the neutron and photon energy group structures of the library. Following the designation adopted also in the BUGLE-96 /8/ library package, it is underlined that, for the sake of simplicity, the term "response functions" is intended here not only for the group dosimeter cross sections or the group KERMA factors available in the package, which can be properly considered in a strict physical sense as "response functions". This denomination is in fact also extended to the group total neutron fission spectra and to the group total neutrons per fission as well as to useful general neutron data as, for example, the neutron group energy boundaries, the midpoint values of the energy groups, etc.. In particular the following four types of broad-group response functions are included in the package.

1. General neutron response functions in the BUGENDF70.BOLIB 47 neutron energy group structure (see TAB. 3.11): group energy boundaries, group energy intervals (widths), group lethargy intervals (widths), midpoint values of the energy groups (E-mid), square-roots of the E-mid values and, finally, the multiplicative factors of the group neutron fluxes to obtain the total neutron flux, the neutron fluxes above 1.0 MeV and 0.1 MeV and the neutron flux below 0.414 eV.

2. Neutron fission spectra (χ) in the BUGENDF70.BOLIB 47 neutron energy group structure are included (see TAB. 3.11) for only 34 fissionable nuclides since the neutron fission spectrum of Am-242 cannot be calculated because MF=4, MT=18 is missing in the corresponding ENDF/B-VII.0 evaluated data file. For each fissionable nuclide, when the delayed component (MF=5, MT=455) of the neutron fission spectrum is available in the corresponding evaluated data file, the total (prompt + delayed) neutron fission spectrum is included, otherwise only the prompt component is given.
Total (prompt + delayed) neutrons per fission (ν) in the BUGENDF70.BOLIB 47 neutron energy group structure are included for 35 fissionable nuclides (see TAB. 3.12).

3. Dosimetry cross sections for 71 nuclear reactions, processed (see /55/) from the IAEA International Reactor Dosimetry File /54/ (IRDF-2002) in the BUGENDF70.BOLIB 47 neutron energy group structure.

4. Total neutron and photon KERMA factors for 183 materials (all the materials available in the library) in the BUGENDF70.BOLIB 47 neutron and 20 photon energy group structures (see respectively TAB. 3.1 and TAB. 3.2).

The list of the available general neutron response functions is reported in TAB. 3.11 together with the list of the fissionable nuclides for which the neutron fission spectra are included in the package. The corresponding numerical values are reported in APPENDIX B, TAB. B.1, for the general neutron response functions and in APPENDIX C, TAB. C.1, for the total (prompt + delayed) or prompt neutron fission spectra.

Concerning the total neutron fission spectra, it is underlined that the ENEA-Bologna 2007 Revision /19/ of the SCAMPI /18/ nuclear data processing system permitted to process (see the procedure described in 2.8) delayed neutron fission spectra to obtain the total (prompt +

delayed) neutron fission spectra (χ) in the VITENDF70.BOLIB /5/ 199 neutron fine-group energy structure (see TAB. 2.4).

The normalized fine-group (total or prompt) neutron fission spectra (χ) were calculated with the procedure described in 2.8, using the BONAMI, NITAWL and ICE modules. For example, a 199 group graphical representation of the U-235, U-238 and Pu-239 total neutron fission spectra is shown in FIG. 2.5 while the total neutron fission spectrum fine-group χ values for U-235, U-238 and Pu-239 are listed in TAB. 2.10.

Then the fine-group χ values were summed up to obtain the corresponding broad-group χ value and this operation was obviously repeated for each broad-group.

In this procedure the small contribution of the first fine-group between 17.332 MeV and 19.640 MeV (see TAB. 2.4 and TAB. 3.1) was neglected and so a new normalization to one neutron per fission of the broad-group χ values was needed.

It is worth of note that the fissionable nuclides included in the BUGLE-96 library contain only the prompt neutron fission spectrum. Differently from the previous case, the total neutron fission spectrum was obtained for all the fissionable nuclides included in the BUGENDF70.BOLIB library which have both prompt and delayed neutron fission spectra in the corresponding ENDF/B-VII.0 evaluated data files.

Total (prompt + delayed) neutrons per fission (ν) data are available for the 35 fissionable nuclides (TAB. 3.12) included in the BUGENDF70.BOLIB library. These data were obtained through collapsing of the corresponding data contained in the VITENDF70.BOLIB library, processed at the infinite dilution background cross section and at the temperature of 300 °K. The data collapsing was performed with the MALOCS module of the ENEA-Bologna Revision 2007 of the SCAMPI nuclear data processing system, using alternatively the "flat weighting" neutron spectrum (see APPENDIX D, TAB. D.1) and the "1/4 T PV weighting" neutron spectrum (see APPENDIX E, TAB. E.1).

The whole set of point-wise dosimetry cross sections for 71 nuclear reactions (see TAB. 3.13), included in IRDF-2002, was processed through the GROUPIE program of the PREPRO 2007 /64/ nuclear data processing system into the BUGENDF70.BOLIB 47 neutron energy group structure.

The collapsed group values of these dosimeter cross sections, obtained using the flat weighting neutron spectrum and the 1/4 T PV weighting neutron spectrum in the VITENDF70.BOLIB 199 neutron fine-group energy structure, are respectively reported in APPENDIX F, TAB. F.1, and in APPENDIX G, TAB. G.1.

The KERMA factors for neutrons and photons were originally generated respectively with the HEATR and GAMINR modules of the NJOY-99.259 /17/ system in the VITENDF70.BOLIB neutron and photon fine-group energy structures with 199 neutron groups and 42 photon groups. These data, contained in the VITENDF70.BOLIB library and processed at the infinite dilution background cross section and at the temperature of 300 °K, were then collapsed with the MALOCS module of the ENEA-Bologna 2007 Revision /19/ of the SCAMPI /18/ system, into the 47 neutron and 20 photon broad-group energy structures of the BUGENDF70.BOLIB library.

The neutron and photon weighting spectra for concrete (see respectively TAB. 3.4 and TAB. 3.5), calculated in a PWR biological shield, were used to perform the collapsing of the neutron and photon KERMA factors.

The neutron KERMA factors were obtained for all the 183 nuclides (see TAB. 3.14) contained in the VITENDF70.BOLIB library (see TAB. 2.1) and in the infinite dilution cross section set of the BUGENDF70.BOLIB library (see TAB. 3.9).

In TAB. 3.14, 15 nuclides (labelled with *) with negative group values of the neutron KERMA factors and 56 nuclides (labelled with #) without photon production data (see also TAB. 2.1) are indicated. If the processed files of ENDF/B-VII.0 nuclides presented one or more negative neutron KERMA factor group values, all the corresponding 47 neutron KERMA factor group values were set to zero, following the same procedure adopted for the BUGLE-96 library.

Anyway, the generation of negative group values for the neutron KERMA factors of the 15 previously cited isotopes is almost certainly due to problems in the corresponding ENDF/B-VII.0 evaluated data files rather than in their data processing.

Moreover it is important to note that, in total reactor power and heating calculations, the results can be heavily affected by the lack of photon production data in some important ENDF/B-VII.0 evaluated data files. Concerning this, it is recommended to check carefully if the ENDF/B-VII.0 data files of the nuclides involved in the specific calculations include photon production data, as it is easily possible to verify in TAB. 3.14.

The elements for which the photon KERMA factors are available are listed in TAB. 3.15: the photon KERMA factors are the same for all the isotopes of each element.

TAB. 3.11

Response Functions Included with BUGENDF70.BOLIB.
General Neutron Responses and Total (Prompt + Delayed) Neutron Fission Spectra (χ).

```
    General Neutron Response                 General Neutron Response


 1  Group upper energy [MeV]          6  Total neutron flux
 2  Group energy width [MeV]          7  E > 1.0 MeV neutron flux
 3  Group lethargy width              8  E > 0.1 MeV neutron flux
 4  Midpoint energy (E-mid) [MeV]     9  E < 0.414 eV neutron flux
 5  Square-root (E-mid) [MeV^{1/2}]
```

```
    Total Fission Spectrum (χ)               Total Fission Spectrum (χ)


 1  Th-230 Fission spectrum (χ)*      19  Pu-240  Fission spectrum (χ)
 2  Th-232 Fission spectrum (χ)       20  Pu-241  Fission spectrum (χ)
 3  Pa-231 Fission spectrum (χ)       21  Pu-242  Fission spectrum (χ)
 4  Pa-233 Fission spectrum (χ)       22  Pu-243  Fission spectrum (χ)*
 5  U-232  Fission spectrum (χ)*      23  Pu-244  Fission spectrum (χ)*
 6  U-233  Fission spectrum (χ)       24  Am-241  Fission spectrum (χ)
 7  U-234  Fission spectrum (χ)       25  Am-242  Fission spectrum (χ)#
 8  U-235  Fission spectrum (χ)       26  Am-242m Fission spectrum (χ)*
 9  U-236  Fission spectrum (χ)       27  Am-243  Fission spectrum (χ)
10  U-237  Fission spectrum (χ)       28  Cm-241  Fission spectrum (χ)*
11  U-238  Fission spectrum (χ)       29  Cm-242  Fission spectrum (χ)
12  Np-237 Fission spectrum (χ)       30  Cm-243  Fission spectrum (χ)*
13  Np-238 Fission spectrum (χ)*      31  Cm-244  Fission spectrum (χ)*
14  Np-239 Fission spectrum (χ)*      32  Cm-245  Fission spectrum (χ)
15  Pu-236 Fission spectrum (χ)*      33  Cm-246  Fission spectrum (χ)*
16  Pu-237 Fission spectrum (χ)*      34  Cm-247  Fission spectrum (χ)*
17  Pu-238 Fission spectrum (χ)       35  Cm-248  Fission spectrum (χ)*
18  Pu-239 Fission spectrum (χ)
```

(*) The neutron fission spectrum contains only the prompt component
    because the delayed component (MF=5, MT=455) is not available in the
    corresponding ENDF/B-VII.0 evaluated data file.
(#) The neutron fission spectrum of Am-242 cannot be calculated by NJOY
    because MF=4, MT=18 is missing in the corresponding ENDF/B-VII.0
    evaluated data file.

TAB. 3.12

Response Functions Included with BUGENDF70.BOLIB.
Total (Prompt + Delayed) Neutrons per Fission (ν) Collapsed Using
Flat Weighting and 1/4 T PV Weighting.

```
            Neutrons per Fission (ν)                Neutrons per Fission (ν)


   1   Th-230 Neutrons per fission (ν)   19  Pu-240  Neutrons per fission (ν)
   2   Th-232 Neutrons per fission (ν)   20  Pu-241  Neutrons per fission (ν)
   3   Pa-231 Neutrons per fission (ν)   21  Pu-242  Neutrons per fission (ν)
   4   Pa-233 Neutrons per fission (ν)   22  Pu-243  Neutrons per fission (ν)
   5   U-232  Neutrons per fission (ν)   23  Pu-244  Neutrons per fission (ν)
   6   U-233  Neutrons per fission (ν)   24  Am-241  Neutrons per fission (ν)
   7   U-234  Neutrons per fission (ν)   25  Am-242  Neutrons per fission (ν)
   8   U-235  Neutrons per fission (ν)   26  Am-242m Neutrons per fission (ν)
   9   U-236  Neutrons per fission (ν)   27  Am-243  Neutrons per fission (ν)
  10   U-237  Neutrons per fission (ν)   28  Cm-241  Neutrons per fission (ν)
  11   U-238  Neutrons per fission (ν)   29  Cm-242  Neutrons per fission (ν)
  12   Np-237 Neutrons per fission (ν)   30  Cm-243  Neutrons per fission (ν)
  13   Np-238 Neutrons per fission (ν)   31  Cm-244  Neutrons per fission (ν)
  14   Np-239 Neutrons per fission (ν)   32  Cm-245  Neutrons per fission (ν)
  15   Pu-236 Neutrons per fission (ν)   33  Cm-246  Neutrons per fission (ν)
  16   Pu-237 Neutrons per fission (ν)   34  Cm-247  Neutrons per fission (ν)
  17   Pu-238 Neutrons per fission (ν)   35  Cm-248  Neutrons per fission (ν)
  18   Pu-239 Neutrons per fission (ν)
```

TAB. 3.13

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using
Flat Weighting and 1/4 T PV Weighting.

	Dosimeter			Dosimeter
1	Li-6 (n,t)		37	Zn-64 (n,p)
2	B-10 (n,α)		38	As-75 (n,2n)
3	F-19 (n,2n)		39	Y-89 (n,2n)
4	Na-23 (n,γ)		40	Zr-90 (n,2n)
5	Na-23 (n,2n)		41	Nb-93 (n,2n)
6	Mg-24 (n,p)		42	Nb-93 (n,n')
7	Al-27 (n,p)		43	Nb-93 (n,γ)
8	Al-27 (n,α)		44	Rh-103 (n,n')
9	P-31 (n,p)		45	Ag-109 (n,γ)
10	S-32 (n,p)		46	In-115 (n,2n)
11	Sc-45 (n,γ)		47	In-115 (n,n')
12	Ti-46 (n,2n)		48	In-115 (n,γ)
13	Ti-46 (n,p)		49	I-127 (n,2n)
14	Ti-47 (n,x)		50	La-139 (n,γ)
15	Ti-47 (n,p)		51	Pr-141 (n,2n)
16	Ti-48 (n,x)		52	Tm-169 (n,2n)
17	Ti-48 (n,p)		53	Ta-181 (n,γ)
18	Ti-49 (n,x)		54	W-186 (n,γ)
19	V-51 (n,α)		55	Au-197 (n,2n)
20	Cr-52 (n,2n)		56	Au-197 (n,γ)
21	Mn-55 (n,γ)		57	Hg-199 (n,n')
22	Fe-54 (n,2n)		58	Pb-204 (n,n')
23	Fe-54 (n,α)		59	Th-232 (n,γ)
24	Fe-54 (n,p)		60	Th-232 (n,f)
25	Fe-56 (n,p)		61	U-235 (n,f)
26	Fe-58 (n,γ)		62	U-238 (n,f)
27	Co-59 (n,2n)		63	U-238 (n,γ)
28	Co-59 (n,α)		64	Np-237 (n,f)
29	Co-59 (n,γ)		65	Pu-239 (n,f)
30	Ni-58 (n,2n)		66	Am-241 (n,f)
31	Ni-58 (n,p)		67	B-nat (n,disap)
32	Ni-60 (n,p)		68	B-nat (n,γ)
33	Cu-63 (n,2n)		69	B-nat (n,α)
34	Cu-63 (n,γ)		70	Cd-nat (n,γ)
35	Cu-63 (n,α)		71	Gd-nat (n,γ)
36	Cu-65 (n,2n)			

TAB. 3.14

Response Functions Included with BUGENDF70.BOLIB.
Neutron KERMA Factor Data [eV·b] Collapsed Using Concrete Weighting Spectrum.

	Nuclide		Nuclide		Nuclide		Nuclide
1	Ag-107	47	Cu-63	93	Li-7	139	S-34
2	Ag-109	48	Cu-65	94	Mg-24	140	S-36*
3	Al-27	49	Er-162	95	Mg-25	141	Si-28
4	Am-241	50	Er-164	96	Mg-26	142	Si-29
5	Am-242#	51	Er-166*	97	Mn-55	143	Si-30
6	Am-242m#	52	Er-167	98	Mo-92*	144	Sn-112#
7	Am-243	53	Er-168	99	Mo-94*	145	Sn-114#
8	Au-197*	54	Er-170	100	Mo-95	146	Sn-115#
9	B-10	55	Eu-151#	101	Mo-96*	147	Sn-116#
10	B-11	56	Eu-152#	102	Mo-97*	148	Sn-117#
11	Ba-138#	57	Eu-153	103	Mo-98*	149	Sn-118#
12	Be-9	58	Eu-154#	104	Mo-100#	150	Sn-119#
13	Be-9 (Thermal)	59	Eu-155#	105	N-14	151	Sn-120#
14	Bi-209*	60	F-19	106	N-15	152	Sn-122#
15	C-nat	61	Fe-54	107	Na-23	153	Sn-123#
16	C (Graphite)	62	Fe-56	108	Nb-93*	154	Sn-124#
17	Ca-40	63	Fe-57	109	Ni-58	155	Sn-125*
18	Ca-42	64	Fe-58	110	Ni-60	156	Sn-126#
19	Ca-43	65	Ga-69#	111	Ni-61	157	Ta-181*
20	Ca-44	66	Ga-71#	112	Ni-62	158	Ta-182#
21	Ca-46	67	Gd-152	113	Ni-64	159	Th-230#
22	Ca-48	68	Gd-154	114	Np-237	160	Th-232
23	Cd-106	69	Gd-155	115	Np-238#	161	Ti-46
24	Cd-108#	70	Gd-156	116	Np-239#	162	Ti-47*
25	Cd-110#	71	Gd-157	117	O-16	163	Ti-48
26	Cd-111	72	Gd-158	118	O-17#	164	Ti-49
27	Cd-112#	73	Gd-160	119	P-31	165	Ti-50
28	Cd-113#	74	H-1 (H$_2$O)	120	Pa-231	166	U-232
29	Cd-114#	75	H-1 (CH$_2$)	121	Pa-233	167	U-233
30	Cd-115m*	76	H-1 (ZrH)	122	Pb-204	168	U-234
31	Cd-116#	77	H-2 (D$_2$O)	123	Pb-206	169	U-235
32	Cl-35	78	H-3#	124	Pb-207	170	U-236
33	Cl-37	79	He-3#	125	Pb-208	171	U-237
34	Cm-241#	80	He-4#	126	Pu-236#	172	U-238
35	Cm-242	81	Hf-174#	127	Pu-237#	173	V-nat
36	Cm-243#	82	Hf-176#	128	Pu-238#	174	W-182
37	Cm-244#	83	Hf-177#	129	Pu-239	175	W-183
38	Cm-245#	84	Hf-178#	130	Pu-240	176	W-184
39	Cm-246#	85	Hf-179#	131	Pu-241	177	W-186
40	Cm-247#	86	Hf-180#	132	Pu-242	178	Y-89
41	Cm-248	87	In-113#	133	Pu-243	179	Zr-90
42	Co-59	88	In-115#	134	Pu-244#	180	Zr-91
43	Cr-50	89	K-39	135	Re-185#	181	Zr-92
44	Cr-52	90	K-40	136	Re-187#	182	Zr-94
45	Cr-53	91	K-41	137	S-32	183	Zr-96*
46	Cr-54	92	Li-6	138	S-33		

TAB. 3.14 Continued

Response Functions Included with BUGENDF70.BOLIB.
Neutron KERMA Factor Data [eV·b] Collapsed Using Concrete Weighting Spectrum.

(*) The KERMA factors of this nuclide are set to zero in all the neutron groups.

(#) The photon production data of this nuclide are not available. All photon energy is deposited locally and this is consistent with the fact that there will be no contribution to the photon transport source from this nuclide.

TAB. 3.15

Response Functions Included with BUGENDF70.BOLIB.
Photon KERMA Factor Data[a] [eV·b] Collapsed Using Concrete Weighting Spectrum.

	Z	Element		Z	Element		Z	Element
1	1	H	19	22	Ti	37	63	Eu
2	2	He	20	23	V	38	64	Gd
3	3	Li	21	24	Cr	39	68	Er
4	4	Be	22	25	Mn	40	72	Hf
5	5	B	23	26	Fe	41	73	Ta
6	6	C	24	27	Co	42	74	W
7	7	N	25	28	Ni	43	75	Re
8	8	O	26	29	Cu	44	79	Au
9	9	F	27	31	Ga	45	82	Pb
10	11	Na	28	39	Y	46	83	Bi
11	12	Mg	29	40	Zr	47	90	Th
12	13	Al	30	41	Nb	48	91	Pa
13	14	Si	31	42	Mo	49	92	U
14	15	P	32	47	Ag	50	93	Np
15	16	S	33	48	Cd	51	94	Pu
16	17	Cl	34	49	In	52	95	Am
17	19	K	35	50	Sn	53	96	Cm
18	20	Ca	36	56	Ba			

(a) **The photon KERMA factors are the same for all the isotopes of each element.**

4 - BUGENDF70.BOLIB PRELIMINARY VALIDATION

The BUGENDF70.BOLIB library was preliminarily tested on two low-flux engineering neutron shielding benchmark experiments, specifically dedicated to improve the accuracy of the neutron fluence calculations in the structural components of the pressurized light water reactors (PWRs), whose geometrical and compositional specifications were contained in the SINBAD /28/ (see also /29/) international database of fission reactor neutron shielding benchmark experiments.

The PCA-Replica 12/13 /25/ (see also /26/) (Winfrith, United Kingdom) and the VENUS-3 /27/ (Mol, Belgium) neutron shielding benchmark experiments were simulated with three-dimensional fixed source transport calculations using the ORNL TORT-3.2 /10/ discrete ordinates deterministic code. The automatic generation and the graphical verification of the benchmark experiment geometrical models for TORT-3.2 were performed through the ENEA-Bologna BOT3P-5.3 /12/ pre/post-processor system.

The BUGENDF70.BOLIB library was alternatively used in the transport calculations with the BUGLE-96 /8/ library, which was specifically employed to obtain corresponding reference dosimetric results. Group-organized files of macroscopic cross sections, requested by TORT-3.2 and derived from the cited working libraries in FIDO-ANISN format, were prepared through the GIP /35/ program, specifically dedicated to the transport codes (ANISN-ORNL, DORT, TORT) of the DOORS /35/ system. The ENEA-Bologna ADEFTA-4.1 /65/ program was used to calculate the atomic densities for the benchmark experiment compositional models and to handle them properly in order to automatically prepare the macroscopic cross section sets of the compositional model material mixtures in the format required by GIP.

The IRDF-2002 /54/ flat weighting dosimeter cross sections /55/ in the 47 neutron energy group structure (see TAB. 3.1) were used in both the calculations for the PCA-Replica 12/13 and VENUS-3 benchmark experiments with the BUGENDF70.BOLIB and BUGLE-96 libraries.

4.1 - PCA-Replica 12/13 Neutron Shielding Benchmark

4.1.1 - PCA-Replica 12/13 Experimental Details

The PCA-Replica 12/13 /25/ low-flux neutron shielding benchmark experiment (Winfrith, United Kingdom) is a water/iron benchmark experiment including PWR thermal shield and pressure vessel simulators. The source of neutrons was a thin fission plate (whose dimensions are 63.5 cm × 40.2 cm × 0.6 cm) of highly enriched uranium (93.0 w% in U-235), irradiated by the NESTOR low-power experimental reactor through a graphite thermal column (total thickness 43.91 cm). Beyond the fission plate, the PCA-Replica shielding array (12/13 experimental configuration with two water gaps of about 12 cm and 13 cm) was arranged in a large parallelepiped steel tank (square section; side 180.0 cm) filled with water. After a first water gap (12.1 cm), there was the stainless steel thermal shield simulator (5.9 cm), the second water gap (12.7 cm), the mild steel pressure vessel simulator (thickness T = 22.5 cm) and a wide box made of a thin layer of aluminium simulating the air cavity between the pressure vessel and the biological shield in a real PWR.

The fission plate, the thermal shield, the pressure vessel and the void box were perfectly orthogonally aligned and centred along an imaginary line Z (horizontal or nuclear axis) passing through the centre of the fission plate. Along this nuclear axis three types of threshold detectors were located in ten positions and gave the integral measurements.

The Rh-103(n,n')Rh-103m, In-115(n,n')In-115m and S-32(n,p)P-32 threshold dosimeters were employed in the PCA-Replica experiment. The corresponding typical parameters in a light water material testing reactor (MTR) spectrum (see /66/), similar to that of PCA-Replica, are reported in TAB. 4.1 to help in the analysis of the calculated results.

In practice the results coming from Rh-103(n,n') and In-115(n,n') are comparable with neutron fluxes above about 1.0 MeV and the results from S-32(n,p) with neutron fluxes above about 3.0 MeV.

Moreover spectral measurements were performed in two positions: in the one-quarter thickness (T) of the reactor pressure vessel (RPV 1/4 T) simulator and in the void box. Two kinds of spectrometer were used. The spherical hydrogen-filled proportional counters employed were of type SP-2 of internal diameter 40.0 mm. Individual counters with gas fillings of approximately 0.5, 1.0, 3.0 and 10.0 atmospheres were used in combination, to cover the energy range from 50.0 keV to 1.2 MeV. The neutron fluxes between 1.0 and 10.0 MeV were determined with a spherical 3.5 ml organic liquid (NE213) scintillator.

TAB. 4.1

PCA-Replica - Dosimeter Parameters in a Light Water MTR Neutron Spectrum.

Dosimeter	Effective Energy Threshold [MeV]	90% Response Energy Range [MeV]	Median Energy [MeV]
Rh-103(n,n')	0.69	0.53 – 5.4	1.9
In-115(n,n')	1.30	1.0 – 5.6	2.4
S-32(n,p)	2.70	2.2 – 7.4	3.9

4.1.2 - PCA-Replica 12/13 Analysis and Results

The BUGENDF70.BOLIB and the BUGLE-96 /8/ libraries were alternatively used to simulate the PCA-Replica 12/13 /25/ engineering neutron shielding benchmark experiment. The whole PCA-Replica 12/13 experimental array was reproduced with the TORT-3.2 /10/ code, using the BOT3P-5.3 /12/ system for the preparation of the input of the geometrical model and the ADEFTA-4.1 /65/ program for the calculation of the atomic densities of the isotopes involved in the compositional model, on the basis of the atomic abundances reported in the BNL-NNDC database /67/. In fact the atomic densities indicated in the official description /25/ of the PCA-Replica experiment are given for natural element except for two uranium isotopes, i.e. U-235 and U-238. All the calculations were performed only with the 29 neutron groups (see TAB. 3.1) above 3.1828E+04 eV since all the energy thresholds of the employed dosimeters are above this energy value.

It was decided to reproduce the whole three-dimensional PCA-Replica experimental array in the (X,Y,Z) cartesian geometry in order to assure a detailed description of the spatial heterogeneity of the neutron source emitted by the fission plate. The origin of the cartesian co-ordinate system was taken in the centre of the fission plate.

In particular a parallelepiped geometry (whose dimensions were 185.08 cm × 180.0 cm × 180.0 cm, respectively along the X, Y and Z axis) was described with a 65X×63Y×182Z fine

spatial mesh grid (see the horizontal section at the height Y=0.0 cm in FIG. 4.1), where Z is the horizontal nuclear axis on which the detector positions are located. Along this axis, in order to obtain the best accuracy in the calculations, volumetric meshes with sides always inferior to 0.5 cm were described. Both infinitely dilute and self-shielded cross sections were selected. Self-shielded cross sections from the library package were used when available: in particular it is underlined that the thermal shield (stainless steel) and the pressure vessel (mild steel) components of the PCA-Replica experiment were characterized by atomic densities quite similar to those used to determine the background cross sections employed in the self-shielding of the BUGENDF70.BOLIB cross sections.

Fixed source transport calculations with one source (outer) iteration were performed using fully symmetrical quadrature sets. The P_3-S_8 approximation was adopted: P_3 corresponds to the order of the expansion in Legendre polynomials of the scattering cross section matrix and S_8 represents the order of the flux angular discretization. Further calculations in the P_5-S_{16} approximation did not give significant differences in the integral dosimetric results.

The theta-weighted difference approximation was selected for the flux extrapolation model. In all the calculations the same numerical value (1.0E-03) for the point-wise flux convergence criterion was employed. The vacuum boundary condition was selected at the left, right, inside, outside, bottom and top geometrical boundaries.

As previously reported, a precise heterogeneous fission neutron source distribution in the fission plate was adopted, following the recommended official specifications (see /25/, page 55, TAB. A6 and page 57, FIG. A1).

The distributed (or volumetric) fission neutron sources used in the calculations with the BUGENDF70.BOLIB library and the BUGLE-96 library were obtained using, respectively, the BUGENDF70.BOLIB U-235 total (prompt + delayed) neutron fission spectrum (χ) data (see APPENDIX C, TAB. C.1) and the BUGLE-96 U-235 prompt neutron fission spectrum (χ) data (see /8/, page 58, TAB. 3.14), since the total neutron fission spectrum data are not available in the BUGLE-96 library. In order to determine the volumetric neutron source, the value of $\bar{\nu}$(U-235) = 2.437 was used for the average number of neutrons produced per U-235 thermal fission, as suggested in the official description (see /25/, page 49) of the PCA-Replica experiment.

The IRDF-2002 /54/ "flat weighting" dosimeter cross sections /55/ in the 47 neutron energy group structure (see TAB. 3.1) for the Rh-103(n,n')Rh-103m, In-115(n,n')In-115m and S-32(n,p)P-32 nuclear reactions were used in the calculations with the BUGENDF70.BOLIB and BUGLE-96 libraries to determine the detector activities in both water and steel locations. The use of the IRDF-2002 "1/4T PV weighting" dosimeter cross sections, to be used more properly in measurement positions located in steel, gave negligible differences in the pressure vessel simulator reaction rate results with respect to the corresponding results obtained with the IRDF-2002 flat weighting dosimeter cross sections. The reaction rate integral results obtained with the flat weighting dosimeter cross sections are reported in TAB. 4.2 and FIG. 4.2 for Rh-103(n,n')Rh-103m, in TAB. 4.3 and FIG. 4.3 for In-115(n,n')In-115m and in TAB. 4.4 and FIG. 4.4 for S-32(n,p)P-32. It is underlined that the calculated reaction rates for both the libraries are practically within the desired target accuracy of ±10%.

The spectral results are reported in FIGs. 4.5÷4.6 for the 1/4 T reactor pressure vessel (RPV 1/4 T) position and in FIGs. 4.7÷4.8 for the void box position.

FIG. 4.1

PCA-Replica Model with TORT-3.2 (X,Y,Z), Horizontal Section at Y=0.0 cm.
Dosimeter Locations "×", 65X×63Y×182Z Spatial Meshes.
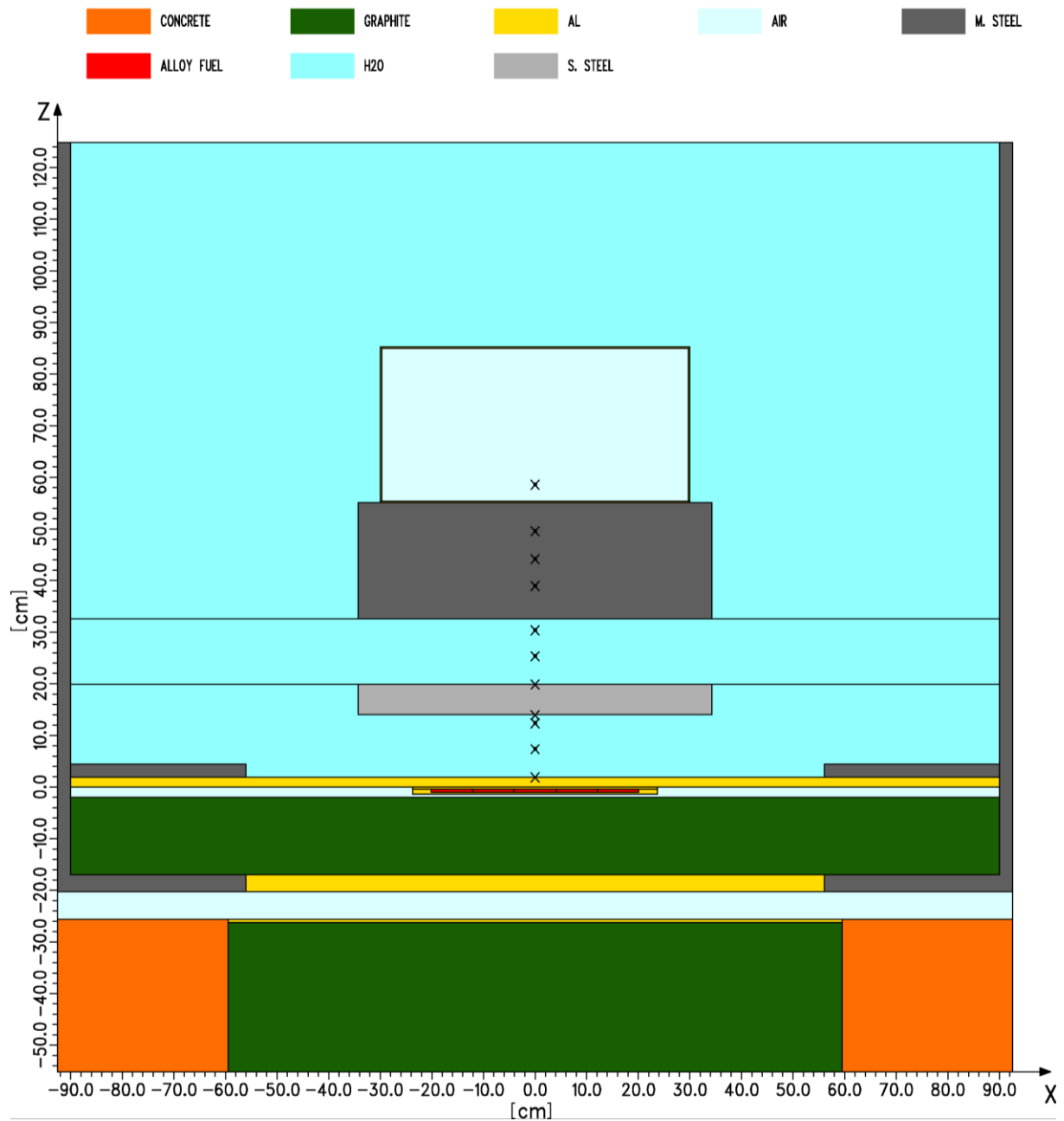
TAB. 4.2

PCA-Replica - Summary of Experimental (E) and Calculated (C) Rh-103(n,n')
Reaction Rates[a] per NESTOR Reactor Watt along the Z Horizontal Axis.

Detec. Pos.	Distance from Fission Plate [cm]	Experimental. Reaction Rates ± Random Error (1σ) (E) Systematic Error ± 3.0%	BUGENDF70.BOLIB Calculation		BUGLE-96 Calculation		Reference Location
			Calculated Reaction Rates (C)	C/E[b]	Calculated Reaction Rates (C)	C/E[b]	
1	1.91	1.69E-20 ± 3.0%	1.81E-20	1.09	1.82E-20	1.10	12 cm Water Gap
2	7.41	3.78E-21 ± 3.0%	3.44E-21	0.93	3.44E-21	0.93	
3	12.41	1.40E-21 ± 3.0%	1.31E-21	0.95	1.30E-21	0.95	
4	14.01	1.27E-21 ± 3.0%	1.16E-21	0.93	1.15E-21	0.93	
5	19.91	4.23E-22 ± 3.0%	4.17E-22	1.00	4.11E-22	0.99	13 cm Water Gap
6	25.41	1.15E-22 ± 4.0%	1.02E-22	0.91	1.01E-22	0.89	
7	30.41	4.73E-23 ± 4.0%	4.13E-23	0.89	4.06E-23	0.88	
8	39.01	2.07E-23 ± 1.0%	2.02E-23	1.01	1.98E-23	1.00	RPV (1/4 T)
9	49.61	5.53E-24 ± 1.9%	5.59E-24	1.05	5.45E-24	1.03	RPV (3/4 T)
10	58.61	1.80E-24 ± 1.6%	1.66E-24	0.96	1.59E-24	0.92	Void Box

Note: The total experimental error (1σ level) should be calculated as the square root of the quadratic sum of the random error listed with each measurement and the systematic error indicated at the head of the column of the experimental results, as indicated on page 12 of reference /25/.

(a) Reaction rates are in units of reactions/(s·atom·NESTOR Watt).
(b) Experimental results contain a contribution from the NESTOR core background. Calculated results refer only to the neutrons produced in the fission plate for 1 Watt of NESTOR power. As indicated on page 10 of reference /25/, the E values, in the C/E ratios, are reduced by 4% in the RPV and Void Box and by 2% in the Water Gaps.

TAB. 4.3

PCA-Replica - Summary of Experimental (E) and Calculated (C) In-115(n,n')
Reaction Rates[a] per NESTOR Reactor Watt along the Z Horizontal Axis.

Detec. Pos.	Distance from Fission Plate [cm]	Experimental. Reaction Rates ± Random Error (1σ) (E) Systematic Error ± 2.0%	BUGENDF70.BOLIB Calculation		BUGLE-96 Calculation		Reference Location
			Calculated Reaction Rates (C)	C/E[b]	Calculated Reaction Rates (C)	C/E[b]	
1	1.91	--	--	--	--	--	12 cm Water Gap
2	7.41	--	--	--	--	--	
3	12.41	--	--	--	--	--	
4	14.01	--	--	--	--	--	
5	19.91	--	--	--	--	--	13 cm Water Gap
6	25.41	--	--	--	--	--	
7	30.41	--	--	--	--	--	
8	39.01	3.93E-24 ± 0.9%	3.87E-24	1.03	3.81E-24	1.01	RPV (1/4 T)
9	49.61	8.23E-25 ± 1.4%	7.76E-25	0.98	7.58E-25	0.96	RPV (3/4 T)
10	58.61	2.31E-25 ± 1.5%	2.16E-25	0.97	2.09E-25	0.94	Void Box

Note: The total experimental error (1σ level) should be calculated as the square root of the quadratic sum of the random error listed with each measurement and the systematic error indicated at the head of the column of the experimental results, as indicated on page 12 of reference /25/.

(a) Reaction rates are in units of reactions/(s·atom·NESTOR Watt).

(b) Experimental results contain a contribution from the NESTOR core background. Calculated results refer only to the neutrons produced in the fission plate for 1 Watt of NESTOR power. As indicated on page 10 of reference /25/, the E values, in the C/E ratios, are reduced by 4% in the RPV and Void Box and by 2% in the Water Gaps.

TAB. 4.4

PCA-Replica - Summary of Experimental (E) and Calculated (C) S-32(n,p)
Reaction Rates[a] per NESTOR Reactor Watt along the Z Horizontal Axis.

Detec. Pos.	Distance from Fission Plate [cm]	Experimental. Reaction Rates ± Random Error (1σ) (E) Systematic Error ± 4.0%	BUGENDF70.BOLIB Calculation		BUGLE-96 Calculation		Reference Location
			Calculated Reaction Rates (C)	C/E[b]	Calculated Reaction Rates (C)	C/E[b]	
1	1.91	--	--	--	--	--	12 cm Water Gap
2	7.41	--	--	--	--	--	
3	12.41	--	--	--	--	--	
4	14.01	--	--	--	--	--	
5	19.91	--	--	--	--	--	13 cm Water Gap
6	25.41	--	--	--	--	--	
7	30.41	--	--	--	--	--	
8	39.01	1.08E-24 ± 1.5%	9.78E-25	0.94	9.67E-25	0.93	RPV (1/4 T)
9	49.61	1.46E-25 ± 1.9%	1.35E-25	0.97	1.34E-25	0.95	RPV (3/4 T)
10	58.61	3.73E-26 ± 1.3%	3.57E-26	1.00	3.52E-26	0.98	Void Box

Note: The total experimental error (1σ level) should be calculated as the square root of the quadratic sum of the random error listed with each measurement and the systematic error indicated at the head of the column of the experimental results, as indicated on page 12 of reference /25/.

([a]) Reaction rates are in units of reactions/(s·atom·NESTOR Watt).
([b]) Experimental results contain a contribution from the NESTOR core background. Calculated results refer only to the neutrons produced in the fission plate for 1 Watt of NESTOR power. As indicated on page 10 of reference /25/, the E values, in the C/E ratios, are reduced by 4% in the RPV and Void Box and by 2% in the Water Gaps.

FIG. 4.2 PCA-Replica - Rh-103(n,n') Reaction Rate Ratios (Calculated/Experimental).
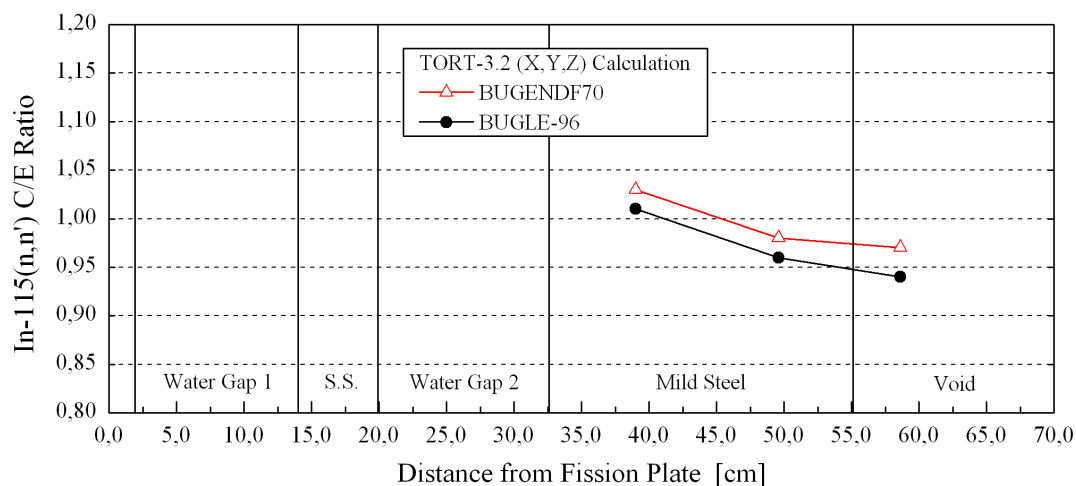
FIG. 4.3 PCA-Replica - In-115(n,n') Reaction Rate Ratios (Calculated/Experimental).
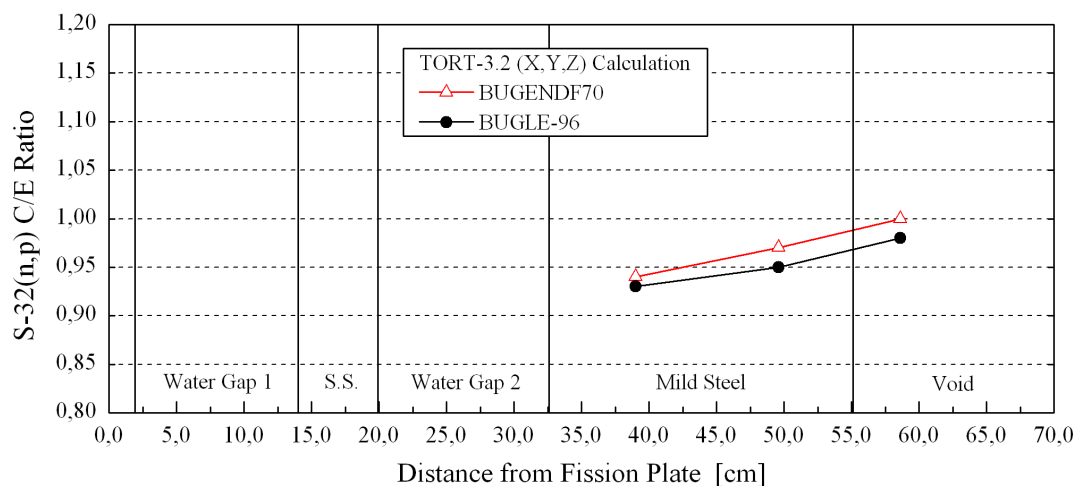
FIG. 4.4 PCA-Replica - S-32(n,p) Reaction Rate Ratios (Calculated/Experimental).

FIG. 4.5

PCA-Replica - Comparison of Experimental and Calculated Neutron Fluxes
in the RPV 1/4 T Position (Mild Steel).
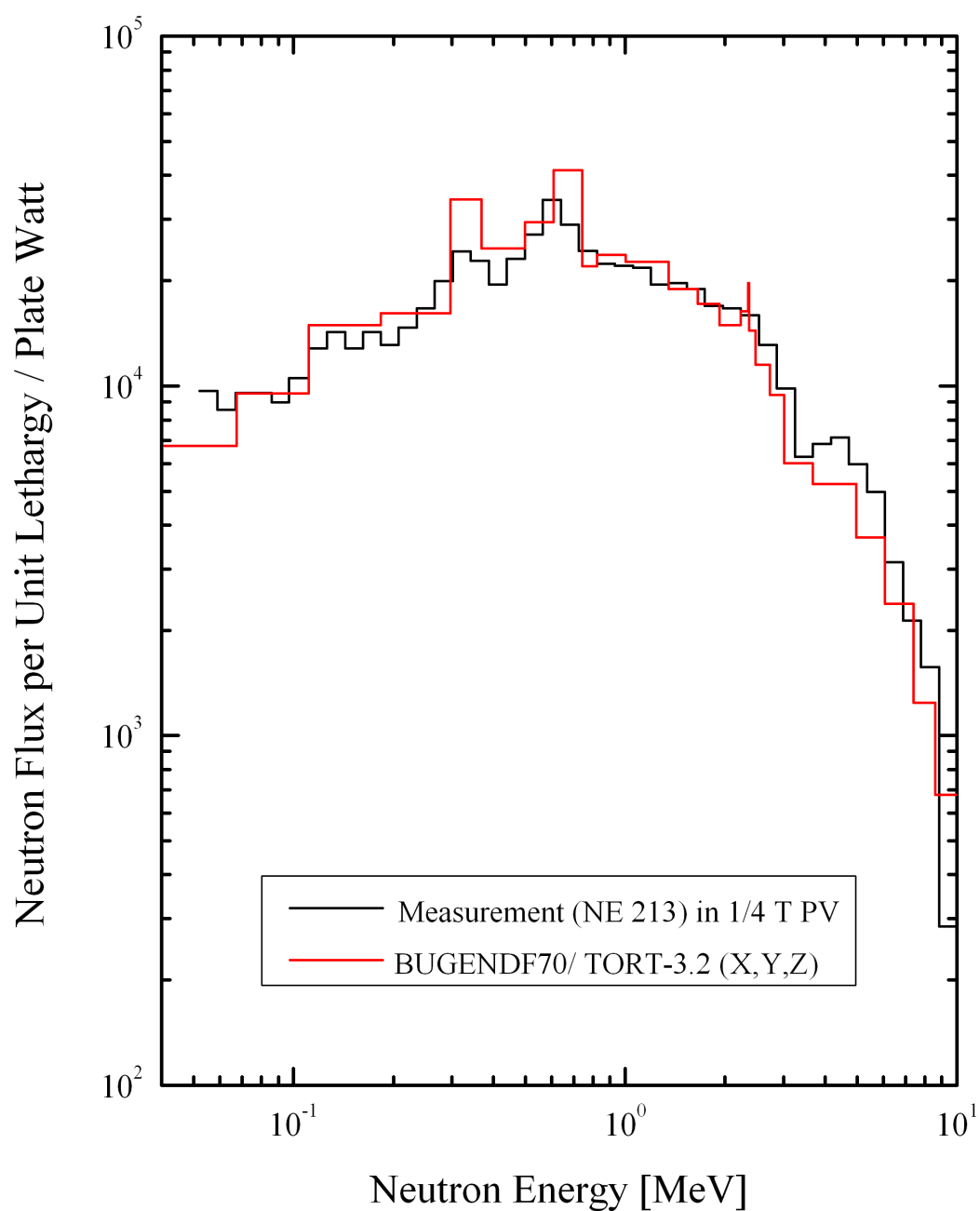
FIG. 4.6

PCA-Replica - Comparison of Experimental and Calculated Neutron Fluxes
in the RPV 1/4 T Position (Mild Steel).
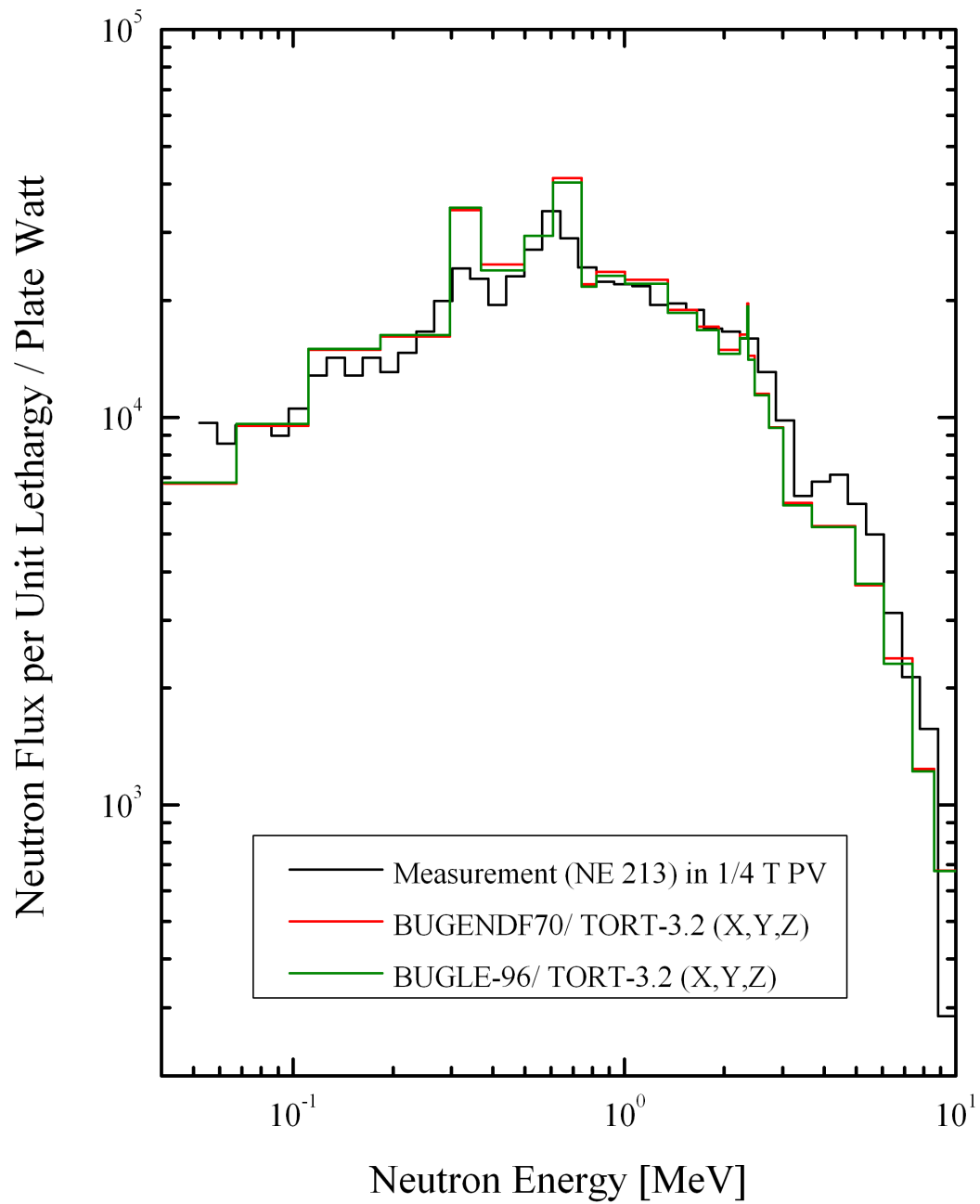
FIG. 4.7

PCA-Replica - Comparison of Experimental and Calculated Neutron Fluxes
in the Void Box Position (Air).

FIG. 4.8

PCA-Replica - Comparison of Experimental and Calculated Neutron Fluxes
in the Void Box Position (Air).

4.2 - VENUS-3 Neutron Shielding Benchmark

4.2.1 - VENUS-3 Experimental Details

The VENUS-3 /27/ low-flux neutron shielding benchmark experiment (Mol, Belgium) is closely related to PWR pressure vessel (PV) safety. It was designed to test the accuracy of the nuclear data and transport codes in the calculation of the neutron radiation damage parameters in stainless steel reactor components, in a context of great precision of the experimental results. Among the available experiments, the VENUS-3 configuration offers the exceptional advantage of exhibiting a realistic radial core shape and a typical PWR neutron spectrum. Typical "17x17" PWR fuel assemblies were employed and a mock-up of the pressure vessel internals, representative of a three-loop Westinghouse power plant, was prepared. VENUS-3 was conceived taking into account that, for some early built reactors, it was proposed to reduce the lead factor at the level of the PV horizontal welding by loading Partial Length Shielded Assemblies (PLSA) at the most critical corners of the core periphery (the shielded part was obtained by replacing part of the fuel length by a stainless steel rod). VENUS-3 was addressed to test this improvement, introducing a PLSA region in the core, and to permit the validation of the analytical methods needed to predict the azimuthal variation of the fluence in the pressure vessel. The VENUS-3 core was designed, in particular, with the following objectives.

1. It had to be representative of typical irradiation conditions of a modern PWR pressure vessel.

2. It had to fit the grid and the vessel geometries of the VENUS facility. This led to a limitation of the core size and of the amount of simulated internals. In particular the second water gap and the pressure vessel of a typical PWR were not simulated.

3. The core loading was projected to obtain a radial power shape factor as low as possible, in order to reach, in the different stainless steel components, fast flux levels high enough to perform accurate measurements. Secondly, the core loading was planned to achieve azimuthal flux variations as high as possible to allow a valuable test of the analytical methods. Finally a quadrangular geometry symmetry was preferred, with two quadrants including both the PLSA fuel region and the unperturbed reference fuel region.

All these objectives were attained with a cruciform-shaped core configuration. The core consisted of three types of fuel pins: 1) stainless-steel-clad UO_2 rods (typical of a "15x15" lattice of the early Generation I reactors of Westinghouse plants) containing 4% enriched U-235, 2) zircaloy-clad UO_2 rods containing 3.3% enriched U-235 and 3) zircaloy-clad UO_2 rods containing 3.3% enriched U-235 over the upper half of their height and zircaloy-clad steel rods over the lower half. The first two types of fuel pins were of uniform composition over their complete height. The 4% enriched rods were positioned in the inner part of the core while the 3.3% enriched rods were located in the arms of the cross configuration together with the PLSA-modified rods. The pin-to-pin pitch for all fuel types was 1.26 cm, typical value of the "17x17" lattices in existing PWR fuel assemblies. In one quadrant of the configuration between 0° and 90° (see the horizontal section in FIG. 4.9 and the vertical section in FIG. 4.10), a mock-up of PWR pressure vessel internals was placed with a total of 639 fuel pins

(including those in the PLSA) and 11 control rods. The other quadrants were loaded with fuel pins "quasi" identical to the fuel pins of the first previous quadrant (due to fuel inventory limitations) and with some absorbing rods for criticality balance adjustment. Starting from the centre, the core quadrant between 0° and 90° may be divided in the following 10 horizontal radial regions:

- the CENTRAL HOLE (water);
- the INNER BAFFLE (stainless steel thickness: 2.858 cm);
- the 4/0 FUEL REGION: 4% enriched uranium fuel rods and 11 pyrex control rods, typical of PWR poison clusters;
- the 3/0 FUEL REGION: 3.3% enriched uranium fuel rods and PLSA rods;
- the OUTER BAFFLE (stainless steel thickness. 2.858 cm);
- the REFLECTOR (water minimum thickness: 2.169 cm);
- the BARREL (stainless steel thickness: 4.99 cm);
- the WATER GAP (water thickness: 5.80 cm);
- the NEUTRON PAD (stainless steel average thickness: 6.72 cm);
- the VENUS environment, i.e., the jacket (air filled), the reactor vessel (stainless steel) and the reactor room (air).

The In-115(n,n')In-115m, Ni-58(n,p)Co-58 and Al-27(n,α)Na-24 threshold dosimeters were employed in the VENUS-3 experiment. The corresponding typical parameters in a light water material testing reactor (MTR) spectrum (see /66/), similar to that of VENUS-3, are reported in TAB. 4.5 to help in the analysis of the obtained calculated results.

TAB. 4.5

VENUS-3 - Dosimeter Parameters in a Light Water MTR Neutron Spectrum.

Dosimeter	Effective Energy Threshold [MeV]	90% Response Energy Range [MeV]	Median Energy [MeV]
In-115(n,n')	1.30	1.0 – 5.6	2.4
Ni-58(n,p)	2.60	1.9 – 7.5	3.9
Al-27(n,α)	7.30	6.5 – 12.0	8.6

In practice (see also /68/) the results coming from In-115(n,n') are comparable with neutron fluxes above about 1.0 MeV, the results from Ni-58(n,p) with neutron fluxes above about 3.0 MeV and the results from Al-27(n,α) with neutron fluxes above about 8.0 MeV.

In the VENUS-3 experiment the total number of the dosimeters was 386: the In-115(n,n') dosimeters were in 104 positions, the Ni-58(n,p) dosimeters were in 244 positions and the Al-27(n,α) dosimeters were in 38 positions. In other words, each set of dosimeters was placed in a part of the 268 total different spatial locations. Axially, the dosimeters were located at 14 different axial levels between 105.0 cm and 155.0 cm, respectively the lower height and the upper height of the active core region (see FIG. 4.10). The maximum total uncertainty of the

VENUS-3 experimental equivalent fission fluxes corresponding to each of the three dosimeters is ±5% (see /51/, page 31).

FIG. 4.9

VENUS-3 Model with TORT-3.2 (R,Θ,Z), Horizontal Section at Z=114.50 cm.
Dosimeter Locations "×", 111R×113Θ×71Z Spatial Meshes.
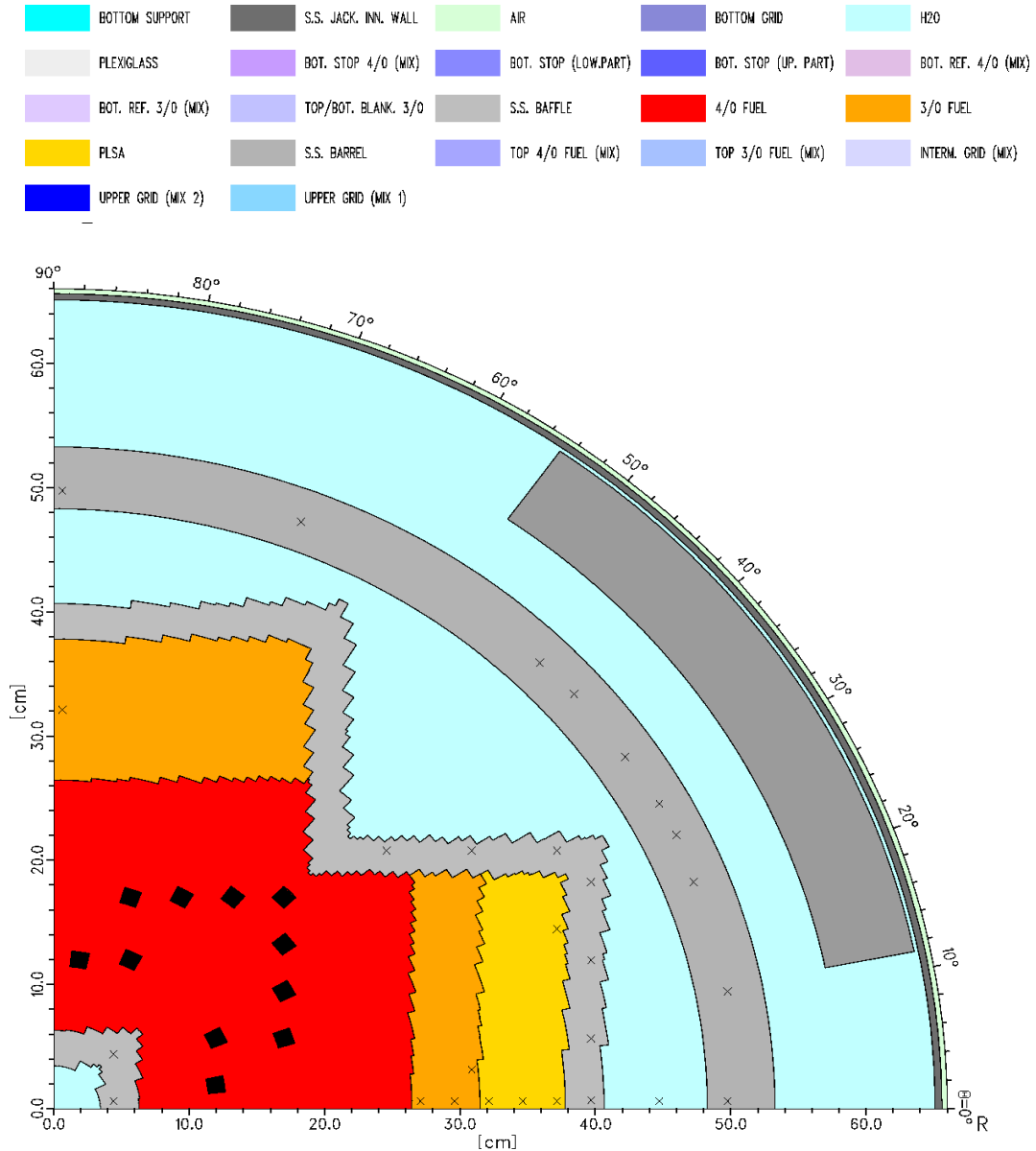
FIG. 4.10

VENUS-3 Model with TORT-3.2 (R,Ɵ,Z), Vertical Section at Ɵ=0°.
Dosimeter Locations "×", 111R×113Ɵ×71Z Spatial Meshes.

4.2.2 - VENUS-3 Analysis and Results

The BUGENDF70.BOLIB and the BUGLE-96 /8/ libraries were alternatively used to simulate the VENUS-3 /27/ (see also /49/ /50/ and /51/) engineering neutron shielding benchmark experiment. Both infinitely dilute and self-shielded cross sections were selected. In the case, e.g., of Fe-56, in the material mixtures of stainless steel components, the self-shielded cross sections were used. All the calculations were performed only with the 26 neutron groups (see TAB. 3.1) above 1.1109E+05 eV since all the energy thresholds of the employed dosimeters are above this energy value. Three-dimensional calculations in cylindrical (R,Θ,Z) geometry were performed through the TORT-3.2 /10/ discrete ordinates (S_N) transport code, included in the ORNL DOORS-3.2 /35/ modular system, using both the previously cited libraries. Fixed source calculations with one source (outer) iteration were performed in the P_3-S_8 approximation: P_3 corresponds to the order of the expansion in Legendre polynomials of the scattering cross section matrix and S_8 represents the order of the flux angular discretization. Fully symmetrical quadrature sets were introduced. The theta-weighted difference approximation was selected for the flux extrapolation model. In all the calculations the same numerical value (1.0E-04) for the point-wise flux convergence criterion was employed. The BOT3P-5.3 /12/ system of pre/post-processor programs was used to prepare the automatic generation of the input data for the neutron source and for the geometrical models together with the graphical visualizations. The ADEFTA-4.1 /65/ program was employed for the calculation of the atomic densities of the isotopes involved in the compositional model, on the basis of the atomic abundances reported in the BNL-NNDC database /67/. The (R,Θ,Z) geometrical model is shown in FIG. 4.9 and FIG. 4.10, reproducing respectively the horizontal and the vertical sections. It is underlined that the (R,Θ,Z) model is also usually adopted for the LWR-PV radiation damage transport analyses. In particular a horizontal plane section between 0° and 90° was described (i.e. the first quadrant up to a radius of 66.0 cm, reported in FIG. 4.9) containing the barrel, the neutron pad and the jacket inner wall with a 111R×113Θ×71Z fine spatial mesh grid. The jacket outer wall and the external regions beyond the jacket outer wall were not included since it was considered that they could only slightly affect the results. The following boundary conditions were selected: reflection at the left, inside and outside boundaries and vacuum at the right, bottom and top boundaries.

The distributed (or volumetric) fission neutron sources used in the BUGENDF70.BOLIB and BUGLE-96 calculations and the calculated results in terms of equivalent fission fluxes for the three threshold detectors were obtained using respectively, the BUGENDF70.BOLIB U-235 total (prompt + delayed) neutron fission spectrum (χ) data (see APPENDIX C, TAB. C.1) and the BUGLE-96 U-235 prompt neutron fission spectrum (χ) data (see /8/, page 58, TAB. 3.14), since the total neutron fission spectrum data are not available in the BUGLE-96 library. To determine the volumetric neutron source, the value of $\bar{\nu}$(U-235) = 2.432 for the average number of neutrons produced per U-235 thermal fission was used, as proposed value (see /69/, page 7) for the ENDF/B-VI U-235 processed data file, contained in the BUGLE-96 library.

The calculated dosimeter reaction rates (activities) were divided by the corresponding value of the "flat weighting" dosimeter cross section, averaged on the specific U-235 fission spectrum. The following flat weighting corresponding values (see TAB. 4.6) were used in all the calculations in order to treat simultaneously both water and steel dosimeter locations.

TAB. 4.6

VENUS-3 - IRDF-2002 Dosimeter Cross Sections Averaged on the U-235 Neutron Fission Spectra Respectively Taken from the BUGENDF70.BOLIB and BUGLE-96 Libraries.

Library	In115(n,n') [barns]	Ni-58(n,p) [barns]	Al-27(n,α) [barns]
BUGENDF70	1.8746E-01	1.0836E-01	7.8151E-04
BUGLE-96	1.8853E-01	1.0906E-01	7.8671E-04

A synthesis of the Calculated/Experimental (C/E) results for the In-115(n,n')In-115m, Ni-58(n,p)Co-58 and Al-27(n,α)Na-24 threshold dosimeters, in terms of equivalent fission flux ratios, are respectively reported in FIG. 4.11, FIG. 4.12 and FIG. 4.13.

Deviations contained within ±5% from the 386 dosimeter experimental data were obtained in about 85% of the calculated results with BUGENDF70.BOLIB and in about 88% of the results with BUGLE-96. The calculated equivalent fission fluxes for both the libraries were practically within the desired target accuracy of ±10% (see TAB. 4.7), recommended by OECD-NEA.

TAB. 4.7

VENUS-3 - Percentages[a] of the Calculated (C) Equivalent Fission Fluxes with Deviations Exceeding 5% and 10% the Corresponding Experimental (E) Fluxes.

Library	In115(n,n')	Ni-58(n,p)	Al-27(n,α)
	$|C-E| / E > 5\%$		
BUGENDF70	3.8%	19.7%	18.4%
BUGLE-96	3.8%	12.3%	28.9%

Library	In115(n,n')	Ni-58(n,p)	Al-27(n,α)
	$|C-E| / E > 10\%$		
BUGENDF70	0.0%	1.2%	0.0%
BUGLE-96	0.0%	1.6%	2.6%

([a])   Percentages calculated on the total number of each dosimeter type: 104 for In, 244 for Ni and 38 for Al.
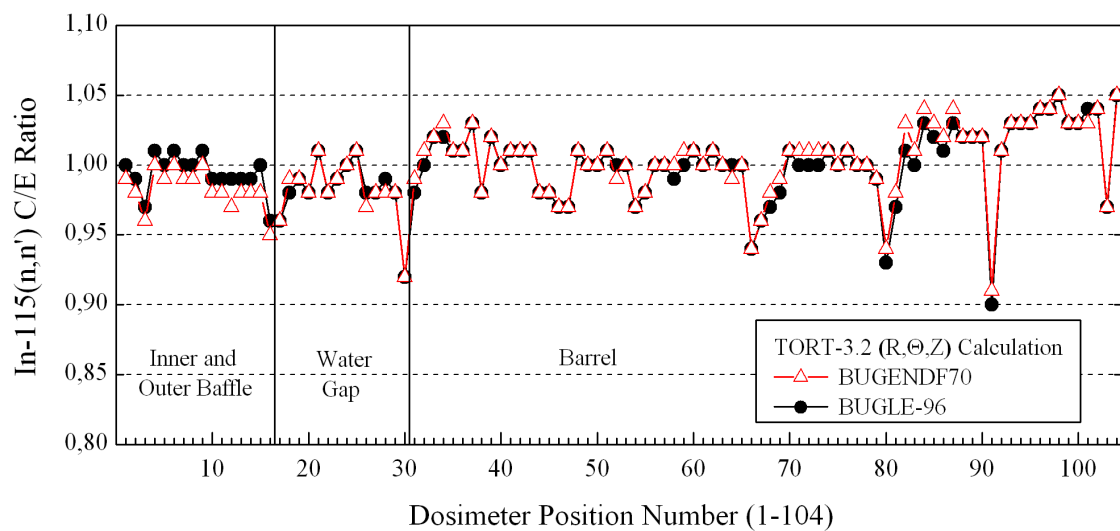
FIG. 4.11 VENUS-3 - In-115(n,n') Equivalent Fission Flux Ratios (Calculated/Experimental).
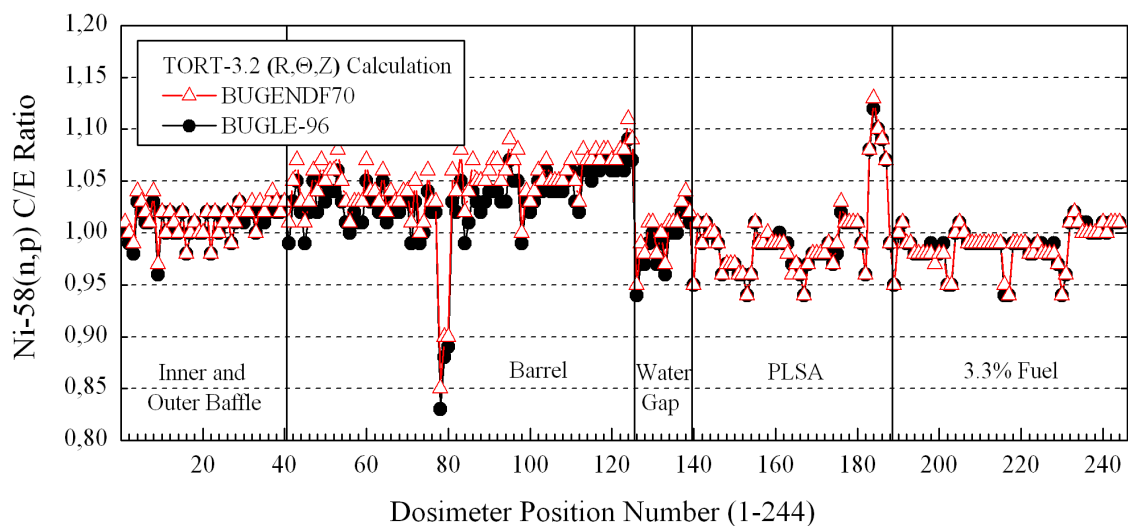
FIG. 4.12 VENUS-3 - Ni-58(n,p) Equivalent Fission Flux Ratios (Calculated/Experimental).
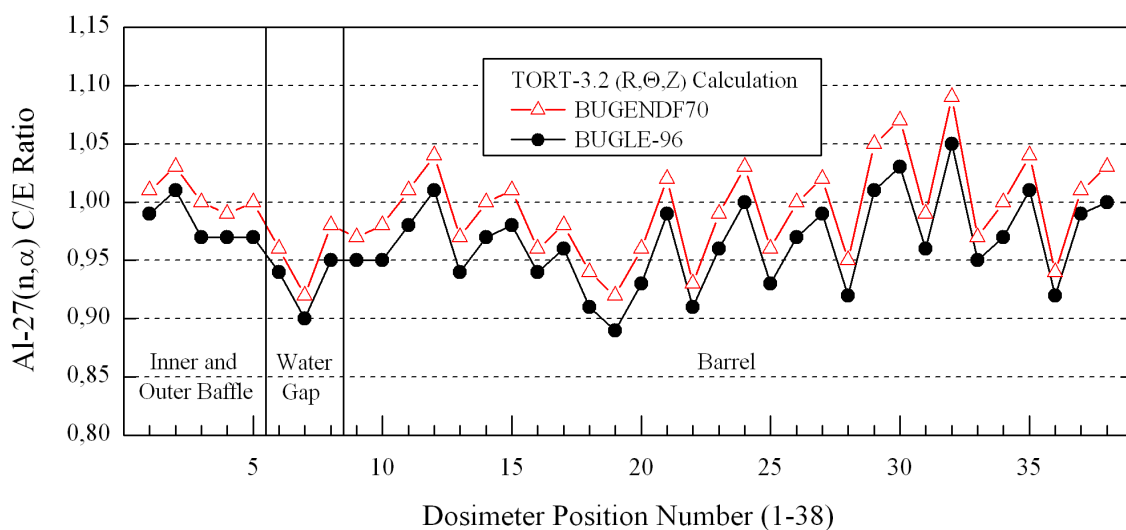
FIG. 4.13 VENUS-3 - Al-27(n,α) Equivalent Fission Flux Ratios (Calculated/Experimental).

5 - CONCLUSION

A new coupled neutron and photon broad-group working cross section library for LWR shielding and pressure vessel dosimetry applications was generated with the name BUGENDF70.BOLIB. Two versions of the library in FIDO-ANISN format are available in the dedicated package: BUGENDF70.BOLIB, without upscattering cross sections in the thermal neutron energy region, and BUGENDF70T.BOLIB, including the upscattering cross sections. This library, in the BUGLE-96 neutron and photon energy group structures (47 n + 20 γ), was obtained through problem-dependent cross section collapsing from the VITENDF70.BOLIB fine-group general-purpose library in AMPX format, based on ENDF/B-VII.0 nuclear data and generated in the VITAMIN-B6 neutron and photon energy group structures (199 n + 42 γ). The BUGENDF70.BOLIB library was processed with an updated and corrected version of the SCAMPI data processing system, the ENEA-Bologna 2007 Revision, already freely available at OECD-NEA Data Bank and at ORNL-RSICC. The library was preliminarily validated on two engineering neutron shielding benchmark experiments (PCA-Replica 12/13 and VENUS-3), specifically designed for the improvement of the accuracy of the LWR radiation shielding and radiation damage calculations. The calculated results for both the integral experiments are practically within the desired target accuracy value of $\pm 10\%$. Further validation of the BUGENDF70.BOLIB library will be performed on other integral neutron shielding benchmark experiments.

The possibility is being studied to obtain from the VITENDF70.BOLIB library a new broad-group working library in FIDO-ANISN format with an increased number of thermal neutron energy groups with respect to those included in the BUGLE-96 library. This to permit more accurate analyses of the thermal neutron and photon radiation damage in applications where these contributions to the total damage are not negligible.

It is believed that it should be very important to further promote, at the international level (UNO-IAEA, OECD-NEA, industrial organizations, R&D institutions, etc.), the generation of new working cross section libraries for radiation shielding applications, dedicated to various types of Generation III and IV nuclear fission reactors with different spectral, geometrical and compositional specifications. In fact the availability of these libraries could promote, in particular, the use of the three-dimensional deterministic codes in radiation shielding and radiation damage applications for nuclear safety. This is increasingly requested, due to the fact that these transport codes (TORT, PARTISN, KATRIN, etc.) have now sophisticated possibilities of complex geometry description, through modern dedicated pre/post-processor systems (BOT3P, TORTWARE, etc.) for the automatic generation of the spatial meshes.

In general, the severe nuclear accidents to the PWR unit No. 2 of the Three Mile Island (Harrisburg, US, March 28, 1979) nuclear power plant and, just recently, to the BWR units No. 1, 2 and 3 of the Fukushima Dai-ichi (Japan, March 11, 2011) nuclear power plant, emphasized and confirmed that the structural integrity of the LWR pressure vessel is one of the most relevant key factors within the defence-in-depth approach of the LWR nuclear safety. The RPV structural integrity is particularly important for PWR long term operations, especially when approaching the End-of-Life (EoL) neutron fluence values in the pressure vessel. The monitoring and the evaluation of the degradation of the RPV materials require not only suitable experimental techniques (reactor dosimetry, thermo-mechanical tests on RPV steels, etc.) but also proven calculation tools as, e.g., group working cross section libraries

like BUGLE-96, BUGLE-B7, BUGJEFF311.BOLIB or BUGENDF70.BOLIB which can play a crucial role to quantify accurately the neutron and photon radiation damage to the pressure vessel, during the whole LWR lifetime.

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APPENDIX A

SCAMPI Input Listings for Key Processing Steps

TAB. A.1

Processing Job for Generating Self-Shielded BWR/PWR Nuclides.
(Table corresponding to TAB. A.5 in reference /8/)

```
#!/bin/csh
# shell script to execute a scampi job
=shell
rm ft*  2>/dev/null
ln -sf /home/VITAMIN/vitamin.b70 ft60f001
end
=ajax
' get data needed for collapsing off of the master tape
0$$ 65 e
1$$ 1 t
2$$ 60 42 t
3$$ 1001 5010 8016
    24050 24052 24053 24054
    26054 26056 26057 26058
    28058 28060 28061 28062 28064
    40090 40091 40092 40094 40096
    92235 92238
    25055
    6012
    11023
    12024 12025 12026
    13027
    14028 14029 14030
    19039 19040 19041
    20040 20042 20043 20044 20046 20048
t
end
=bonami
' pwr fuel cell from resar westinghouse plant
0$$ 65 0 18 81
1$$ 2 3 24 1 1 2
2** 0.001 1.0 t
3$$ 3r3 18r2 3r1
4$$ 1001 5010 8016
    24050 24052 24053 24054
    26054 26056 26057 26058
    28058 28060 28061 28062 28064
    40090 40091 40092 40094 40096
    92235 92238 8016
5** 4.714e-2 4.2e-6 2.357e-2
    3.3196e-6 6.40156e-5 7.258e-6 1.8069e-6
    8.555e-6 1.32994e-4 3.045e-6 4.06e-7
    5.98728e-4 2.28897e-4 9.91e-6 3.1484e-5 7.981e-6
    2.196915e-2 4.79094e-3 7.32305e-3 7.42126e-3 1.1956e-3
    6.325e-4 2.166e-2 4.465e-2
6$$ 1 2 3
7** 0.41783 0.47498 0.71079
8** 921 672 583
10$$ 100113 501013 801613
    2405012 2405212 2405312 2405412
    2605412 2605612 2605712 2605812
```

TAB. A.1 Continued

Processing Job for Generating Self-Shielded BWR/PWR Nuclides.
(Table corresponding to TAB. A.5 in reference /8/)

```
      2805812 2806012 2806112 2806212 2806412
      4009012 4009112 4009212 4009412 4009612
      9223511 9223811 801611
11$$ 0 1 1 t
end
=bonami
' bwr fuel cell from gesar 238 general electric
0$$ 65 0 18 82
1$$ 2 3 23 1 1 2
2** 0.001 1.0 t
3$$ 2r3 18r2 3r1
4$$ 1001 8016
      24050 24052 24053 24054
      26054 26056 26057 26058
      28058 28060 28061 28062 28064
      40090 40091 40092 40094 40096
      92235 92238 8016
5** 2.475e-2 1.2375e-2
      3.3196e-6 6.40156e-5 7.258e-6 1.8069e-6
      8.555e-6 1.32994e-4 3.045e-6 4.06e-7
      5.98728e-4 2.28897e-4 9.91e-6 3.1484e-5 7.981e-6
      2.196915e-2 4.79094e-3 7.32305e-3 7.42126e-3 1.1956e-3
      4.959e-4 2.177e-2 4.455e-2
6$$ 1 2 3
7** 0.53213 0.6134 0.9174
8** 921 672 551
10$$ 100123 801623
       2405022 2405222 2405322 2405422
       2605422 2605622 2605722 2605822
       2805822 2806022 2806122 2806222 2806422
       4009022 4009122 4009222 4009422 4009622
       9223521 9223821 801621
11$$ 0 1 1 t
end
=bonami
' infinite medium collapse for iron-water mixture
0$$ 65 0 18 83
1$$ 0 1 16 1 1 0 t
3$$ f1
4$$ 1001 8016
      24050 24052 24053 24054
      26054 26056 26057 26058
      28058 28060 28061 28062 28064
      25055
5** 1.1785e-2 5.8925e-3
      5.67e-4 1.09346e-2 1.2398e-3 3.086e-4
      2.5798e-3 4.01046e-2 9.182e-4 1.224e-4
      4.3778e-3 1.67366e-3 7.246e-5 2.3021e-4 5.835e-5
      1.14e-3
6$$ 1
7** 10.0
```

TAB. A.1 Continued

Processing Job for Generating Self-Shielded BWR/PWR Nuclides.
(Table corresponding to TAB. A.5 in reference /8/)

```
8** 590.
10$$ 100131 801631
     2405031 2405231 2405331 2405431
     2605431 2605631 2605731 2605831
     2805831 2806031 2806131 2806231 2806431
     2505531
11$$ 1 t
end
=bonami
' infinite medium collapse for concrete type 04
0$$ 65 0 18 84
1$$ 0 1 24 1 1 0 t
3$$ f1
4$$ 1001 8016
     26054 26056 26057 26058
     6012
     11023
     12024 12025 12026
     13027
     14028 14029 14030
     19039 19040 19041
     20040 20042 20043 20044 20046 20048
5** 8.6e-3 4.33e-2
     2.0355e-5 3.16434e-4 7.245e-6 9.66e-7
     1.15e-4
     9.64e-4
     9.79476e-5 1.24e-5 1.36524e-5
     1.74e-4
     1.531018e-3 7.7522e-5 5.146e-5
     4.289873e-4 5.382e-8 3.095892e-5
     1.454115e-3 9.705e-6 2.025e-6 3.129e-5 6.e-8 2.805e-6
6$$ 1
7** 100.0
8** 300.0
10$$ 100141 801641
     2605441 2605641 2605741 2605841
     601241
     1102341
     1202441 1202541 1202641
     1302741
     1402841 1402941 1403041
     1903941 1904041 1904141
     2004041 2004241 2004341 2004441 2004641 2004841
11$$ 1 t
end
=bonami
' self-shield carbon steel (1) & stainless steel (2)
0$$ 65 0 18 85 e
1$$ 1 2 30 2r0 0
2** 0.001 0.0 t
3$$ 15r1 15r2
```

TAB. A.1 Continued

Processing Job for Generating Self-Shielded BWR/PWR Nuclides.
(Table corresponding to TAB. A.5 in reference /8/)

```
4$$ 26054 26056 26057 26058
    24050 24052 24053 24054
    28058 28060 28061 28062 28064
    25055 6012
    26054 26056 26057 26058
    24050 24052 24053 24054
    28058 28060 28061 28062 28064
    25055 6012
5** 4.75e-3 7.518e-2 1.72e-3 2.457e-4
    5.518e-6 1.06413e-4 1.2065e-5 3.004e-6
    3.03119e-4 1.15884e-4 5.017e-6 1.594e-5 4.04e-6
    1.12e-3 9.81e-4
    3.381e-3 5.352e-2 1.224e-3 1.749e-4
    7.56e-4 1.45795e-2 1.653e-3 4.115e-4
    5.83709e-3 2.23155e-3 9.662e-5 3.0695e-4 7.781e-5
    1.52e-3 2.37e-4
6$$ 1 2
7** 10.0 20.0
8** f600.0
9** f0.0
10$$ 2605451 2605651 2605751 2605851
     2405051 2405251 2405351 2405451
     2805851 2806051 2806151 2806251 2806451
     2505551 601251
     2605452 2605652 2605752 2605852
     2405052 2405252 2405352 2405452
     2805852 2806052 2806152 2806252 2806452
     2505552 601252
11$$ 1 1 t
end
=ajax
0$$ 75 81 e
1$$ 5 t
2$$ 81 24 t
3$$ 100113 501013 801613
    2405012 2405212 2405312 2405412
    2605412 2605612 2605712 2605812
    2805812 2806012 2806112 2806212 2806412
    4009012 4009112 4009212 4009412 4009612
    9223511 9223811 801611 t
2$$ 82 23 t
3$$ 100123 801623
    2405022 2405222 2405322 2405422
    2605422 2605622 2605722 2605822
    2805822 2806022 2806122 2806222 2806422
    4009022 4009122 4009222 4009422 4009622
    9223521 9223821 801621 t
2$$ 83 16 t
3$$  100131  801631
    2405031 2405231 2405331 2405431
    2605431 2605631 2605731 2605831
```

TAB. A.1 Continued

Processing Job for Generating Self-Shielded BWR/PWR Nuclides.
(Table corresponding to TAB. A.5 in reference /8/)

```
      2805831 2806031 2806131 2806231 2806431
      2505531 t
2$$ 84 24 t
3$$  100141  801641
      2605441 2605641 2605741 2605841
      601241 1102341
      1202441 1202541 1202641
      1302741
      1402841 1402941 1403041
      1903941 1904041 1904141
      2004041 2004241 2004341 2004441 2004641 2004841
      t
2$$ 85 30 t
3$$ 2605451 2605651 2605751 2605851
      2405051 2405251 2405351 2405451
      2805851 2806051 2806151 2806251 2806451
      2505551 601251
      2605452 2605652 2605752 2605852
      2405052 2405252 2405352 2405452
      2805852 2806052 2806152 2806252 2806452
      2505552 601252 t
end
=nitawl
' xsdrn library will be produced on unit 76
0$$ 75 a4 76 e
1$$ a2 117 a11 -1 e t
2$$  100113  501013  801613
      2405012 2405212 2405312 2405412
      2605412 2605612 2605712 2605812
      2805812 2806012 2806112 2806212 2806412
      4009012 4009112 4009212 4009412 4009612
      9223511 9223811 801611
       100123  801623
      2405022 2405222 2405322 2405422
      2605422 2605622 2605722 2605822
      2805822 2806022 2806122 2806222 2806422
      4009022 4009122 4009222 4009422 4009622
      9223521 9223821 801621
      100131  801631
      2405031 2405231 2405331 2405431
      2605431 2605631 2605731 2605831
      2805831 2806031 2806131 2806231 2806431
      2505531
      100141  801641
      2605441 2605641 2605741 2605841
      601241 1102341
      1202441 1202541 1202641
      1302741
      1402841 1402941 1403041
      1903941 1904041 1904141
      2004041 2004241 2004341 2004441 2004641 2004841
```

TAB. A.1 Continued

Processing Job for Generating Self-Shielded BWR/PWR Nuclides.
(Table corresponding to TAB. A.5 in reference /8/)

```
      2605451 2605651 2605751 2605851
      2405051 2405251 2405351 2405451
      2805851 2806051 2806151 2806251 2806451
      2505551 601251
      2605452 2605652 2605752 2605852
      2405052 2405252 2405352 2405452
      2805852 2806052 2806152 2806252 2806452
      2505552 601252
4**  21r600.
      3r1000.
      20r600.
      3r1000.
      16r600.
      24r300.
      30r600. t
end
=shell
mv ft75f001 /home/BUGLE/A05.bonami.b70
mv ft76f001 /home/BUGLE/A05.nitawl.b70
rm ft*  2>/dev/null
end
```

TAB. A.2

Processing Job for Calculating BWR Weighting Spectrum.
(Table corresponding to TAB. A.6 in reference /8/)

```
#!/bin/csh
# shell script to execute a scampi job
=shell
rm ft*  2>/dev/null
ln -sf /home/BUGLE/A05.nitawl.b70 ft04f001
end
=xsdrnpm
epri np-152 bwr case 3579 mwt gesar 238" core midplane 50% void
-1$$ 2000000
1$$ 2   7  154  1  0
    5  76    8  3  0
   40 100    0  0  3
2$$ -1 33 0 3z -1 1 e
3$$ 1 1 0 0 0
    0 0 1 0 0
    0 0
4$$ 0 67 41 -2 3
    4 70 -1 0
5** 0.00001 0.00001 7.46419+17 2z
    1.420892 4z
    0.001 0.75 t
13$$ 14r1 2r2 18r3 18r4
        24r5
'    fuel water ss304 a533b concrete
14$$  100123  801623
    4009022 4009122 4009222 4009422 4009622
    2605422 2605622 2605722 2605822
    9223521 9223821 801621
    100123  801631
    601252
    1402841 1402941 1403041
    2405052 2405252 2405352 2405452
    2505552
    2605452 2605652 2605752 2605852
    2805852 2806052 2806152 2806252 2806452
    601251
    1402841 1402941 1403041
    2405051 2405251 2405351 2405451
    2505551
    2605451 2605651 2605751 2605851
    2805851 2806051 2806151 2806251 2806451
    100141  601241  801641 1102341
    1202441 1202541 1202641
    1302741
    1402841 1402941 1403041
    1903941 1904041 1904141
    2004041 2004241 2004341 2004441 2004641 2004841
    2605441 2605641 2605741 2605841
15** 1.5354-2 7.677-3
    2.96584-3 6.46777-4 9.88612-4 1.00187-3 1.61406-4
    1.1818-6 1.83715-5 4.206-7 5.61-8
```

TAB. A.2 Continued

Processing Job for Calculating BWR Weighting Spectrum.
(Table corresponding to TAB. A.6 in reference /8/)

```
       1.2125-4 5.322-3 1.0884-2
       4.950-2 2.475-2
       2.37-4
       8.23614-4 4.17031-5 2.76830-5
       7.5603-4 1.457946-2 1.653-3 4.1151-4
       1.52-3
       3.4397-3 5.347276-2 1.2243-3 1.6324-4
       5.837085-3 2.23155-3 9.6615-5 3.06945-4 7.7805-5
       9.81-4
       3.42173-4 1.73257-5 1.15010-5
       5.51815-6 1.064133-4 1.2065-5 3.00355-6
       1.12-3
       4.832094-3 7.5118588-2 1.719898-3 2.2932-4
       3.031188-4 1.15884-4 5.0172-6 1.59396-5 4.0404-6
       7.77-3 1.15-4 4.38-2 1.05-3
       1.16905-04 1.48000-05 1.62948-05
       2.39-3
       1.45723-2 7.37860-4 4.89800-4
       6.46279-04 8.10810-08 4.66403-05
       2.83068-3 1.88924-5 3.942-6 6.09112-5 1.168-7 5.4604-6
       1.8467-5 2.870836-4 6.573-6 8.764-7
t
30$$ 64r1 f0
' lwr source spectrum for 199-group structure
31**
944991-14 408572-11 410082-11 1407-8 2346-8 1676-8 212-7 2669-8 3318-8
9212-8 689696-10 689304-10 2022-7 2892-7 4059-7 5583-7 7525-7 9963-7
1297-6 166-5 2092-6 2597-6 3179-6 3839-6 144-5 3137-6 539-5 6274-6
7224-6 8229-6 9282-6 1037-5 1148-5 2632-5 3072-5 3482-5 188-4 1965-5
203858-7 210743-7 2164-5 2212-5 1496-5 3787-6 3769-6 758-5 1523-5 23-3
231-4 2312-5 2306-5 229-4 2266-5 2237-5 2202-5 2162-5 2111-5 20628-6
200921-7 1953-5 189-4 3591-5 1422-5 1912-5 1568-5 1503-5 1438-5 1373-5
131-4 1248-5 1188-5 1128-5 1071-5 1015-5 9643-6 9099-6 1672-5 1488-5
6804-6 6405-6 1168-5 1031-5 111318-8 408612-9 857096-9 2305-6 4395-6
7978-6 3614-6 3382-6 3164-6 296-5 2765-6 2584-6 2417-6 2253-6 2107-6
1962-6 1834-6 1708-6 1595-6 1487-6 1383-6 1291-6 2853-6 2386-6 800687-9
588195-9 142617-8 846046-9 1879-6 6696-7 9652-7 8036-7 9037-7 3215-7
412869-9 179031-9 11-5 1452-7 6873-8 6624-8 1846-7 2654-7 4036-7 2781-7
766153-10 114985-9 1319-7 9077-8 6245-8 2768-8 1528-8 1316-8 1132-8
5056-9 4693-9 8395-9 7227-9 1397-8 9602-9 6601-9 4538-9 3119-9 2144-9
147411-11 101289-11 6962-10 4784-10 329-9 226-9 1553-10 1067-10 7339-11
504437-13 345483-13 23938-12 163763-13 112537-13 7737-12 5317-12 3654-12
2512-12 1727-12 1186-12 813315-15 562685-15 385-12 112103-15 152962-15
299068-16 274697-16 285471-16 960036-16 45626-15 793822-16 302382-16
556619-16 144778-16 446222-16 97115-16 970882-17 13405-15 161026-16
161422-16 163938-16 146302-16 179059-16 596373-18 562593-18 373104-18
481014-18 596373-18 619278-18 621267-18 115897-17 153207-17 231793-17
654104-17 f0
t
33** f0
t
```

TAB. A.2 Continued

Processing Job for Calculating BWR Weighting Spectrum.
(Table corresponding to TAB. A.6 in reference /8/)

```
35** 49i 0 3i 200 4i 210 4i 225 14i 235.15 2i 253.75
     19i 258.83 25i 302.26 317.5 8i 318 345 4i 347.22 9i 357 390
36$$ 64r1 15r2 3r3 20r4 26r5 11r6 15r7
38** 128r1 11r 1.0-10 f1
39$$ 1 2 3 2 4 2 5
40$$ f3
51$$
 0 6r1 5r2 4r3 3r4 3r5 5r6 4r7 4r8 3r9 2r10 2r11 2r12 13 2r14 3r15
 3r16 4r17 5r18 4r19 2r20 4r21 4r22 4r23 4r24 10r25 10r26 6r27 4r28
 2r29 3r30 2r31 2r32 2r33 4r34 4r35 7r36 5r37 3r38 3r39 4r40 5r41
 3r42 4r43 7r44 6r45 8r46 11r47 2z 2r48 49 2r50 2r51 2r52 2r53 2r54
 2r55 2r56 3r57 58 59 60 4r61 2r62 2r63 3r64 3r65 66 67 0
t
end
=shell
cp ft33f001 /home/BUGLE/A06.scalar.b70
cp ft03f001 /home/BUGLE/A06.wk.b70
rm ft*  2>/dev/null
end
```

TAB. A.3

Processing Job for Calculating PWR Weighting Spectra.
(Table corresponding to TAB. A.7 in reference /8/)

```
#!/bin/csh
# shell script to execute a scampi job
=shell
rm ft*  2>/dev/null
ln -sf /home/BUGLE/A05.nitawl.b70 ft04f001
end
=xsdrnpm
epri np-152 pwr problem no pads 3425 mwt flat power dist.
-1$$ 2000000
1$$ 2   8 119 1 0
     5  78   8 3 0
    40 100   0 0 3
2$$ -2 33 0 3z -1 1 e
3$$ 1 1 0 0 0
     0 0 1 0 0
     0 0
4$$ 0 67 41 -2 3
     4 70 -1 0
5** 0.00001 0.00001 7.2676e+17 2z
    1.420892 4z
    0.001 0.75 t
13$$ 15r1 3r2 18r3 18r4
     24r5
'      fuel water 55304 a533b concrete
14$$ 100113 501013 801613
     4009012 4009112 4009212 4009412 4009612
     2605412 2605612 2605712 2605812
     9223511 9223811 801611
     100113 501013  801631
     601252
     1402841 1402941 1403041
     2405052 2405252 2405352 2405452
     2505552
     2605452 2605652 2605752 2605852
     2805852 2806052 2806152 2806252 2806452
     601251
     1402841 1402941 1403041
     2405051 2405251 2405351 2405451
     2505551
     2605451 2605651 2605751 2605851
     2805851 2806051 2806151 2806251 2806451
     100141 601241 801641 1102341
     1202441 1202541 1202641
     1302741
     1402841 1402941 1403041
     1903941 1904041 1904141
     2004041 2004241 2004341 2004441 2004641 2004841
     2605441 2605641 2605741 2605841
15** 2.768e-2 2.466e-6 1.384e-2
     2.19023E-03 4.77635E-04 7.30076E-04 7.39867E-04 1.19196E-04
     8.5196e-7 1.3244368e-5 3.0324e-7 4.0432e-8
```

TAB. A.3 Continued

Processing Job for Calculating PWR Weighting Spectra.
(Table corresponding to TAB. A.7 in reference /8/)

```
        1.903e-4 6.515e-3 1.343e-2
        4.714e-2 4.200e-6 2.357e-2
        2.37e-4
        8.23614E-4 4.17031E-5 2.76830E-5
        7.5603e-4 1.457946e-2 1.653e-3 4.1151e-4
        1.52e-3
        3.4397e-3 5.347276e-2 1.2243e-3 1.6324e-4
        5.837085e-3 2.23155e-3 9.6615e-5 3.06945e-4 7.7805e-5
        9.81e-4
        3.42173E-4 1.73257E-5 1.15010E-5
        5.51815e-6 1.064133e-4 1.2065e-5 3.00355e-6
        1.12e-3
        4.832094e-3 7.5118588e-2 1.719898e-3 2.2932e-4
        3.031188e-4 1.15884e-4 5.0172e-6 1.59396e-5 4.0404e-6
        7.77e-3 1.15e-4 4.38e-2 1.05e-3
        1.16905E-04 1.48000E-05 1.62948E-05
        2.39e-3
        1.45723E-2 7.37860E-4 4.89800E-4
        6.46279E-04 8.10810E-08 4.66403E-05
        2.83068E-3 1.88924E-5 3.942E-6 6.09112E-5 1.16800E-7 5.46040E-6
        1.8467e-5 2.870836e-4 6.573e-6 8.764e-7
t
30$$ 46r1 f0
' pwr source spectrum for 199-group structure
31**
944991-14 408572-11 410082-11 1407-8 2346-8 1676-8 212-7 2669-8 3318-8
9212-8 689696-10 689304-10 2022-7 2892-7 4059-7 5583-7 7525-7 9963-7
1297-6 166-5 2092-6 2597-6 3179-6 3839-6 144-5 3137-6 539-5 6274-6
7224-6 8229-6 9282-6 1037-5 1148-5 2632-5 3072-5 3482-5 188-4 1965-5
203858-7 210743-7 2164-5 2212-5 1496-5 3787-6 3769-6 758-5 1523-5 23-3
231-4 2312-5 2306-5 229-4 2266-5 2237-5 2202-5 2162-5 2111-5 20628-6
200921-7 1953-5 189-4 3591-5 1422-5 1912-5 1568-5 1503-5 1438-5 1373-5
131-4 1248-5 1188-5 1128-5 1071-5 1015-5 9643-6 9099-6 1672-5 1488-5
6804-6 6405-6 1168-5 1031-5 111318-8 408612-9 857096-9 2305-6 4395-6
7978-6 3614-6 3382-6 3164-6 296-5 2765-6 2584-6 2417-6 2253-6 2107-6
1962-6 1834-6 1708-6 1595-6 1487-6 1383-6 1291-6 2853-6 2386-6 800687-9
588195-9 142617-8 846046-9 1879-6 6696-7 9652-7 8036-7 9037-7 3215-7
412869-9 179031-9 11-5 1452-7 6873-8 6624-8 1846-7 2654-7 4036-7 2781-7
766153-10 114985-9 1319-7 9077-8 6245-8 2768-8 1528-8 1316-8 1132-8
5056-9 4693-9 8395-9 7227-9 1397-8 9602-9 6601-9 4538-9 3119-9 2144-9
147411-11 101289-11 6962-10 4784-10 329-9 226-9 1553-10 1067-10 7339-11
504437-13 345483-13 23938-12 163763-13 112537-13 7737-12 5317-12 3654-12
2512-12 1727-12 1186-12 813315-15 562685-15 385-12 112103-15 152962-15
299068-16 274697-16 285471-16 960036-16 45626-15 793822-16 302382-16
556619-16 144778-16 446222-16 97115-16 970882-17 13405-15 161026-16
161422-16 163938-16 146302-16 179059-16 596373-18 562593-18 373104-18
481014-18 596373-18 619278-18 621267-18 115897-17 153207-17 231793-17
654104-17 f0
t
33** f0
t
```

TAB. A.3 Continued

Processing Job for Calculating PWR Weighting Spectra.
(Table corresponding to TAB. A.7 in reference /8/)

```
35** 29i 0 9i 120 5i 151.66 2i 168.51 7i 171.40
     4i 187.96 4i 193.68 9i 200.66 14i 219.71 241.62
     4i242 259.5 8i 260 10i 275 300
36$$ 46r1 3r2 8r3 5r4 15r5 15r6 7r7 20r8
38** 92r 1 7r 1.0-10 f1
39$$ 1 3 2 3 2 4 2 5
40$$ f3
51$$
 0 6r1 5r2 4r3 3r4 3r5 5r6 4r7 4r8 3r9 2r10 2r11 2r12 13 2r14 3r15
 3r16 4r17 5r18 4r19 2r20 4r21 4r22 4r23 4r24 10r25 10r26 6r27 4r28
 2r29 3r30 2r31 2r32 2r33 4r34 4r35 7r36 5r37 3r38 3r39 4r40 5r41
 3r42 4r43 7r44 6r45 8r46 11r47 2z 2r48 49 2r50 2r51 2r52 2r53 2r54
 2r55 2r56 3r57 58 59 60 4r61 2r62 2r63 3r64 3r65 66 67 0
t
end
=shell
cp ft33f001 /home/BUGLE/A07.scalar.b70
cp ft03f001 /home/BUGLE/A07.wk.b70
rm ft*  2>/dev/null
end
```

TAB. A.4

Processing Job for Collapsing BWR Core Nuclides without Upscattering Cross Sections.
(Table corresponding to TAB. A.8a in reference /8/)

```
#!/bin/csh
# shell script to execute a scampi job
=shell
rm ft*  2>/dev/null
ln -sf /home/BUGLE/A05.bonami.b70 ft85f001
end
=ajax
0$$ 82 85 e
1$$ 1 t
2$$ 85 14 t
3$$ 100123 801623
     4009022 4009122 4009222 4009422 4009622
     2605422 2605622 2605722 2605822
     9223521 9223821 801621  t
end
=malocs
1$$ 199 47 42 20 e
2$$ 82 81
3$$ 0 0 1 1 a7 4 f0 t
4$$
 0 6r1 5r2 4r3 3r4 3r5 5r6 4r7 4r8 3r9 2r10
 2r11 2r12 13 2r14 3r15 3r16 4r17 5r18 4r19 2r20
 4r21 4r22 4r23 4r24 10r25 10r26 6r27 4r28 2r29 3r30
 2r31 2r32 2r33 4r34 4r35 7r36 5r37 3r38 3r39 4r40
 5r41 3r42 4r43 7r44 6r45 8r46 11r47
5**
' neutron weighting flux for bwr core (int #57)
 7.40252E+05 2.83438E+08 3.18948E+08 1.05397E+09 1.73532E+09 1.23262E+09
 1.57869E+09 1.99345E+09 2.46470E+09 6.79071E+09 5.03852E+09 5.20218E+09
 1.43653E+10 2.03579E+10 3.07288E+10 4.02557E+10 5.50661E+10 7.47688E+10
 9.49337E+10 1.19266E+11 1.55110E+11 1.92202E+11 2.32282E+11 2.94954E+11
 1.12626E+11 2.51094E+11 4.29003E+11 4.73100E+11 5.32084E+11 6.46258E+11
 7.04815E+11 8.11780E+11 8.55237E+11 1.93796E+12 2.07768E+12 2.29459E+12
 1.36101E+12 1.61082E+12 1.62425E+12 1.75053E+12 1.91767E+12 1.93089E+12
 1.33790E+12 3.62977E+11 3.70551E+11 7.28645E+11 1.32764E+12 1.86765E+12
 1.81586E+12 1.77119E+12 1.70465E+12 2.01629E+12 1.96835E+12 1.91734E+12
 2.00230E+12 2.09942E+12 2.05584E+12 1.71850E+12 2.14030E+12 2.14836E+12
 1.99858E+12 3.29285E+12 1.28852E+12 2.15793E+12 2.08214E+12 2.40202E+12
 2.66453E+12 2.55349E+12 2.38119E+12 2.29647E+12 2.23540E+12 2.17482E+12
 2.11484E+12 2.05465E+12 1.99449E+12 1.91589E+12 3.19470E+12 2.45172E+12
 1.43887E+12 1.44354E+12 3.28250E+12 3.34722E+12 3.74362E+11 1.37652E+11
 2.89391E+11 7.85169E+11 1.53100E+12 2.91248E+12 1.39100E+12 1.34581E+12
 1.30940E+12 1.27867E+12 1.24630E+12 1.21227E+12 1.18281E+12 1.15218E+12
 1.12743E+12 1.09954E+12 1.07624E+12 1.04842E+12 1.02558E+12 1.00355E+12
 9.80438E+11 9.62918E+11 2.32574E+12 2.21496E+12 8.29178E+11 6.19066E+11
 1.65961E+12 1.05478E+12 2.75280E+12 1.13112E+12 1.82201E+12 1.74308E+12
 2.46292E+12 1.01567E+12 1.46875E+12 7.04528E+11 4.60463E+11 6.45215E+11
 3.16047E+11 3.14117E+11 9.57705E+11 1.53432E+12 3.07949E+12 2.97480E+12
 1.16827E+12 1.72351E+12 2.81682E+12 2.82111E+12 2.75639E+12 1.63570E+12
 1.09197E+12 1.07512E+12 1.04798E+12 5.08887E+11 5.82291E+11 1.04740E+12
 1.07857E+12 2.60548E+12 2.59656E+12 2.59201E+12 2.54306E+12 2.51047E+12
```

TAB. A.4 Continued

Processing Job for Collapsing BWR Core Nuclides without Upscattering Cross Sections.
(Table corresponding to TAB. A.8a in reference /8/)

```
    2.48293E+12 2.46192E+12 2.43763E+12 2.38879E+12 2.37932E+12 2.33877E+12
    2.14575E+12 2.33910E+12 2.22104E+12 2.21887E+12 2.11935E+12 2.07572E+12
    2.09700E+12 1.77267E+12 1.98641E+12 1.96286E+12 1.90563E+12 1.42093E+12
    1.56316E+12 1.76669E+12 1.76539E+12 1.79850E+12 1.80912E+12 1.83634E+12
    7.82466E+11 1.05768E+12 2.98921E+11 2.76732E+11 2.89851E+11 9.88161E+11
    6.94292E+11 1.23068E+12 6.96363E+11 1.31638E+12 5.12472E+11 1.65243E+12
    1.14148E+12 1.23764E+12 1.92728E+12 2.76149E+12 3.29327E+12 3.75875E+12
    3.49969E+12 4.20295E+12 5.98185E+12 4.29900E+12 2.06738E+12 1.87700E+12
    1.42980E+12 8.13020E+11 4.23024E+11 3.07979E+11 9.15979E+10 1.69243E+10
    1.16163E+09
6$$ 2z 2r1 2 2r3 2r4 2r5 2r6 2r7 2r8 2r9 3r10 11 12 13 4r14 2r15
      2r16 3r17 3r18 19 20 0
7**
' photon weighting flux for bwr core (int #57)
 5.08310E+02 3.76940E+05 3.02683E+06 7.06180E+07 4.09433E+09 1.95593E+10
 5.29249E+10 1.02024E+11 1.94978E+11 2.76834E+11 3.69974E+11 4.86314E+11
 1.39601E+12 1.94758E+12 2.95680E+12 5.44471E+12 1.09139E+13 9.52250E+12
 5.30835E+12 6.31140E+12 4.27827E+11 1.61079E+13 1.26629E+13 6.84048E+12
 7.64646E+12 6.71356E+12 3.94788E+12 4.50207E+12 3.39601E+12 5.60618E+12
 3.44193E+12 6.37528E+11 2.84587E+11 1.17873E+11 1.12427E+10 1.35047E+10
 2.80562E+10 4.53002E+09 3.12688E+09 1.46218E+09 1.12596E+09 3.96909E+07
t
end
=nitawl
6$$ a3 500 1000 e
0$$ 81 e
1$$ 0 14 a11 -1 e t
2$$ 100123 801623
      4009022 4009122 4009222 4009422 4009622
      2605422 2605622 2605722 2605822
      9223521 9223821 801621
4**  11r600. 3r1000. e t
end
=rade
1$$ 0 4 e
2$$ a2 100 1 e t
end
=alpo
0$$ 87 4
1$$ 1 3 4 70  7 -1 -1 e t
2$$ 4 0 t
end
=shell
mv ft87f001 /home/BUGLE/bugle.b70.bcore
rm ft*  2>/dev/null
end
```

TAB. A.5

Processing Job for Collapsing PWR Core Nuclides without Upscattering Cross Sections.
(Table corresponding to TAB. A.9 in reference /8/)

```
#!/bin/csh
# shell script to execute a scampi job
=shell
rm ft*  2>/dev/null
ln -sf /home/BUGLE/A05.bonami.b70 ft85f001
end
=ajax
0$$ 82 85 e
1$$ 1 t
2$$ 85 24 t
3$$ 100113 501013 801613
 2405012 2405212 2405312 2405412
 2605412 2605612 2605712 2605812
 2805812 2806012 2806112 2806212 2806412
 4009012 4009112 4009212 4009412 4009612
 9223511 9223811 801611 t
end
=malocs
1$$ 199 47 42 20 e
2$$ 82 81
3$$ 0 0 1 1 a7 4 f0 t
4$$
 0 6r1 5r2 4r3 3r4 3r5 5r6 4r7 4r8 3r9 2r10
 2r11 2r12 13 2r14 3r15 3r16 4r17 5r18 4r19 2r20
 4r21 4r22 4r23 4r24 10r25 10r26 6r27 4r28 2r29 3r30
 2r31 2r32 2r33 4r34 4r35 7r36 5r37 3r38 3r39 4r40
 5r41 3r42 4r43 7r44 6r45 8r46 11r47
5**
' neutron weighting flux for pwr mid-core (int #37)
 1.16502E+06 4.37316E+08 4.98203E+08 1.63524E+09 2.67940E+09 1.89823E+09
 2.43521E+09 3.07489E+09 3.79465E+09 1.04244E+10 7.71622E+09 7.97805E+09
 2.17952E+10 3.07577E+10 4.69482E+10 6.08275E+10 8.32652E+10 1.13301E+11
 1.42872E+11 1.78460E+11 2.32989E+11 2.87974E+11 3.46432E+11 4.42050E+11
 1.68885E+11 3.77284E+11 6.42419E+11 7.01539E+11 7.85275E+11 9.60782E+11
 1.03944E+12 1.20031E+12 1.25525E+12 2.82854E+12 2.98451E+12 3.26000E+12
 1.93402E+12 2.30175E+12 2.30252E+12 2.47358E+12 2.70265E+12 2.70580E+12
 1.87359E+12 5.10742E+11 5.21203E+11 1.01739E+12 1.82670E+12 2.54769E+12
 2.46202E+12 2.38823E+12 2.28214E+12 2.69829E+12 2.60346E+12 2.52281E+12
 2.62650E+12 2.73625E+12 2.65854E+12 2.20834E+12 2.75330E+12 2.73287E+12
 2.52663E+12 4.15534E+12 1.62332E+12 2.69570E+12 2.56849E+12 2.92881E+12
 3.21126E+12 3.03994E+12 2.81789E+12 2.71027E+12 2.63054E+12 2.55335E+12
 2.47912E+12 2.40557E+12 2.33320E+12 2.24125E+12 3.76651E+12 2.93661E+12
 1.72548E+12 1.72015E+12 3.85757E+12 3.88685E+12 4.32182E+11 1.58900E+11
 3.34095E+11 9.06733E+11 1.76701E+12 3.35788E+12 1.60252E+12 1.55277E+12
 1.51240E+12 1.47808E+12 1.44196E+12 1.40327E+12 1.36987E+12 1.33559E+12
 1.30780E+12 1.27681E+12 1.25065E+12 1.21861E+12 1.19269E+12 1.16788E+12
 1.14183E+12 1.12227E+12 2.71503E+12 2.59336E+12 9.64309E+11 7.31850E+11
 1.93645E+12 1.24823E+12 3.21738E+12 1.32917E+12 2.15293E+12 2.07974E+12
 2.87474E+12 1.19486E+12 1.72992E+12 8.33818E+11 5.43818E+11 7.61273E+11
 3.76284E+11 3.74856E+11 1.12902E+12 1.83727E+12 3.63614E+12 3.53583E+12
 1.39278E+12 2.06412E+12 3.39001E+12 3.36555E+12 3.31179E+12 1.97004E+12
```

TAB. A.5 Continued

Processing Job for Collapsing PWR Core Nuclides without Upscattering Cross Sections.
(Table corresponding to TAB. A.9 in reference /8/)

```
   1.31341E+12 1.30021E+12 1.28346E+12 6.30849E+11 6.76075E+11 1.27245E+12
   1.29949E+12 3.17343E+12 3.16295E+12 3.15963E+12 3.11555E+12 3.08582E+12
   3.06222E+12 3.04597E+12 3.02675E+12 2.97748E+12 2.97014E+12 2.93441E+12
   2.74188E+12 2.93436E+12 2.81414E+12 2.81545E+12 2.72342E+12 2.66116E+12
   2.69787E+12 2.34300E+12 2.58671E+12 2.55441E+12 2.49369E+12 1.95081E+12
   2.14293E+12 2.36858E+12 2.36479E+12 2.40826E+12 2.42600E+12 2.46572E+12
   1.05228E+12 1.42590E+12 4.03689E+11 3.73842E+11 3.91598E+11 1.33537E+12
   9.38860E+11 1.66595E+12 9.43909E+11 1.78779E+12 6.97674E+11 2.26050E+12
   1.57585E+12 1.72548E+12 2.72438E+12 3.97179E+12 4.81420E+12 5.56951E+12
   5.24357E+12 6.34823E+12 9.08805E+12 6.56678E+12 3.17272E+12 2.89101E+12
   2.20922E+12 1.25964E+12 6.56862E+11 4.79196E+11 1.42843E+11 2.64428E+10
   1.81458E+09
6$$ 2z 2r1 2 2r3 2r4 2r5 2r6 2r7 2r8 2r9 3r10 11 12 13 4r14 2r15
      2r16 3r17 3r18 19 20 0
7**
' photon weighting flux for pwr mid-core (int #37)
 4.75793E+02 3.71995E+05 4.48099E+06 1.14221E+08 2.57284E+09 2.31267E+10
 8.22118E+10 1.54236E+11 3.32248E+11 5.16677E+11 6.85877E+11 8.79398E+11
 2.38736E+12 3.40460E+12 5.21945E+12 9.55528E+12 1.87469E+13 1.64003E+13
 9.39297E+12 1.12377E+13 7.68953E+11 2.88298E+13 2.21083E+13 1.21875E+13
 1.34530E+13 1.16283E+13 6.78281E+12 8.37130E+12 5.88165E+12 9.62183E+12
 5.79936E+12 1.06405E+12 4.51121E+11 1.88561E+11 1.80738E+10 2.18428E+10
 3.86140E+10 6.42594E+09 4.75485E+09 2.21746E+09 1.68560E+09 6.88885E+07
t
end
=nitawl
6$$ a3 500 1000 e
0$$ 81 e
1$$ 0  24 a11 -1 e t
2$$ 100113 501013 801613
 2405012 2405212 2405312 2405412
 2605412 2605612 2605712 2605812
 2805812 2806012 2806112 2806212 2806412
 4009012 4009112 4009212 4009412 4009612
 9223511 9223811 801611
4**  21r600. 3r1000. e t
end
=rade
1$$ 0 4 e
2$$ a2 100 1 e t
end
=alpo
0$$ 87 4
1$$ 1 3 4 70  7 -1 -1 e t
2$$ 4 0 t
end
=shell
mv ft87f001 /home/BUGLE/bugle.b70.pcore
rm ft*  2>/dev/null
end
```

TAB. A.6

Processing Job for Collapsing PWR Downcomer Nuclides without Upscattering
Cross Sections.
(Table corresponding to TAB. A.10 in reference /8/)

```
#!/bin/csh
# shell script to execute a scampi job
=shell
rm ft*  2>/dev/null
ln -sf /home/BUGLE/A05.bonami.b70 ft85f001
end
=ajax
0$$ 82 85 e
1$$ 1 t
2$$ 85 17 t
3$$ 100131 801631
 2605452 2605652 2605752 2605852
 2405052 2405252 2405352 2405452
 2805852 2806052 2806152 2806252 2806452
 2505552  601252 t
end
=malocs
1$$ 199 47 42 20 e
2$$ 82 81
3$$ 0 0 1 1 a7 4 f0 t
4$$
 0 6r1 5r2 4r3 3r4 3r5 5r6 4r7 4r8 3r9 2r10
 2r11 2r12 13 2r14 3r15 3r16 4r17 5r18 4r19 2r20
 4r21 4r22 4r23 4r24 10r25 10r26 6r27 4r28 2r29 3r30
 2r31 2r32 2r33 4r34 4r35 7r36 5r37 3r38 3r39 4r40
 5r41 3r42 4r43 7r44 6r45 8r46 11r47
5**
' neutron weighting flux for downcomer (int #69)
 1.36811E+04 3.49191E+06 5.11997E+06 1.54934E+07 2.32491E+07 1.56887E+07
 2.02220E+07 2.53289E+07 3.03811E+07 7.96514E+07 5.60967E+07 6.06219E+07
 1.40484E+08 1.84563E+08 3.12512E+08 3.58747E+08 4.82579E+08 6.54697E+08
 7.57528E+08 8.58844E+08 1.12132E+09 1.32094E+09 1.47074E+09 1.90589E+09
 7.19232E+08 1.62465E+09 2.66295E+09 2.56331E+09 2.63662E+09 3.37395E+09
 3.24197E+09 3.74636E+09 3.65371E+09 7.46210E+09 6.62847E+09 6.43376E+09
 3.75558E+09 4.87419E+09 4.98147E+09 5.25465E+09 5.75226E+09 5.61337E+09
 4.23665E+09 1.30515E+09 1.43299E+09 2.46237E+09 4.30318E+09 5.66744E+09
 5.32131E+09 5.12939E+09 4.66833E+09 5.76389E+09 5.45524E+09 5.12003E+09
 5.34608E+09 5.37843E+09 5.26895E+09 4.10955E+09 5.21138E+09 5.35771E+09
 4.85049E+09 7.56223E+09 2.85576E+09 4.79822E+09 4.42741E+09 4.89701E+09
 5.09579E+09 4.77960E+09 4.59172E+09 4.46191E+09 4.40372E+09 4.29152E+09
 4.21342E+09 4.07308E+09 3.94075E+09 3.78983E+09 6.45885E+09 5.19982E+09
 3.03023E+09 3.01888E+09 6.42937E+09 6.34185E+09 7.06863E+08 2.60383E+08
 5.47723E+08 1.48586E+09 2.89862E+09 5.55309E+09 2.67279E+09 2.60978E+09
 2.55946E+09 2.51284E+09 2.46271E+09 2.40878E+09 2.36350E+09 2.31200E+09
 2.27374E+09 2.23071E+09 2.19039E+09 2.14349E+09 2.10634E+09 2.07026E+09
 2.03037E+09 2.00145E+09 4.87124E+09 4.69878E+09 1.74287E+09 1.35240E+09
 3.52905E+09 2.31373E+09 5.93014E+09 2.47056E+09 4.04000E+09 3.94836E+09
 5.39215E+09 2.26716E+09 3.29198E+09 1.59802E+09 1.05175E+09 1.47627E+09
 7.35641E+08 7.31226E+08 2.18146E+09 3.59639E+09 7.08839E+09 7.00569E+09
 2.78411E+09 4.15562E+09 6.88689E+09 6.87145E+09 6.86388E+09 4.12016E+09
```

TAB. A.6 Continued

Processing Job for Collapsing PWR Downcomer Nuclides without Upscattering
Cross Sections.
(Table corresponding to TAB. A.10 in reference /8/)

```
    2.75016E+09 2.75180E+09 2.75369E+09 1.37631E+09 1.37637E+09 2.75447E+09
    2.76474E+09 6.93496E+09 6.96021E+09 6.98732E+09 7.02654E+09 7.06566E+09
    7.10586E+09 7.13871E+09 7.19511E+09 7.23801E+09 7.27947E+09 7.32467E+09
    7.36641E+09 7.41016E+09 7.45384E+09 7.49732E+09 7.54068E+09 7.58287E+09
    7.62553E+09 7.66571E+09 7.70814E+09 7.74953E+09 7.78791E+09 7.82658E+09
    7.86457E+09 8.07897E+09 8.15818E+09 8.29058E+09 8.44431E+09 8.63321E+09
    3.71844E+09 5.15341E+09 1.48493E+09 1.37189E+09 1.43378E+09 4.88970E+09
    3.45530E+09 6.21083E+09 3.60980E+09 7.28111E+09 3.13962E+09 1.24793E+10
    1.20246E+10 1.75334E+10 3.76891E+10 7.34161E+10 1.09913E+11 1.47084E+11
    1.52983E+11 1.99705E+11 3.07088E+11 2.35513E+11 1.18198E+11 1.10666E+11
    8.67863E+10 5.05733E+10 2.67990E+10 1.98147E+10 5.96945E+09 1.10729E+09
    7.50930E+07
6$$ 2z 2r1 2 2r3 2r4 2r5 2r6 2r7 2r8 2r9 3r10 11 12 13 4r14 2r15
      2r16 3r17 3r18 19 20 0
7**
' photon weighting flux for downcomer (int #69)
    8.44980E+03 4.49962E+05 3.09681E+06 2.10447E+08 1.02453E+11 1.60514E+11
    4.92061E+10 3.40178E+10 5.49568E+10 5.73076E+10 3.93513E+10 4.77376E+10
    5.98845E+10 6.53919E+10 8.56238E+10 1.07983E+11 5.54998E+11 1.90018E+11
    1.00535E+11 1.17853E+11 8.28999E+09 2.84537E+11 2.56470E+11 1.47640E+11
    1.73903E+11 1.85041E+11 1.12793E+11 3.83180E+11 2.09552E+11 5.46888E+11
    9.24046E+11 7.51354E+11 1.09594E+12 8.16262E+11 1.80494E+11 3.44137E+11
    5.11948E+11 1.18051E+11 1.09115E+11 1.07884E+10 7.96993E+07 3.57505E+03
t
end
=nitawl
6$$ a3 500 1000 e
0$$ 81 e
1$$ 0 17 a11 -1 e t
2$$ 100131 801631
 2605452 2605652 2605752 2605852
 2405052 2405252 2405352 2405452
 2805852 2806052 2806152 2806252 2806452
 2505552  601252
4**  16r600. 300. e t
end
=rade
1$$ 0 4 e
2$$ a2 100 1 e t
end
=alpo
0$$ 87 4
1$$ 1 3 4 70  7 -1 -1 e t
2$$ 4 0 t
end
=shell
mv ft87f001 /home/BUGLE/bugle.b70.pdown
rm ft*  2>/dev/null
end
```

TAB. A.7

Processing Job for Collapsing Pressure Vessel Nuclides without Upscattering
Cross Sections.
(Table corresponding to TAB. A.11 in reference /8/)

```
#!/bin/csh
# shell script to execute a scampi job
=shell
rm ft*  2>/dev/null
ln -sf /home/BUGLE/A05.bonami.b70 ft85f001
end
=ajax
0$$ 82 85 e
1$$ 1 t
2$$ 85 15 t
3$$ 2605451 2605651 2605751 2605851
     2405051 2405251 2405351 2405451
     2805851 2806051 2806151 2806251 2806451
     2505551  601251 t
end
=malocs
1$$ 199 47 42 20 e
2$$ 82 81
3$$ 0 0 1 1 a7 4 f0 t
4$$ 0 6r1 5r2 4r3 3r4 3r5 5r6 4r7 4r8 3r9 2r10
 2r11 2r12 13 2r14 3r15 3r16 4r17 5r18 4r19 2r20
 4r21 4r22 4r23 4r24 10r25 10r26 6r27 4r28 2r29 3r30
 2r31 2r32 2r33 4r34 4r35 7r36 5r37 3r38 3r39 4r40
 5r41 3r42 4r43 7r44 6r45 8r46 11r47
5**
' neutron weighting flux for pwr 1/4 t (int #82)
 1.98286E+03 4.12419E+05 6.74571E+05 2.01414E+06 2.86724E+06 1.87945E+06
 2.40344E+06 2.98969E+06 3.54185E+06 9.10513E+06 6.23704E+06 6.83381E+06
 1.49593E+07 1.87347E+07 3.23356E+07 3.68533E+07 4.86868E+07 6.47930E+07
 7.33177E+07 7.97092E+07 1.00240E+08 1.16074E+08 1.25903E+08 1.60050E+08
 5.98672E+07 1.34826E+08 2.20989E+08 2.07112E+08 2.04680E+08 2.52061E+08
 2.36681E+08 2.61993E+08 2.63978E+08 5.23566E+08 4.83027E+08 4.79840E+08
 2.63889E+08 3.29144E+08 3.82940E+08 4.13807E+08 4.75586E+08 4.53583E+08
 3.58084E+08 1.10563E+08 1.25313E+08 2.02685E+08 4.30656E+08 5.99451E+08
 5.71173E+08 5.55166E+08 5.59530E+08 6.89975E+08 7.47133E+08 7.04774E+08
 7.43026E+08 7.16585E+08 7.70572E+08 7.96525E+08 7.25915E+08 1.11065E+09
 1.02499E+09 1.68271E+09 6.71153E+08 1.26677E+09 8.97869E+08 9.95587E+08
 8.68350E+08 8.95821E+08 1.28827E+09 1.52894E+09 2.03461E+09 2.09642E+09
 1.49318E+09 1.28869E+09 1.19034E+09 1.07621E+09 2.34155E+09 1.53934E+09
 6.79412E+08 1.74529E+09 3.16893E+09 3.31883E+09 4.00891E+08 1.31034E+08
 2.11064E+08 3.40305E+08 7.38785E+08 1.67533E+09 9.96178E+08 5.51941E+08
 9.49496E+08 5.61242E+08 3.38071E+08 7.86818E+08 1.09478E+09 8.72316E+08
 7.90149E+08 4.67115E+08 2.45858E+08 1.15770E+09 5.33216E+08 9.33786E+08
 5.84250E+08 5.45515E+08 9.36121E+08 6.81933E+08 1.96150E+08 8.80293E+08
 6.84525E+08 8.48741E+08 1.23093E+09 3.72432E+08 8.56309E+08 6.11765E+08
 6.25553E+08 2.07449E+08 1.05483E+08 3.21751E+07 1.72690E+08 1.20493E+09
 5.75697E+08 3.03363E+08 6.81746E+08 6.49265E+08 9.18392E+08 8.59427E+08
 3.25802E+08 3.61344E+08 3.70978E+08 9.77042E+08 1.02074E+09 5.54601E+08
 4.02390E+08 3.77637E+08 3.33417E+08 1.45977E+08 1.19905E+08 1.74150E+08
 2.57302E+08 8.50979E+08 8.63652E+08 7.03852E+08 6.94778E+08 6.85140E+08
```

TAB. A.7 Continued

Processing Job for Collapsing Pressure Vessel Nuclides without Upscattering
Cross Sections.
(Table corresponding to TAB. A.11 in reference /8/)

```
   6.58788E+08 3.93691E+08 5.87024E+08 6.25235E+08 6.40081E+08 6.46994E+08
   6.49635E+08 6.49705E+08 6.47403E+08 6.42781E+08 6.35752E+08 6.26110E+08
   6.13851E+08 5.98639E+08 5.80769E+08 5.59969E+08 5.36150E+08 5.09666E+08
   4.80648E+08 4.58460E+08 4.25999E+08 3.91763E+08 3.57633E+08 3.24961E+08
   1.29981E+08 1.66702E+08 4.56819E+07 4.10312E+07 4.15696E+07 1.32915E+08
   8.64937E+07 1.40239E+08 7.32891E+07 1.28077E+08 4.72245E+07 1.45558E+08
   9.78623E+07 1.05921E+08 1.62849E+08 2.18944E+08 2.33826E+08 2.33775E+08
   1.89884E+08 2.02638E+08 2.60926E+08 1.66137E+08 6.81521E+07 5.13874E+07
   3.07688E+07 1.33437E+07 5.31916E+06 2.87523E+06 5.95191E+05 8.06397E+04
   4.42195E+03
6$$ 2z 2r1 2 2r3 2r4 2r5 2r6 2r7 2r8 2r9 3r10 11 12 13 4r14 2r15
     2r16 3r17 3r18 19 20 0
7**
' photon weighting flux for pwr 1/4 t (int #82)
   1.25708E+03 4.84585E+04 3.15930E+05 1.93070E+07 8.86365E+09 1.43832E+10
   5.03881E+09 3.87113E+09 5.83878E+09 6.31574E+09 5.15435E+09 6.09643E+09
   7.47139E+09 8.43862E+09 1.05168E+10 1.30826E+10 2.70518E+10 1.93220E+10
   1.03535E+10 1.16818E+10 7.80635E+08 2.93842E+10 2.61362E+10 1.43987E+10
   1.69078E+10 1.77640E+10 2.34328E+10 1.98378E+10 1.87707E+10 4.72790E+10
   7.17299E+10 4.54291E+10 3.23703E+10 5.69409E+09 3.22647E+08 2.39756E+08
   2.58081E+07 4.17256E+05 1.76375E+05 2.05216E+05 3.14946E+05 5.69300E+03
t
end
=nitawl
6$$ a3 500 1000 e
0$$ 81 e
1$$ 0 15 a11 -1 e t
2$$ 2605451 2605651 2605751 2605851
     2405051 2405251 2405351 2405451
     2805851 2806051 2806151 2806251 2806451
     2505551  601251
4** 14r600. 300.  e t
end
=rade
1$$ 0 4 e
2$$ a2 100 1 e t
end
=alpo
0$$ 87 4
1$$ 1 3 4 70  7 -1 -1 e t
2$$ 4 0 t
end
=shell
mv ft87f001 /home/BUGLE/bugle.b70.pp14t
rm ft*  2>/dev/null
end
```

TAB. A.8

Processing Job for Collapsing Concrete Nuclides without Upscattering Cross Sections.
(Table corresponding to TAB. A.12 in reference /8/)

```
#!/bin/csh
# shell script to execute a scampi job
=shell
rm ft*  2>/dev/null
ln -sf /home/BUGLE/A05.bonami.b70 ft85f001
end
=ajax
0$$ 82 85 e
1$$ 1 t
2$$ 85 24 t
3$$ 100141 801641
 2605441 2605641 2605741 2605841
 601241  1102341
 1202441 1202541 1202641
 1302741
 1402841 1402941 1403041
 1903941 1904041 1904141
 2004041 2004241 2004341 2004441 2004641 2004841 t
end
=malocs
1$$ 199 47 42 20 e
2$$ 82 81
3$$ 0 0 1 1 a7 4 f0 t
4$$
 0 6r1 5r2 4r3 3r4 3r5 5r6 4r7 4r8 3r9 2r10
 2r11 2r12 13 2r14 3r15 3r16 4r17 5r18 4r19 2r20
 4r21 4r22 4r23 4r24 10r25 10r26 6r27 4r28 2r29 3r30
 2r31 2r32 2r33 4r34 4r35 7r36 5r37 3r38 3r39 4r40
 5r41 3r42 4r43 7r44 6r45 8r46 11r47
5**
' neutron weighting flux for concrete (xsdrnpm pwr calc. int #106)
 7.29347E+01 1.15858E+04 2.16810E+04 6.38868E+04 8.44980E+04 5.32721E+04
 6.64980E+04 8.24386E+04 9.72686E+04 2.44419E+05 1.66633E+05 1.85270E+05
 3.81228E+05 4.64720E+05 8.44247E+05 9.02182E+05 1.19494E+06 1.62534E+06
 1.79656E+06 1.85923E+06 2.30233E+06 2.58884E+06 2.77701E+06 3.59210E+06
 1.38828E+06 3.12911E+06 5.10439E+06 4.66159E+06 4.63761E+06 6.21271E+06
 5.86660E+06 6.52650E+06 6.40871E+06 1.36120E+07 1.19272E+07 1.13266E+07
 7.02453E+06 9.44667E+06 1.06145E+07 1.23500E+07 1.49920E+07 1.50563E+07
 1.37684E+07 4.20412E+06 5.52473E+06 9.09001E+06 1.58075E+07 1.97704E+07
 1.83115E+07 1.54698E+07 1.40926E+07 2.44114E+07 2.20573E+07 2.03030E+07
 2.45409E+07 2.71187E+07 2.81788E+07 1.96223E+07 2.87708E+07 3.40429E+07
 3.36810E+07 4.18245E+07 1.35009E+07 2.75173E+07 3.38958E+07 3.92836E+07
 4.72407E+07 5.23836E+07 5.78279E+07 6.47017E+07 7.52565E+07 8.56770E+07
 8.65734E+07 7.86350E+07 8.66430E+07 8.21358E+07 1.27787E+08 7.59820E+07
 4.44202E+07 4.66708E+07 1.23264E+08 1.48330E+08 1.72910E+07 6.47230E+06
 1.33075E+07 3.58180E+07 7.44322E+07 1.45110E+08 7.15691E+07 7.10896E+07
 6.39294E+07 6.25478E+07 5.91705E+07 5.66331E+07 6.52191E+07 8.05716E+07
 9.11865E+07 8.83925E+07 8.09986E+07 8.78136E+07 8.06470E+07 9.12996E+07
 7.96552E+07 8.08025E+07 1.90551E+08 1.65740E+08 6.29882E+07 5.50126E+07
 1.41789E+08 9.27476E+07 2.17575E+08 8.35274E+07 1.78699E+08 1.56769E+08
 1.98625E+08 9.15462E+07 1.36427E+08 6.35769E+07 4.39068E+07 6.21433E+07
```

TAB. A.8 Continued

Processing Job for Collapsing Concrete Nuclides without Upscattering Cross Sections.
(Table corresponding to TAB. A.12 in reference /8/)

```
    3.16168E+07 3.12031E+07 9.25966E+07 1.51759E+08 2.98772E+08 2.95250E+08
    1.17077E+08 1.72623E+08 2.89718E+08 2.85216E+08 2.88335E+08 1.69889E+08
    1.09253E+08 9.73156E+07 7.26310E+07 4.42706E+07 5.81257E+07 1.30973E+08
    1.25942E+08 3.05032E+08 2.95894E+08 2.92202E+08 2.91318E+08 2.91099E+08
    2.91042E+08 2.90483E+08 2.90458E+08 2.90311E+08 2.90033E+08 2.89790E+08
    2.89306E+08 2.88807E+08 2.88223E+08 2.87556E+08 2.86812E+08 2.85956E+08
    2.85047E+08 2.83978E+08 2.82920E+08 2.81750E+08 2.80389E+08 2.78955E+08
    2.77410E+08 2.81952E+08 2.79388E+08 2.80426E+08 2.81617E+08 2.83382E+08
    1.20518E+08 1.65361E+08 4.72695E+07 4.35180E+07 4.53019E+07 1.53165E+08
    1.06791E+08 1.87862E+08 1.05378E+08 1.96696E+08 7.52576E+07 2.34861E+08
    1.52838E+08 1.55474E+08 2.22645E+08 2.93228E+08 3.54638E+08 4.76902E+08
    5.76977E+08 9.69327E+08 2.17377E+09 2.41832E+09 1.54320E+09 1.71197E+09
    1.57635E+09 1.04794E+09 6.10169E+08 4.89143E+08 1.57783E+08 3.03672E+07
    2.07969E+06
6$$ 2z 2r1 2 2r3 2r4 2r5 2r6 2r7 2r8 2r9 3r10 11 12 13 4r14 2r15
      2r16 3r17 3r18 19 20 0
7**
' photon weighting flux for concrete (xsdrnpm pwr calc. int #106)
    3.15691E+01 9.79614E+02 4.59269E+03 1.43771E+06 9.44209E+07 3.41969E+08
    1.11170E+08 6.48519E+07 1.73983E+08 1.86948E+08 1.73882E+08 3.75841E+08
    1.80878E+08 4.73675E+08 2.70875E+08 3.57831E+08 8.26411E+08 4.36238E+08
    2.03888E+08 2.01998E+08 1.28382E+07 5.28507E+08 4.35567E+08 2.53035E+08
    2.82375E+08 3.10777E+08 2.86509E+08 3.43181E+08 3.12050E+08 8.32199E+08
    1.47642E+09 1.21393E+09 1.84323E+09 1.24205E+09 2.25526E+08 3.43201E+08
    2.11349E+08 1.82409E+07 5.75789E+06 2.79798E+05 3.75042E+04 2.67899E+03
t
end
=nitawl
6$$ a3 500 1000 e
0$$ 81 e
1$$ 0  24 a11 -1 e t
2$$ 100141 801641
 2605441 2605641 2605741 2605841
 601241 1102341
 1202441 1202541 1202641
 1302741
 1402841 1402941 1403041
 1903941 1904041 1904141
 2004041 2004241 2004341 2004441 2004641 2004841 t
end
=rade
1$$ 0 4 e
2$$ a2 100 1 e t
end
=alpo
0$$ 87 4
1$$ 1 3 4 70  7 -1 -1 e t
2$$ 4 0 t
end
=shell
mv ft87f001 /home/BUGLE/bugle.b70.pconc
```

TAB. A.8 Continued

Processing Job for Collapsing Concrete Nuclides without Upscattering Cross Sections.
(Table corresponding to TAB. A.12 in reference /8/)

```
    rm ft*  2>/dev/null
    end
```

TAB. A.9

Processing Job for Collapsing Carbon/Stainless Steel Nuclides without Upscattering
Cross Sections.
(Table corresponding to TAB. A.13 in reference /8/)

```
#!/bin/csh
# shell script to execute a scampi job
=shell
rm ft*  2>/dev/null
ln -sf /home/BUGLE/A05.bonami.b70 ft85f001
end
=ajax
0$$ 82 85 e
1$$ 1 t
2$$ 85 30 t
3$$
  2405051 2405251 2405351 2405451
  2605451 2605651 2605751 2605851
  2805851 2806051 2806151 2806251 2806451
  2505551 601251
  2405052 2405252 2405352 2405452
  2605452 2605652 2605752 2605852
  2805852 2806052 2806152 2806252 2806452
  2505552 601252 t
end
=malocs
1$$ 199 47 42 20 e
2$$ 82 81
3$$ 0 0 1 1 a7 4 f0 t
4$$
 0 6r1 5r2 4r3 3r4 3r5 5r6 4r7 4r8 3r9 2r10
 2r11 2r12 13 2r14 3r15 3r16 4r17 5r18 4r19 2r20
 4r21 4r22 4r23 4r24 10r25 10r26 6r27 4r28 2r29 3r30
 2r31 2r32 2r33 4r34 4r35 7r36 5r37 3r38 3r39 4r40
 5r41 3r42 4r43 7r44 6r45 8r46 11r47
5**
' neutron weighting flux for pwr 1/4 t (int #82)
 1.98286E+03 4.12419E+05 6.74571E+05 2.01414E+06 2.86724E+06 1.87945E+06
 2.40344E+06 2.98969E+06 3.54185E+06 9.10513E+06 6.23704E+06 6.83381E+06
 1.49593E+07 1.87347E+07 3.23356E+07 3.68533E+07 4.86868E+07 6.47930E+07
 7.33177E+07 7.97092E+07 1.00240E+08 1.16074E+08 1.25903E+08 1.60050E+08
 5.98672E+07 1.34826E+08 2.20989E+08 2.07112E+08 2.04680E+08 2.52061E+08
 2.36681E+08 2.61993E+08 2.63978E+08 5.23566E+08 4.83027E+08 4.79840E+08
 2.63889E+08 3.29144E+08 3.82940E+08 4.13807E+08 4.75586E+08 4.53583E+08
 3.58084E+08 1.10563E+08 1.25313E+08 2.02685E+08 4.30656E+08 5.99451E+08
 5.71173E+08 5.55166E+08 5.59530E+08 6.89975E+08 7.47133E+08 7.04774E+08
 7.43026E+08 7.16585E+08 7.70572E+08 7.96525E+08 7.25915E+08 1.11065E+09
 1.02499E+09 1.68271E+09 6.71153E+08 1.26677E+09 8.97869E+08 9.95587E+08
 8.68350E+08 8.95821E+08 1.28827E+09 1.52894E+09 2.03461E+09 2.09642E+09
 1.49318E+09 1.28869E+09 1.19034E+09 1.07621E+09 2.34155E+09 1.53934E+09
 6.79412E+08 1.74529E+09 3.16893E+09 3.31883E+09 4.00891E+08 1.31034E+08
 2.11064E+08 3.40305E+08 7.38785E+08 1.67533E+09 9.96178E+08 5.51941E+08
 9.49496E+08 5.61242E+08 3.38071E+08 7.86818E+08 1.09478E+09 8.72316E+08
 7.90149E+08 4.67115E+08 2.45858E+08 1.15770E+09 5.33216E+08 9.33786E+08
 5.84250E+08 5.45515E+08 9.36121E+08 6.81933E+08 1.96150E+08 8.80293E+08
```

TAB. A.9 Continued

Processing Job for Collapsing Carbon/Stainless Steel Nuclides without Upscattering
Cross Sections.
(Table corresponding to TAB. A.13 in reference /8/)

```
   6.84525E+08 8.48741E+08 1.23093E+09 3.72432E+08 8.56309E+08 6.11765E+08
   6.25553E+08 2.07449E+08 1.05483E+08 3.21751E+07 1.72690E+08 1.20493E+09
   5.75697E+08 3.03363E+08 6.81746E+08 6.49265E+08 9.18392E+08 8.59427E+08
   3.25802E+08 3.61344E+08 3.70978E+08 9.77042E+08 1.02074E+09 5.54601E+08
   4.02390E+08 3.77637E+08 3.33417E+08 1.45977E+08 1.19905E+08 1.74150E+08
   2.57302E+08 8.50979E+08 8.63652E+08 7.03852E+08 6.94778E+08 6.85140E+08
   6.58788E+08 3.93691E+08 5.87024E+08 6.25235E+08 6.40081E+08 6.46994E+08
   6.49635E+08 6.49705E+08 6.47403E+08 6.42781E+08 6.35752E+08 6.26110E+08
   6.13851E+08 5.98639E+08 5.80769E+08 5.59969E+08 5.36150E+08 5.09666E+08
   4.80648E+08 4.58460E+08 4.25999E+08 3.91763E+08 3.57633E+08 3.24961E+08
   1.29981E+08 1.66702E+08 4.56819E+07 4.10312E+07 4.15696E+07 1.32915E+08
   8.64937E+07 1.40239E+08 7.32891E+07 1.28077E+08 4.72245E+07 1.45558E+08
   9.78623E+07 1.05921E+08 1.62849E+08 2.18944E+08 2.33826E+08 2.33775E+08
   1.89884E+08 2.02638E+08 2.60926E+08 1.66137E+08 6.81521E+07 5.13874E+07
   3.07688E+07 1.33437E+07 5.31916E+06 2.87523E+06 5.95191E+05 8.06397E+04
   4.42195E+03
6$$ 2z 2r1 2 2r3 2r4 2r5 2r6 2r7 2r8 2r9 3r10 11 12 13 4r14 2r15
     2r16 3r17 3r18 19 20 0
7**
' photon weighting flux for pwr 1/4 t (int #82)
 1.25708E+03 4.84585E+04 3.15930E+05 1.93070E+07 8.86365E+09 1.43832E+10
 5.03881E+09 3.87113E+09 5.83878E+09 6.31574E+09 5.15435E+09 6.09643E+09
 7.47139E+09 8.43862E+09 1.05168E+10 1.30826E+10 2.70518E+10 1.93220E+10
 1.03535E+10 1.16818E+10 7.80635E+08 2.93842E+10 2.61362E+10 1.43987E+10
 1.69078E+10 1.77640E+10 2.34328E+10 1.98378E+10 1.87707E+10 4.72790E+10
 7.17299E+10 4.54291E+10 3.23703E+10 5.69409E+09 3.22647E+08 2.39756E+08
 2.58081E+07 4.17256E+05 1.76375E+05 2.05216E+05 3.14946E+05 5.69300E+03
t
end
=nitawl
6$$ a3 500 1000 e
0$$ 81 e
1$$ 0 30 a11 -1 e t
2$$
 2405051 2405251 2405351 2405451
  2605451 2605651 2605751 2605851
  2805851 2806051 2806151 2806251 2806451
  2505551 601251
  2405052 2405252 2405352 2405452
  2605452 2605652 2605752 2605852
  2805852 2806052 2806152 2806252 2806452
  2505552 601252
4** 14r600. 300. 14r600. 300. e t
end
=rade
1$$ 0 4 e
2$$ a2 100 1 e t
end
=alpo
0$$ 87 4
```

TAB. A.9 Continued

Processing Job for Collapsing Carbon/Stainless Steel Nuclides without Upscattering
Cross Sections.
(Table corresponding to TAB. A.13 in reference /8/)

```
1$$ 1 3 4 70  7 -1 -1 e t
2$$ 4 0 t
end
=shell
mv ft87f001 /home/BUGLE/bugle.b70.steel
rm ft*  2>/dev/null
end
```

TAB. A.10

Processing Job for Collapsing All VITJEFF311.BOLIB Nuclides with Concrete
Weighting Spectrum without Upscattering Cross Sections.
(Table corresponding to TAB. A.14 in reference /8/)

```
#!/bin/csh
# shell script to execute a scampi job
=shell
rm ft*  2>/dev/null
ln -sf /home/VITAMIN/vitamin.b70 ft01f001
end
=malocs
1$$ 199 47 42 20 e
2$$ 1 81
3$$ 0 0 1 1 a7 4 f0 t
4$$
 0 6r1 5r2 4r3 3r4 3r5 5r6 4r7 4r8 3r9 2r10
 2r11 2r12 13 2r14 3r15 3r16 4r17 5r18 4r19 2r20
 4r21 4r22 4r23 4r24 10r25 10r26 6r27 4r28 2r29 3r30
 2r31 2r32 2r33 4r34 4r35 7r36 5r37 3r38 3r39 4r40
 5r41 3r42 4r43 7r44 6r45 8r46 11r47
5**
' neutron weighting flux for concrete (xsdrnpm pwr calc. int #106)
 7.29347E+01 1.15858E+04 2.16810E+04 6.38868E+04 8.44980E+04 5.32721E+04
 6.64980E+04 8.24386E+04 9.72686E+04 2.44419E+05 1.66633E+05 1.85270E+05
 3.81228E+05 4.64720E+05 8.44247E+05 9.02182E+05 1.19494E+06 1.62534E+06
 1.79656E+06 1.85923E+06 2.30233E+06 2.58884E+06 2.77701E+06 3.59210E+06
 1.38828E+06 3.12911E+06 5.10439E+06 4.66159E+06 4.63761E+06 6.21271E+06
 5.86660E+06 6.52650E+06 6.40871E+06 1.36120E+07 1.19272E+07 1.13266E+07
 7.02453E+06 9.44667E+06 1.06145E+07 1.23500E+07 1.49920E+07 1.50563E+07
 1.37684E+07 4.20412E+06 5.52473E+06 9.09001E+06 1.58075E+07 1.97704E+07
 1.83115E+07 1.54698E+07 1.40926E+07 2.44114E+07 2.20573E+07 2.03030E+07
 2.45409E+07 2.71187E+07 2.81788E+07 1.96223E+07 2.87708E+07 3.40429E+07
 3.36810E+07 4.18245E+07 1.35009E+07 2.75173E+07 3.38958E+07 3.92836E+07
 4.72407E+07 5.23836E+07 5.78279E+07 6.47017E+07 7.52565E+07 8.56770E+07
 8.65734E+07 7.86350E+07 8.66430E+07 8.21358E+07 1.27787E+08 7.59820E+07
 4.44202E+07 4.66708E+07 1.23264E+08 1.48330E+08 1.72910E+07 6.47230E+06
 1.33075E+07 3.58180E+07 7.44322E+07 1.45110E+08 7.15691E+07 7.10896E+07
 6.39294E+07 6.25478E+07 5.91705E+07 5.66331E+07 6.52191E+07 8.05716E+07
 9.11865E+07 8.83925E+07 8.09986E+07 8.78136E+07 8.06470E+07 9.12996E+07
 7.96552E+07 8.08025E+07 1.90551E+08 1.65740E+08 6.29882E+07 5.50126E+07
 1.41789E+08 9.27476E+07 2.17575E+08 8.35274E+07 1.78699E+08 1.56769E+08
 1.98625E+08 9.15462E+07 1.36427E+08 6.35769E+07 4.39068E+07 6.21433E+07
 3.16168E+07 3.12031E+07 9.25966E+07 1.51759E+08 2.98772E+08 2.95250E+08
 1.17077E+08 1.72623E+08 2.89718E+08 2.85216E+08 2.88335E+08 1.69889E+08
 1.09253E+08 9.73156E+07 7.26310E+07 4.42706E+07 5.81257E+07 1.30973E+08
 1.25942E+08 3.05032E+08 2.95894E+08 2.92202E+08 2.91318E+08 2.91099E+08
 2.91042E+08 2.90483E+08 2.90458E+08 2.90311E+08 2.90033E+08 2.89790E+08
 2.89306E+08 2.88807E+08 2.88223E+08 2.87556E+08 2.86812E+08 2.85956E+08
 2.85047E+08 2.83978E+08 2.82920E+08 2.81750E+08 2.80389E+08 2.78955E+08
 2.77410E+08 2.81952E+08 2.79388E+08 2.80426E+08 2.81617E+08 2.83382E+08
 1.20518E+08 1.65361E+08 4.72695E+07 4.35180E+07 4.53019E+07 1.53165E+08
 1.06791E+08 1.87862E+08 1.05378E+08 1.96696E+08 7.52576E+07 2.34861E+08
 1.52838E+08 1.55474E+08 2.22645E+08 2.93228E+08 3.54638E+08 4.76902E+08
 5.76977E+08 9.69327E+08 2.17377E+09 2.41832E+09 1.54320E+09 1.71197E+09
```

TAB. A.10 Continued

Processing Job for Collapsing All VITJEFF311.BOLIB Nuclides with Concrete
Weighting Spectrum without Upscattering Cross Sections.
(Table corresponding to TAB. A.14 in reference /8/)

```
  1.57635E+09 1.04794E+09 6.10169E+08 4.89143E+08 1.57783E+08 3.03672E+07
  2.07969E+06
6$$ 2z 2r1 2 2r3 2r4 2r5 2r6 2r7 2r8 2r9 3r10 11 12 13 4r14 2r15
      2r16 3r17 3r18 19 20 0
7**
' photon weighting flux for concrete (xsdrnpm pwr calc. int #106)
 3.15691E+01 9.79614E+02 4.59269E+03 1.43771E+06 9.44209E+07 3.41969E+08
 1.11170E+08 6.48519E+07 1.73983E+08 1.86948E+08 1.73882E+08 3.75841E+08
 1.80878E+08 4.73675E+08 2.70875E+08 3.57831E+08 8.26411E+08 4.36238E+08
 2.03888E+08 2.01998E+08 1.28382E+07 5.28507E+08 4.35567E+08 2.53035E+08
 2.82375E+08 3.10777E+08 2.86509E+08 3.43181E+08 3.12050E+08 8.32199E+08
 1.47642E+09 1.21393E+09 1.84323E+09 1.24205E+09 2.25526E+08 3.43201E+08
 2.11349E+08 1.82409E+07 5.75789E+06 2.79798E+05 3.75042E+04 2.67899E+03
t
end
=nitawl
6$$ a3 500 1000 e
0$$ 81 e
1$$ 0 183 a11 -1 e t
2$$ 47107 47109
  13027
  95241 95242 95601 95243
  79197
  5010 5011
  56138
  4009
  4309
  83209
  6012
  6312
  20040 20042 20043 20044 20046 20048
  48106 48108 48110 48111 48112 48113 48114 48115 48116
  17035 17037
  96241 96242 96243 96244 96245 96246 96247 96248
  27059
  24050 24052 24053 24054
  29063 29065
  68162 68164 68166 68167 68168 68170
  63151 63152 63153 63154 63155
  9019
  26054 26056 26057 26058
  31069 31071
  64152 64154 64155 64156 64157 64158 64160
  1001 1901 1401 1002 1003
  2003 2004
  72174 72176 72177 72178 72179 72180
  49113 49115
  19039 19040 19041
  3006 3007
  12024 12025 12026
```

TAB. A.10 Continued

Processing Job for Collapsing All VITJEFF311.BOLIB Nuclides with Concrete
Weighting Spectrum without Upscattering Cross Sections.
(Table corresponding to TAB. A.14 in reference /8/)

```
    25055
    42092 42094 42095 42096 42097 42098 42100
    7014 7015
    11023
    41093
    28058 28060 28061 28062 28064
    93237 93238 93239
    8016 8017
    15031
    91231 91233
    82204 82206 82207 82208
    94236 94237 94238 94239 94240 94241 94242 94243 94244
    75185 75187
    16032 16033 16034 16036
    14028 14029 14030
    50112 50114 50115 50116 50117 50118 50119
    50120 50122 50123 50124 50125 50126
    73181 73182
    90230 90232
    22046 22047 22048 22049 22050
    92232 92233 92234 92235 92236 92237 92238
    23000
    74182 74183 74184 74186
    39089
    40090 40091 40092 40094 40096
t
end
=alpo
0$$ 87 4
1$$ 1 3 4 70  7 -1 -1 e t
2$$ 4 0 t
end
=rade
1$$ 0 4 e
2$$ a2 100 1 e t
end
=shell
mv ft87f001 /home/BUGLE/bugle.b70
rm ft*  2>/dev/null
end
```

TAB. A.11

Processing Job for Collapsing PWR Concrete Nuclides with Upscattering Cross Sections.
(Table corresponding to TAB. A.15 in reference /8/)

```
#!/bin/csh
# shell script to execute a scampi job
=shell
rm ft*  2>/dev/null
ln -sf /home/BUGLE/A05.bonami.b70 ft85f001
end
=ajax
0$$ 82 85 e
1$$ 1 t
2$$ 85 24 t
3$$ 100141 801641
 2605441 2605641 2605741 2605841
 601241  1102341
 1202441 1202541 1202641
 1302741
 1402841 1402941 1403041
 1903941 1904041 1904141
 2004041 2004241 2004341 2004441 2004641 2004841 t
end
=malocs
1$$ 199 47 42 20 e
2$$ 82 81
3$$ 0 0 1 1 a7 0 f0 t
4$$
 0 6r1 5r2 4r3 3r4 3r5 5r6 4r7 4r8 3r9 2r10
 2r11 2r12 13 2r14 3r15 3r16 4r17 5r18 4r19 2r20
 4r21 4r22 4r23 4r24 10r25 10r26 6r27 4r28 2r29 3r30
 2r31 2r32 2r33 4r34 4r35 7r36 5r37 3r38 3r39 4r40
 5r41 3r42 4r43 7r44 6r45 8r46 11r47
5**
' neutron weighting flux for concrete (xsdrnpm pwr calc. int #106)
 7.29347E+01 1.15858E+04 2.16810E+04 6.38868E+04 8.44980E+04 5.32721E+04
 6.64980E+04 8.24386E+04 9.72686E+04 2.44419E+05 1.66633E+05 1.85270E+05
 3.81228E+05 4.64720E+05 8.44247E+05 9.02182E+05 1.19494E+06 1.62534E+06
 1.79656E+06 1.85923E+06 2.30233E+06 2.58884E+06 2.77701E+06 3.59210E+06
 1.38828E+06 3.12911E+06 5.10439E+06 4.66159E+06 4.63761E+06 6.21271E+06
 5.86660E+06 6.52650E+06 6.40871E+06 1.36120E+07 1.19272E+07 1.13266E+07
 7.02453E+06 9.44667E+06 1.06145E+07 1.23500E+07 1.49920E+07 1.50563E+07
 1.37684E+07 4.20412E+06 5.52473E+06 9.09001E+06 1.58075E+07 1.97704E+07
 1.83115E+07 1.54698E+07 1.40926E+07 2.44114E+07 2.20573E+07 2.03030E+07
 2.45409E+07 2.71187E+07 2.81788E+07 1.96223E+07 2.87708E+07 3.40429E+07
 3.36810E+07 4.18245E+07 1.35009E+07 2.75173E+07 3.38958E+07 3.92836E+07
 4.72407E+07 5.23836E+07 5.78279E+07 6.47017E+07 7.52565E+07 8.56770E+07
 8.65734E+07 7.86350E+07 8.66430E+07 8.21358E+07 1.27787E+08 7.59820E+07
 4.44202E+07 4.66708E+07 1.23264E+08 1.48330E+08 1.72910E+07 6.47230E+06
 1.33075E+07 3.58180E+07 7.44322E+07 1.45110E+08 7.15691E+07 7.10896E+07
 6.39294E+07 6.25478E+07 5.91705E+07 5.66331E+07 6.52191E+07 8.05716E+07
 9.11865E+07 8.83925E+07 8.09986E+07 8.78136E+07 8.06470E+07 9.12996E+07
 7.96552E+07 8.08025E+07 1.90551E+08 1.65740E+08 6.29882E+07 5.50126E+07
 1.41789E+08 9.27476E+07 2.17575E+08 8.35274E+07 1.78699E+08 1.56769E+08
 1.98625E+08 9.15462E+07 1.36427E+08 6.35769E+07 4.39068E+07 6.21433E+07
```

TAB. A.11 Continued

Processing Job for Collapsing PWR Concrete Nuclides with Upscattering Cross Sections.
(Table corresponding to TAB. A.15 in reference /8/)

```
   3.16168E+07 3.12031E+07 9.25966E+07 1.51759E+08 2.98772E+08 2.95250E+08
   1.17077E+08 1.72623E+08 2.89718E+08 2.85216E+08 2.88335E+08 1.69889E+08
   1.09253E+08 9.73156E+07 7.26310E+07 4.42706E+07 5.81257E+07 1.30973E+08
   1.25942E+08 3.05032E+08 2.95894E+08 2.92202E+08 2.91318E+08 2.91099E+08
   2.91042E+08 2.90483E+08 2.90458E+08 2.90311E+08 2.90033E+08 2.89790E+08
   2.89306E+08 2.88807E+08 2.88223E+08 2.87556E+08 2.86812E+08 2.85956E+08
   2.85047E+08 2.83978E+08 2.82920E+08 2.81750E+08 2.80389E+08 2.78955E+08
   2.77410E+08 2.81952E+08 2.79388E+08 2.80426E+08 2.81617E+08 2.83382E+08
   1.20518E+08 1.65361E+08 4.72695E+07 4.35180E+07 4.53019E+07 1.53165E+08
   1.06791E+08 1.87862E+08 1.05378E+08 1.96696E+08 7.52576E+07 2.34861E+08
   1.52838E+08 1.55474E+08 2.22645E+08 2.93228E+08 3.54638E+08 4.76902E+08
   5.76977E+08 9.69327E+08 2.17377E+09 2.41832E+09 1.54320E+09 1.71197E+09
   1.57635E+09 1.04794E+09 6.10169E+08 4.89143E+08 1.57783E+08 3.03672E+07
   2.07969E+06
6$$ 2z 2r1 2 2r3 2r4 2r5 2r6 2r7 2r8 2r9 3r10 11 12 13 4r14 2r15
      2r16 3r17 3r18 19 20 0
7**
' photon weighting flux for concrete (xsdrnpm pwr calc. int #106)
   3.15691E+01 9.79614E+02 4.59269E+03 1.43771E+06 9.44209E+07 3.41969E+08
   1.11170E+08 6.48519E+07 1.73983E+08 1.86948E+08 1.73882E+08 3.75841E+08
   1.80878E+08 4.73675E+08 2.70875E+08 3.57831E+08 8.26411E+08 4.36238E+08
   2.03888E+08 2.01998E+08 1.28382E+07 5.28507E+08 4.35567E+08 2.53035E+08
   2.82375E+08 3.10777E+08 2.86509E+08 3.43181E+08 3.12050E+08 8.32199E+08
   1.47642E+09 1.21393E+09 1.84323E+09 1.24205E+09 2.25526E+08 3.43201E+08
   2.11349E+08 1.82409E+07 5.75789E+06 2.79798E+05 3.75042E+04 2.67899E+03
t
end
=nitawl
6$$ a3 500 1000 e
0$$ 81 e
1$$ 0  24 a11 -1 e t
2$$ 100141 801641
 2605441 2605641 2605741 2605841
 601241 1102341
 1202441 1202541 1202641
 1302741
 1402841 1402941 1403041
 1903941 1904041 1904141
 2004041 2004241 2004341 2004441 2004641 2004841 t
end
=rade
1$$ 0 4 e
2$$ a2 100 1 e t
end
=alpo
0$$ 87 4
1$$ 1 3 8 74  7 -1 -1 e t
2$$ 4 0 t
end
=shell
mv ft87f001 /home/BUGLE/bugle.b70.pconc
```

TAB. A.11 Continued

Processing Job for Collapsing PWR Concrete Nuclides with Upscattering Cross Sections.
(Table corresponding to TAB. A.15 in reference /8/)

```
rm ft*  2>/dev/null
end
```

APPENDIX B

General Neutron Response Functions

TAB. B.1

General Neutron Response Functions Included with BUGENDF70.BOLIB.

Grp	Upper Energy [MeV]	Energy Width [MeV]	Lethargy Width	Midpoint Energy (E-mid) [MeV]	Square-Root E-mid [MeV$^{1/2}$]
1	1.7332E+01	3.1410E+00	1.9995E-01	1.5762E+01	3.9701E+00
2	1.4191E+01	1.9770E+00	1.5002E-01	1.3203E+01	3.6335E+00
3	1.2214E+01	2.2140E+00	2.0000E-01	1.1107E+01	3.3327E+00
4	1.0000E+01	1.3929E+00	1.5000E-01	9.3035E+00	3.0502E+00
5	8.6071E+00	1.1989E+00	1.5000E-01	8.0077E+00	2.8298E+00
6	7.4082E+00	1.3429E+00	2.0000E-01	6.7368E+00	2.5955E+00
7	6.0653E+00	1.0994E+00	1.9999E-01	5.5156E+00	2.3485E+00
8	4.9659E+00	1.2871E+00	3.0001E-01	4.3224E+00	2.0790E+00
9	3.6788E+00	6.6690E-01	2.0002E-01	3.3454E+00	1.8290E+00
10	3.0119E+00	2.8660E-01	9.9993E-02	2.8686E+00	1.6937E+00
11	2.7253E+00	2.5930E-01	9.9981E-02	2.5956E+00	1.6111E+00
12	2.4660E+00	1.0070E-01	4.1693E-02	2.4156E+00	1.5542E+00
13	2.3653E+00	1.9600E-02	8.3210E-03	2.3555E+00	1.5348E+00
14	2.3457E+00	1.1440E-01	4.9999E-02	2.2885E+00	1.5128E+00
15	2.2313E+00	3.1080E-01	1.5000E-01	2.0759E+00	1.4408E+00
16	1.9205E+00	2.6750E-01	1.4999E-01	1.7867E+00	1.3367E+00
17	1.6530E+00	2.9960E-01	1.9997E-01	1.5032E+00	1.2261E+00
18	1.3534E+00	3.5080E-01	3.0002E-01	1.1780E+00	1.0854E+00
19	1.0026E+00	1.8175E-01	2.0001E-01	9.1172E-01	9.5484E-01
20	8.2085E-01	7.8110E-02	9.9994E-02	7.8180E-01	8.8419E-01
21	7.4274E-01	1.3464E-01	2.0001E-01	6.7542E-01	8.2184E-01
22	6.0810E-01	1.1023E-01	2.0000E-01	5.5299E-01	7.4363E-01
23	4.9787E-01	1.2904E-01	3.0000E-01	4.3335E-01	6.5829E-01
24	3.6883E-01	7.1620E-02	2.1590E-01	3.3302E-01	5.7708E-01
25	2.9721E-01	1.1405E-01	4.8408E-01	2.4018E-01	4.9009E-01
26	1.8316E-01	7.2070E-02	5.0002E-01	1.4713E-01	3.8357E-01
27	1.1109E-01	4.3711E-02	5.0001E-01	8.9235E-02	2.9872E-01
28	6.7379E-02	2.6511E-02	4.9999E-01	5.4123E-02	2.3265E-01
29	4.0868E-02	9.0400E-03	2.5000E-01	3.6348E-02	1.9065E-01
30	3.1828E-02	5.7700E-03	2.0002E-01	2.8943E-02	1.7013E-01
31	2.6058E-02	1.8820E-03	7.4964E-02	2.5117E-02	1.5848E-01
32	2.4176E-02	2.3010E-03	1.0002E-01	2.3025E-02	1.5174E-01
33	2.1875E-02	6.8410E-03	3.7503E-01	1.8454E-02	1.3585E-01
34	1.5034E-02	7.9323E-03	7.4998E-01	1.1068E-02	1.0520E-01
35	7.1017E-03	3.7471E-03	7.5000E-01	5.2282E-03	7.2306E-02
36	3.3546E-03	1.7700E-03	7.5000E-01	2.4696E-03	4.9695E-02
37	1.5846E-03	1.1306E-03	1.2500E+00	1.0193E-03	3.1927E-02
38	4.5400E-04	2.3955E-04	7.5002E-01	3.3423E-04	1.8282E-02
39	2.1445E-04	1.1315E-04	7.4999E-01	1.5787E-04	1.2565E-02
40	1.0130E-04	6.4034E-05	1.0000E+00	6.9283E-05	8.3236E-03
41	3.7266E-05	2.6589E-05	1.2500E+00	2.3971E-05	4.8961E-03
42	1.0677E-05	5.6335E-06	7.4999E-01	7.8602E-06	2.8036E-03
43	5.0435E-06	3.1881E-06	1.0000E+00	3.4495E-06	1.8573E-03
44	1.8554E-06	9.7897E-07	7.5000E-01	1.3659E-06	1.1687E-03
45	8.7643E-07	4.6244E-07	7.5002E-01	6.4521E-07	8.0325E-04
46	4.1399E-07	3.1399E-07	1.4207E+00	2.5700E-07	5.0695E-04
47	1.0000E-07	9.9990E-08	9.2103E+00	5.0005E-08	2.2362E-04
	1.0000E-11				

TAB. B.1 Continued

General Neutron Response Functions Included with BUGENDF70.BOLIB.

Grp	Total Neutron Flux	E > 1.0 MeV Neutron Flux	E > 0.1 MeV Neutron Flux	E < 0.414 eV Neutron Flux
1	1.	1.	1.	0.
2	1.	1.	1.	0.
3	1.	1.	1.	0.
4	1.	1.	1.	0.
5	1.	1.	1.	0.
6	1.	1.	1.	0.
7	1.	1.	1.	0.
8	1.	1.	1.	0.
9	1.	1.	1.	0.
10	1.	1.	1.	0.
11	1.	1.	1.	0.
12	1.	1.	1.	0.
13	1.	1.	1.	0.
14	1.	1.	1.	0.
15	1.	1.	1.	0.
16	1.	1.	1.	0.
17	1.	1.	1.	0.
18	1.	1.	1.	0.
19	1.	$1.3000E-02^a$	1.	0.
20	1.	0.	1.	0.
21	1.	0.	1.	0.
22	1.	0.	1.	0.
23	1.	0.	1.	0.
24	1.	0.	1.	0.
25	1.	0.	1.	0.
26	1.	0.	1.	0.
27	1.	0.	$2.1034E-01^b$	0.
28	1.	0.	0.	0.
29	1.	0.	0.	0.
30	1.	0.	0.	0.
31	1.	0.	0.	0.
32	1.	0.	0.	0.
33	1.	0.	0.	0.
34	1.	0.	0.	0.
35	1.	0.	0.	0.
36	1.	0.	0.	0.
37	1.	0.	0.	0.
38	1.	0.	0.	0.
39	1.	0.	0.	0.
40	1.	0.	0.	0.
41	1.	0.	0.	0.
42	1.	0.	0.	0.
43	1.	0.	0.	0.
44	1.	0.	0.	0.
45	1.	0.	0.	0.
46	1.	0.	0.	1.
47	1.	0.	0.	1.

TAB. B.1 Continued

General Neutron Response Functions Included with BUGENDF70.BOLIB.

[a] $1.3000E-02 = \ln(E_0/E_g) / \ln(E_{g+1}/E_g)$ **where:**

$E_0 = 1.0$ **[MeV];**
$E_g = 1.0026$ **[MeV] (Upper energy limit Grp No. 19);**
$E_{g+1} = 0.82085$ **[MeV] (Lower energy limit Grp No. 19).**

[b] $2.1034E-01 = \ln(E_0/E_g) / \ln(E_{g+1}/E_g)$ **where:**

$E_0 = 0.1$ **[MeV];**
$E_g = 1.1109E-01$ **[MeV] (Upper energy limit Grp No. 27);**
$E_{g+1} = 6.7379E-02$ **[MeV] (Lower energy limit Grp No. 27).**

APPENDIX C

Total (Prompt + Delayed) Neutron Fission Spectra (χ)

TAB. C.1

Response Functions Included with BUGENDF70.BOLIB.
Total or Prompt (∗) Neutron Fission Spectra (χ).

Grp	Th-230*	Th-232	Pa-231	Pa-233	U-232*	U-233
1	8.0233E-05	3.2372E-05	6.2657E-05	5.2863E-05	6.5454E-05	4.4420E-05
2	2.8001E-04	1.3920E-04	2.3949E-04	2.0685E-04	2.3640E-04	1.9128E-04
3	1.4154E-03	8.3986E-04	1.2957E-03	1.1491E-03	1.2350E-03	1.1531E-03
4	2.9767E-03	1.9874E-03	2.8307E-03	2.5640E-03	2.6635E-03	2.7162E-03
5	6.2303E-03	4.5122E-03	6.0555E-03	5.5747E-03	5.6844E-03	6.1119E-03
6	1.6736E-02	1.3073E-02	1.6557E-02	1.5485E-02	1.5576E-02	1.7381E-02
7	3.0671E-02	2.5575E-02	3.0717E-02	2.9204E-02	2.9073E-02	3.2988E-02
8	7.8337E-02	6.9707E-02	7.9177E-02	7.6418E-02	7.5658E-02	8.5394E-02
9	7.2968E-02	6.8061E-02	7.4293E-02	7.2478E-02	7.1469E-02	7.9009E-02
10	4.1352E-02	3.9424E-02	4.2106E-02	4.1363E-02	4.0789E-02	4.4307E-02
11	4.3684E-02	4.2367E-02	4.4744E-02	4.4187E-02	4.3270E-02	4.6388E-02
12	1.8727E-02	1.8443E-02	1.9296E-02	1.9084E-02	1.8600E-02	1.9750E-02
13	3.7655E-03	3.7267E-03	3.8816E-03	3.8416E-03	3.7435E-03	3.9611E-03
14	2.2785E-02	2.2629E-02	2.3475E-02	2.3245E-02	2.2675E-02	2.3896E-02
15	6.9209E-02	6.9170E-02	7.0875E-02	7.0316E-02	6.9105E-02	7.1813E-02
16	6.8657E-02	6.8896E-02	6.9369E-02	6.9324E-02	6.8859E-02	7.0082E-02
17	8.7258E-02	8.7952E-02	8.7009E-02	8.7110E-02	8.7898E-02	8.7409E-02
18	1.1537E-01	1.1781E-01	1.1439E-01	1.1548E-01	1.1680E-01	1.1270E-01
19	6.4275E-02	6.6974E-02	6.3551E-02	6.4531E-02	6.5336E-02	6.1380E-02
20	2.8220E-02	2.9587E-02	2.7657E-02	2.8339E-02	2.8744E-02	2.6630E-02
21	4.8948E-02	5.1815E-02	4.7919E-02	4.8978E-02	4.9937E-02	4.5719E-02
22	3.9757E-02	4.2884E-02	3.9185E-02	4.0073E-02	4.0637E-02	3.6705E-02
23	4.5036E-02	4.9796E-02	4.4658E-02	4.5852E-02	4.6118E-02	4.1081E-02
24	2.3651E-02	2.6404E-02	2.3394E-02	2.4214E-02	2.4256E-02	2.1329E-02
25	3.4196E-02	3.8804E-02	3.3916E-02	3.5666E-02	3.5122E-02	3.0556E-02
26	1.8143E-02	2.0920E-02	1.7978E-02	1.8920E-02	1.8661E-02	1.6076E-02
27	8.9543E-03	1.0253E-02	8.7294E-03	9.1741E-03	9.2180E-03	7.8871E-03
28	4.3436E-03	4.7859E-03	4.0187E-03	4.2840E-03	4.4741E-03	3.8164E-03
29	1.2328E-03	1.2255E-03	9.9578E-04	1.0661E-03	1.2702E-03	1.0819E-03
30	7.0626E-04	6.5300E-04	5.1500E-04	5.5981E-04	7.2775E-04	6.2329E-04
31	2.1530E-04	1.9092E-04	1.4752E-04	1.6213E-04	2.2186E-04	1.9070E-04
32	2.5241E-04	2.1792E-04	1.6634E-04	1.8380E-04	2.6012E-04	2.2386E-04
33	6.7318E-04	5.5521E-04	4.0505E-04	4.5615E-04	6.9378E-04	5.9938E-04
34	6.0531E-04	4.2868E-04	2.9188E-04	3.3858E-04	6.2390E-04	5.4009E-04
35	1.9740E-04	1.0752E-04	6.7754E-05	8.1050E-05	2.0348E-04	1.7549E-04
36	6.4237E-05	3.5056E-05	1.7774E-05	2.3648E-05	6.6218E-05	5.8167E-05
37	2.6207E-05	1.7109E-05	6.5739E-06	1.0190E-05	2.7015E-05	2.4506E-05
38	3.2262E-06	3.0961E-06	9.1454E-07	1.6671E-06	3.3252E-06	3.1984E-06
39	1.0594E-06	1.3981E-06	3.7382E-07	7.2765E-07	1.0917E-06	1.1074E-06
40	4.0778E-07	7.7294E-07	1.9502E-07	3.9479E-07	4.1993E-07	4.5758E-07
41	1.0987E-07	3.1707E-07	7.7465E-08	1.6032E-07	1.1292E-07	1.3354E-07
42	1.7436E-08	6.6885E-08	1.6147E-08	3.3695E-08	1.7848E-08	2.1101E-08
43	9.5458E-09	3.7805E-08	9.0961E-09	1.9026E-08	9.7296E-09	1.0050E-08
44	3.4068E-09	1.1599E-08	2.7867E-09	5.8343E-09	3.4580E-09	2.6712E-09
45	2.0564E-09	5.4788E-09	1.3150E-09	2.7550E-09	2.0825E-09	1.1567E-09
46	2.1275E-09	3.7200E-09	8.9287E-10	1.8706E-09	2.1513E-09	7.2635E-10
47	1.9784E-09	1.1848E-09	2.8436E-10	5.9576E-10	1.9988E-09	2.1334E-10

TAB. C.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
Total or Prompt (∗) Neutron Fission Spectra (χ).

Grp	U-234	U-235	U-236	U-237	U-238	Np-237
1	5.1168E-05	4.9839E-05	1.0533E-04	6.6717E-05	4.3897E-05	3.8862E-05
2	2.0383E-04	2.0031E-04	3.4002E-04	2.3920E-04	1.7673E-04	1.6707E-04
3	1.1453E-03	1.1384E-03	1.6131E-03	1.2418E-03	1.0144E-03	1.0137E-03
4	2.5749E-03	2.5055E-03	3.2549E-03	2.6665E-03	2.3395E-03	2.4193E-03
5	5.6281E-03	5.6195E-03	6.6283E-03	5.6729E-03	5.2694E-03	5.5266E-03
6	1.5699E-02	1.6041E-02	1.7393E-02	1.5498E-02	1.5147E-02	1.6061E-02
7	2.9640E-02	3.0511E-02	3.1331E-02	2.8852E-02	2.9046E-02	3.1206E-02
8	7.7645E-02	7.9986E-02	7.9066E-02	7.4897E-02	7.6783E-02	8.3294E-02
9	7.3401E-02	7.6258E-02	7.3101E-02	7.0628E-02	7.3724E-02	7.8954E-02
10	4.1816E-02	4.3728E-02	4.1257E-02	4.0289E-02	4.2498E-02	4.4803E-02
11	4.4263E-02	4.6388E-02	4.3472E-02	4.2725E-02	4.5261E-02	4.7174E-02
12	1.8992E-02	1.9903E-02	1.8606E-02	1.8364E-02	1.9478E-02	2.0152E-02
13	3.8200E-03	4.0013E-03	3.7395E-03	3.6961E-03	3.9203E-03	4.0283E-03
14	2.3117E-02	2.4193E-02	2.2613E-02	2.2386E-02	2.3733E-02	2.4452E-02
15	7.0219E-02	7.3105E-02	6.8564E-02	6.8224E-02	7.2044E-02	7.3584E-02
16	6.9577E-02	7.1538E-02	6.7876E-02	6.8002E-02	7.0985E-02	7.1817E-02
17	8.8208E-02	8.9083E-02	8.6194E-02	8.6957E-02	8.9059E-02	8.9274E-02
18	1.1607E-01	1.1474E-01	1.1402E-01	1.1600E-01	1.1576E-01	1.1394E-01
19	6.4366E-02	6.2734E-02	6.3688E-02	6.5277E-02	6.3849E-02	6.1491E-02
20	2.8191E-02	2.7329E-02	2.8024E-02	2.8837E-02	2.7963E-02	2.6556E-02
21	4.8798E-02	4.7130E-02	4.8743E-02	5.0323E-02	4.8492E-02	4.5370E-02
22	3.9587E-02	3.8038E-02	3.9859E-02	4.1408E-02	3.9565E-02	3.6253E-02
23	4.4758E-02	4.2436E-02	4.5393E-02	4.7405E-02	4.4575E-02	4.0440E-02
24	2.3433E-02	2.1807E-02	2.3897E-02	2.5057E-02	2.3099E-02	2.1080E-02
25	3.3797E-02	3.0779E-02	3.4707E-02	3.6520E-02	3.2849E-02	2.9995E-02
26	1.7863E-02	1.5922E-02	1.8583E-02	1.9658E-02	1.7171E-02	1.5815E-02
27	8.8206E-03	7.7147E-03	9.2267E-03	9.7959E-03	8.3623E-03	7.8949E-03
28	4.2688E-03	3.6943E-03	4.4795E-03	4.7562E-03	3.9848E-03	3.7430E-03
29	1.2109E-03	1.0459E-03	1.2800E-03	1.3664E-03	1.1373E-03	1.0545E-03
30	6.9790E-04	6.0374E-04	7.4192E-04	7.9488E-04	6.6181E-04	6.0026E-04
31	2.1365E-04	1.8501E-04	2.2787E-04	2.4472E-04	2.0392E-04	1.8598E-04
32	2.5062E-04	2.1735E-04	2.6805E-04	2.8826E-04	2.4027E-04	2.2739E-04
33	6.7202E-04	5.8781E-04	7.2638E-04	7.8673E-04	6.6097E-04	5.7219E-04
34	6.0488E-04	5.3054E-04	6.5895E-04	7.1710E-04	6.0227E-04	5.6001E-04
35	1.9640E-04	1.7033E-04	2.1306E-04	2.3173E-04	1.9224E-04	1.6944E-04
36	6.5387E-05	5.7386E-05	7.2355E-05	8.0077E-05	6.7486E-05	5.8235E-05
37	2.7818E-05	2.4884E-05	3.1762E-05	3.6131E-05	3.1223E-05	2.4800E-05
38	3.7102E-06	3.4200E-06	4.4612E-06	5.2982E-06	4.7458E-06	3.1926E-06
39	1.3091E-06	1.2414E-06	1.6544E-06	2.0318E-06	1.8633E-06	1.1084E-06
40	5.5521E-07	5.4440E-07	7.4937E-07	9.5543E-07	8.9469E-07	4.4846E-07
41	1.6810E-07	1.7323E-07	2.5326E-07	3.3636E-07	3.1829E-07	1.3160E-07
42	2.7748E-08	2.9006E-08	4.7990E-08	6.5298E-08	5.9716E-08	2.0779E-08
43	1.3629E-08	1.3842E-08	2.6865E-08	3.6593E-08	3.1220E-08	9.9619E-09
44	3.7283E-09	3.8754E-09	8.7229E-09	1.1701E-08	9.2114E-09	2.6093E-09
45	1.6466E-09	1.7694E-09	4.5624E-09	5.9697E-09	4.2897E-09	1.0962E-09
46	5.3584E-05	1.1790E-09	3.8192E-09	4.7796E-09	2.8893E-09	7.4334E-10
47	5.3583E-05	3.7167E-10	2.4980E-09	2.8161E-09	9.1590E-10	2.3626E-10

TAB. C.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
Total or Prompt (∗) Neutron Fission Spectra (χ).

Grp	Np-238*	Np-239*	Pu-236*	Pu-237*	Pu-238	Pu-239
1	9.0289E-05	7.1628E-05	9.2392E-05	8.0233E-05	8.0118E-05	7.9158E-05
2	3.0886E-04	2.5482E-04	3.1481E-04	2.8001E-04	2.7961E-04	2.9392E-04
3	1.5315E-03	1.3120E-03	1.5551E-03	1.4154E-03	1.4133E-03	1.5465E-03
4	3.1742E-03	2.7983E-03	3.2140E-03	2.9767E-03	2.9725E-03	3.2119E-03
5	6.5688E-03	5.9207E-03	6.6364E-03	6.2303E-03	6.2213E-03	6.8541E-03
6	1.7443E-02	1.6082E-02	1.7583E-02	1.6736E-02	1.6712E-02	1.8658E-02
7	3.1630E-02	2.9773E-02	3.1818E-02	3.0671E-02	3.0627E-02	3.4131E-02
8	7.9914E-02	7.6839E-02	8.0221E-02	7.8337E-02	7.8225E-02	8.5979E-02
9	7.3832E-02	7.2135E-02	7.3998E-02	7.2968E-02	7.2863E-02	7.9318E-02
10	4.1670E-02	4.1040E-02	4.1731E-02	4.1352E-02	4.1293E-02	4.4717E-02
11	4.3913E-02	4.3456E-02	4.3956E-02	4.3684E-02	4.3622E-02	4.6993E-02
12	1.8795E-02	1.8658E-02	1.8808E-02	1.8727E-02	1.8701E-02	2.0047E-02
13	3.7772E-03	3.7535E-03	3.7794E-03	3.7655E-03	3.7603E-03	4.0228E-03
14	2.2842E-02	2.2725E-02	2.2853E-02	2.2785E-02	2.2753E-02	2.4276E-02
15	6.9249E-02	6.9156E-02	6.9255E-02	6.9209E-02	6.9115E-02	7.2934E-02
16	6.8520E-02	6.8774E-02	6.8492E-02	6.8657E-02	6.8567E-02	7.0893E-02
17	8.6864E-02	8.7621E-02	8.6785E-02	8.7258E-02	8.7161E-02	8.7701E-02
18	1.1451E-01	1.1617E-01	1.1435E-01	1.1537E-01	1.1529E-01	1.1198E-01
19	6.3648E-02	6.4868E-02	6.3526E-02	6.4275E-02	6.4283E-02	6.0697E-02
20	2.7913E-02	2.8513E-02	2.7853E-02	2.8220E-02	2.8239E-02	2.6301E-02
21	4.8369E-02	4.9500E-02	4.8256E-02	4.8948E-02	4.9000E-02	4.5125E-02
22	3.9244E-02	4.0248E-02	3.9144E-02	3.9757E-02	3.9850E-02	3.6164E-02
23	4.4408E-02	4.5639E-02	4.4286E-02	4.5036E-02	4.5191E-02	4.0098E-02
24	2.3299E-02	2.3988E-02	2.3232E-02	2.3651E-02	2.3746E-02	2.0523E-02
25	3.3661E-02	3.4711E-02	3.3558E-02	3.4196E-02	3.4359E-02	2.8868E-02
26	1.7844E-02	1.8431E-02	1.7787E-02	1.8143E-02	1.8253E-02	1.4852E-02
27	8.8022E-03	9.1009E-03	8.7728E-03	8.9543E-03	9.0174E-03	7.1771E-03
28	4.2685E-03	4.4161E-03	4.2540E-03	4.3437E-03	4.3745E-03	3.4301E-03
29	1.2113E-03	1.2536E-03	1.2071E-03	1.2328E-03	1.2427E-03	9.6819E-04
30	6.9388E-04	7.1820E-04	6.9150E-04	7.0626E-04	7.1336E-04	5.5497E-04
31	2.1152E-04	2.1894E-04	2.1079E-04	2.1530E-04	2.1774E-04	1.6930E-04
32	2.4798E-04	2.5669E-04	2.4712E-04	2.5241E-04	2.5542E-04	1.9853E-04
33	6.6132E-04	6.8462E-04	6.5904E-04	6.7318E-04	6.8326E-04	5.3068E-04
34	5.9461E-04	6.1564E-04	5.9255E-04	6.0531E-04	6.1489E-04	4.7675E-04
35	1.9390E-04	2.0078E-04	1.9323E-04	1.9740E-04	1.9975E-04	1.5458E-04
36	6.3097E-05	6.5337E-05	6.2878E-05	6.4237E-05	6.5389E-05	5.0921E-05
37	2.5742E-05	2.6656E-05	2.5653E-05	2.6207E-05	2.6964E-05	2.1251E-05
38	3.1693E-06	3.2812E-06	3.1583E-06	3.2262E-06	3.3900E-06	2.7285E-06
39	1.0409E-06	1.0773E-06	1.0373E-06	1.0594E-06	1.1374E-06	9.3057E-07
40	4.0080E-07	4.1453E-07	3.9945E-07	4.0779E-07	4.5221E-07	3.7515E-07
41	1.0811E-07	1.1157E-07	1.0777E-07	1.0987E-07	1.2840E-07	1.0518E-07
42	1.7199E-08	1.7665E-08	1.7153E-08	1.7436E-08	2.1370E-08	1.4912E-08
43	9.4396E-09	9.6481E-09	9.4191E-09	9.5459E-09	1.1773E-08	5.9741E-09
44	3.3771E-09	3.4353E-09	3.3714E-09	3.4068E-09	4.0900E-09	1.4753E-09
45	2.0411E-09	2.0709E-09	2.0382E-09	2.0564E-09	2.3784E-09	6.3819E-10
46	2.1136E-09	2.1408E-09	2.1109E-09	2.1275E-09	2.3451E-09	4.1186E-10
47	1.9665E-09	1.9898E-09	1.9642E-09	1.9784E-09	2.0459E-09	1.2752E-10

TAB. C.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
Total or Prompt (∗) Neutron Fission Spectra (χ).

Grp	Pu-240	Pu-241	Pu-242	Pu-243*	Pu-244*	Am-241
1	1.0688E-04	9.9131E-05	1.1203E-04	1.0373E-04	8.0233E-05	1.2256E-04
2	3.5395E-04	3.3346E-04	3.6790E-04	3.4656E-04	2.8001E-04	4.2583E-04
3	1.7035E-03	1.6270E-03	1.7565E-03	1.6800E-03	1.4154E-03	2.1115E-03
4	3.4533E-03	3.3310E-03	3.5396E-03	3.4224E-03	2.9767E-03	4.3297E-03
5	7.0281E-03	6.8281E-03	7.1706E-03	6.9886E-03	6.2303E-03	8.8141E-03
6	1.8356E-02	1.7958E-02	1.8641E-02	1.8308E-02	1.6736E-02	2.2818E-02
7	3.2789E-02	3.2273E-02	3.3157E-02	3.2787E-02	3.0671E-02	4.0043E-02
8	8.1606E-02	8.0801E-02	8.2167E-02	8.1786E-02	7.8337E-02	9.6552E-02
9	7.4564E-02	7.4142E-02	7.4830E-02	7.4839E-02	7.2968E-02	8.5095E-02
10	4.1856E-02	4.1706E-02	4.1938E-02	4.2036E-02	4.1352E-02	4.6612E-02
11	4.3970E-02	4.3863E-02	4.4012E-02	4.4172E-02	4.3684E-02	4.8117E-02
12	1.8782E-02	1.8750E-02	1.8787E-02	1.8871E-02	1.8727E-02	2.0292E-02
13	3.7721E-03	3.7666E-03	3.7722E-03	3.7901E-03	3.7655E-03	4.0567E-03
14	2.2794E-02	2.2766E-02	2.2789E-02	2.2903E-02	2.2785E-02	2.4381E-02
15	6.8941E-02	6.8915E-02	6.8866E-02	6.9277E-02	6.9209E-02	7.2353E-02
16	6.8017E-02	6.8060E-02	6.7860E-02	6.8341E-02	6.8657E-02	6.9315E-02
17	8.6025E-02	8.6162E-02	8.5727E-02	8.6378E-02	8.7258E-02	8.4625E-02
18	1.1322E-01	1.1353E-01	1.1270E-01	1.1348E-01	1.1537E-01	1.0602E-01
19	6.2914E-02	6.3153E-02	6.2581E-02	6.2902E-02	6.4275E-02	5.6398E-02
20	2.7599E-02	2.7718E-02	2.7447E-02	2.7548E-02	2.8220E-02	2.4203E-02
21	4.7836E-02	4.8065E-02	4.7563E-02	4.7683E-02	4.8948E-02	4.1157E-02
22	3.8904E-02	3.9177E-02	3.8762E-02	3.8638E-02	3.9757E-02	3.2713E-02
23	4.4103E-02	4.4489E-02	4.4014E-02	4.3668E-02	4.5036E-02	3.6398E-02
24	2.3146E-02	2.3368E-02	2.3102E-02	2.2887E-02	2.3651E-02	1.8844E-02
25	3.3461E-02	3.3831E-02	3.3438E-02	3.3034E-02	3.4196E-02	2.6893E-02
26	1.7761E-02	1.8011E-02	1.7804E-02	1.7494E-02	1.8143E-02	1.4073E-02
27	8.7722E-03	8.9229E-03	8.8335E-03	8.6243E-03	8.9543E-03	6.8829E-03
28	4.2466E-03	4.3213E-03	4.2731E-03	4.1807E-03	4.3437E-03	3.3165E-03
29	1.2068E-03	1.2321E-03	1.2185E-03	1.1861E-03	1.2328E-03	9.3914E-04
30	6.9385E-04	7.1082E-04	7.0339E-04	6.7943E-04	7.0626E-04	5.3852E-04
31	2.1201E-04	2.1767E-04	2.1550E-04	2.0710E-04	2.1530E-04	1.6429E-04
32	2.4880E-04	2.5572E-04	2.5322E-04	2.4279E-04	2.5241E-04	1.9267E-04
33	6.6772E-04	6.9018E-04	6.8468E-04	6.4747E-04	6.7318E-04	5.1518E-04
34	6.0117E-04	6.2326E-04	6.1889E-04	5.8211E-04	6.0531E-04	4.6308E-04
35	1.9440E-04	2.0086E-04	1.9932E-04	1.8982E-04	1.9740E-04	1.5018E-04
36	6.4057E-05	6.6964E-05	6.6697E-05	6.1766E-05	6.4237E-05	4.9201E-05
37	2.6744E-05	2.8510E-05	2.8575E-05	2.5199E-05	2.6207E-05	2.0316E-05
38	3.4446E-06	3.8028E-06	3.8534E-06	3.1027E-06	3.2262E-06	2.5543E-06
39	1.1839E-06	1.3496E-06	1.3809E-06	1.0192E-06	1.0594E-06	8.5305E-07
40	4.8717E-07	5.7944E-07	6.0029E-07	3.9263E-07	4.0779E-07	3.3323E-07
41	1.4576E-07	1.8356E-07	1.9323E-07	1.0606E-07	1.0987E-07	8.8436E-08
42	2.5378E-08	3.3327E-08	3.5495E-08	1.6921E-08	1.7436E-08	1.2418E-08
43	1.4106E-08	1.8592E-08	1.9843E-08	9.3150E-09	9.5458E-09	5.3663E-09
44	4.8111E-09	6.1873E-09	6.5734E-09	3.3422E-09	3.4068E-09	1.2832E-09
45	2.7145E-09	3.3651E-09	3.5461E-09	2.0232E-09	2.0564E-09	5.1375E-10
46	2.5622E-09	3.0056E-09	3.1248E-09	2.0972E-09	2.1275E-09	2.9693E-10
47	2.0925E-09	2.2372E-09	2.2677E-09	1.9524E-09	1.9784E-09	7.8735E-11

TAB. C.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
Total or Prompt (∗) Neutron Fission Spectra (χ).

Grp	Am-242m*	Am-243	Cm-241*	Cm-242	Cm-243*	Cm-244*
1	1.5532E-04	1.4011E-04	8.0233E-05	8.0203E-05	6.5734E-04	1.5294E-04
2	4.3614E-04	4.7371E-04	2.8001E-04	2.7991E-04	6.6425E-04	4.7626E-04
3	1.7898E-03	2.2911E-03	1.4154E-03	1.4148E-03	2.2612E-03	2.1626E-03
4	3.7038E-03	4.6128E-03	2.9767E-03	2.9756E-03	4.4204E-03	4.1942E-03
5	7.6268E-03	9.2641E-03	6.2303E-03	6.2279E-03	8.7552E-03	8.2507E-03
6	2.0234E-02	2.3668E-02	1.6736E-02	1.6730E-02	2.2378E-02	2.0820E-02
7	3.6215E-02	4.1053E-02	3.0671E-02	3.0660E-02	3.8576E-02	3.6032E-02
8	8.9288E-02	9.7882E-02	7.8337E-02	7.8307E-02	9.2823E-02	8.6836E-02
9	8.1009E-02	8.5580E-02	7.2968E-02	7.2940E-02	8.2644E-02	7.7437E-02
10	4.5286E-02	4.6701E-02	4.1352E-02	4.1336E-02	4.5775E-02	4.2939E-02
11	4.7361E-02	4.8106E-02	4.3684E-02	4.3667E-02	4.7634E-02	4.4780E-02
12	2.0144E-02	2.0261E-02	1.8727E-02	1.8720E-02	2.0162E-02	1.9037E-02
13	4.0389E-03	4.0485E-03	3.7655E-03	3.7641E-03	4.0378E-03	3.8169E-03
14	2.4343E-02	2.4320E-02	2.2785E-02	2.2776E-02	2.4305E-02	2.3023E-02
15	7.2874E-02	7.2061E-02	6.9209E-02	6.9183E-02	7.2393E-02	6.9222E-02
16	7.0542E-02	6.8902E-02	6.8657E-02	6.8632E-02	6.9724E-02	6.7744E-02
17	8.6904E-02	8.3984E-02	8.7258E-02	8.7230E-02	8.5553E-02	8.4955E-02
18	1.1030E-01	1.0508E-01	1.1537E-01	1.1535E-01	1.0806E-01	1.1062E-01
19	5.9474E-02	5.5860E-02	6.4275E-02	6.4275E-02	5.8047E-02	6.0871E-02
20	2.5729E-02	2.3967E-02	2.8220E-02	2.8224E-02	2.5028E-02	2.6564E-02
21	4.3985E-02	4.0746E-02	4.8948E-02	4.8960E-02	4.2678E-02	4.5846E-02
22	3.4961E-02	3.2421E-02	3.9757E-02	3.9781E-02	3.3780E-02	3.7025E-02
23	3.8749E-02	3.6098E-02	4.5036E-02	4.5079E-02	3.7420E-02	4.1708E-02
24	1.9811E-02	1.8685E-02	2.3651E-02	2.3677E-02	1.9105E-02	2.1800E-02
25	2.7799E-02	2.6670E-02	3.4196E-02	3.4245E-02	2.6831E-02	3.1386E-02
26	1.4244E-02	1.3967E-02	1.8143E-02	1.8175E-02	1.3755E-02	1.6579E-02
27	6.8606E-03	6.8394E-03	8.9543E-03	8.9726E-03	6.6177E-03	8.1602E-03
28	3.2780E-03	3.2937E-03	4.3436E-03	4.3529E-03	3.1521E-03	3.9519E-03
29	9.2229E-04	9.3380E-04	1.2328E-03	1.2355E-03	8.8827E-04	1.1207E-03
30	5.2640E-04	5.3629E-04	7.0626E-04	7.0828E-04	5.0312E-04	6.4180E-04
31	1.5971E-04	1.6377E-04	2.1530E-04	2.1600E-04	1.5333E-04	1.9561E-04
32	1.8715E-04	1.9214E-04	2.5241E-04	2.5328E-04	1.8026E-04	2.2931E-04
33	4.9715E-04	5.1495E-04	6.7318E-04	6.7588E-04	4.8111E-04	6.1144E-04
34	4.3424E-04	4.6326E-04	6.0531E-04	6.0778E-04	4.2047E-04	5.4960E-04
35	1.0591E-04	1.4988E-04	1.9740E-04	1.9798E-04	1.0256E-04	1.7919E-04
36	2.3632E-05	4.9342E-05	6.4237E-05	6.4519E-05	2.2884E-05	5.8305E-05
37	6.2304E-06	2.0548E-05	2.6207E-05	2.6392E-05	6.0330E-06	2.3788E-05
38	4.3285E-07	2.6260E-06	3.2262E-06	3.2664E-06	4.1914E-07	2.9298E-06
39	9.6577E-08	8.9146E-07	1.0594E-06	1.0786E-06	9.3517E-08	9.6287E-07
40	2.3985E-08	3.5689E-07	4.0779E-07	4.1870E-07	2.3225E-08	3.7139E-07
41	3.4459E-09	9.8902E-08	1.0987E-07	1.1443E-07	3.3367E-09	1.0071E-07
42	2.3940E-10	1.4717E-08	1.7436E-08	1.8404E-08	2.3181E-10	1.6194E-08
43	5.9455E-11	6.6883E-09	9.5458E-09	1.0094E-08	5.7571E-11	8.9881E-09
44	7.2294E-12	1.6937E-09	3.4068E-09	3.5748E-09	5.7006E-12	3.2499E-09
45	1.6131E-12	7.0883E-10	2.0564E-09	2.1356E-09	4.9404E-15	1.9755E-09
46	4.3609E-13	4.2983E-10	2.1275E-09	2.1810E-09	2.3897E-08	2.0534E-09
47	2.7018E-14	1.2119E-10	1.9784E-09	1.9950E-09	2.3897E-08	1.9146E-09

TAB. C.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
Total or Prompt (∗) Neutron Fission Spectra (χ).

Grp	Cm-245	Cm-246*	Cm-247*	Cm-248*
1	1.2202E-04	1.9676E-04	2.0714E-04	8.0233E-05
2	4.1588E-04	5.1820E-04	6.1452E-04	2.8001E-04
3	2.0310E-03	2.1834E-03	2.6605E-03	1.4154E-03
4	4.1253E-03	4.3124E-03	4.9692E-03	2.9767E-03
5	8.3758E-03	8.6266E-03	9.4910E-03	6.2303E-03
6	2.1708E-02	2.2140E-02	2.3231E-02	1.6736E-02
7	3.8084E-02	3.8552E-02	3.9076E-02	3.0671E-02
8	9.2074E-02	9.2558E-02	9.1440E-02	7.8337E-02
9	8.2251E-02	8.2250E-02	7.9725E-02	7.2968E-02
10	4.5621E-02	4.5514E-02	4.3707E-02	4.1352E-02
11	4.7500E-02	4.7330E-02	4.5274E-02	4.3684E-02
12	2.0147E-02	2.0060E-02	1.9162E-02	1.8727E-02
13	4.0354E-03	4.0170E-03	3.8364E-03	3.7655E-03
14	2.4303E-02	2.4187E-02	2.3101E-02	2.2785E-02
15	7.2571E-02	7.2189E-02	6.9084E-02	6.9209E-02
16	7.0032E-02	6.9649E-02	6.7124E-02	6.8657E-02
17	8.6075E-02	8.5562E-02	8.3581E-02	8.7258E-02
18	1.0901E-01	1.0826E-01	1.0794E-01	1.1537E-01
19	5.8665E-02	5.8181E-02	5.9004E-02	6.4275E-02
20	2.5384E-02	2.5158E-02	2.5665E-02	2.8220E-02
21	4.3380E-02	4.2977E-02	4.4176E-02	4.8948E-02
22	3.4525E-02	3.4170E-02	3.5566E-02	3.9757E-02
23	3.8263E-02	3.7896E-02	3.9944E-02	4.5036E-02
24	1.9575E-02	1.9389E-02	2.0824E-02	2.3651E-02
25	2.7478E-02	2.7232E-02	2.9912E-02	3.4196E-02
26	1.3994E-02	1.3966E-02	1.5763E-02	1.8143E-02
27	6.4199E-03	6.7314E-03	7.7471E-03	8.9543E-03
28	2.3975E-03	3.2163E-03	3.7485E-03	4.3436E-03
29	5.5369E-04	9.0447E-04	1.0625E-03	1.2328E-03
30	2.8417E-04	5.1624E-04	6.0836E-04	7.0626E-04
31	8.1016E-05	1.5663E-04	1.8540E-04	2.1530E-04
32	9.1176E-05	1.8354E-04	2.1733E-04	2.5241E-04
33	2.2076E-04	4.8752E-04	5.7942E-04	6.7318E-04
34	1.5724E-04	4.2566E-04	5.2074E-04	6.0531E-04
35	3.5822E-05	1.0381E-04	1.6975E-04	1.9740E-04
36	8.9307E-06	2.3162E-05	5.5231E-05	6.4237E-05
37	3.0211E-06	6.1065E-06	2.2535E-05	2.6207E-05
38	3.7154E-07	4.2424E-07	2.7762E-06	3.2262E-06
39	1.4284E-07	9.4656E-08	9.1284E-07	1.0594E-06
40	7.1551E-08	2.3508E-08	3.5253E-07	4.0778E-07
41	2.7739E-08	3.3774E-09	9.5952E-08	1.0987E-07
42	5.7285E-09	2.3464E-10	1.5547E-08	1.7436E-08
43	3.2189E-09	5.8186E-11	8.6956E-09	9.5458E-09
44	9.8508E-10	6.1796E-12	3.1669E-09	3.4068E-09
45	4.6478E-10	5.8730E-13	1.9324E-09	2.0564E-09
46	3.1538E-10	8.5930E-05	2.0137E-09	2.1275E-09
47	1.0041E-10	8.5930E-05	1.8802E-09	1.9784E-09

APPENDIX D

Flat Weighting Total Neutrons per Fission (ν)

TAB. D.1

Response Functions Included with BUGENDF70.BOLIB.
Total Neutrons per Fission (ν) Collapsed Using Flat Weighting.

Grp	Th-230	Th-232	Pa-231	Pa-233	U-232	U-233
1	3.5822E+00	4.1708E+00	4.0448E+00	4.3609E+00	4.8094E+00	4.4535E+00
2	3.3332E+00	3.8114E+00	3.6744E+00	3.7947E+00	4.5388E+00	4.1644E+00
3	3.1152E+00	3.5327E+00	3.5173E+00	3.5911E+00	4.3116E+00	3.9074E+00
4	2.9374E+00	3.3084E+00	3.4120E+00	3.4791E+00	4.1224E+00	3.6944E+00
5	2.8085E+00	3.1467E+00	3.3387E+00	3.4162E+00	3.9845E+00	3.5409E+00
6	2.6843E+00	2.9552E+00	3.1438E+00	3.1907E+00	3.8467E+00	3.3901E+00
7	2.5594E+00	2.6624E+00	2.7024E+00	2.5972E+00	3.7186E+00	3.2296E+00
8	2.4491E+00	2.5092E+00	2.5805E+00	2.5061E+00	3.5990E+00	3.0571E+00
9	2.3376E+00	2.3568E+00	2.4959E+00	2.4189E+00	3.4806E+00	2.8838E+00
10	2.2965E+00	2.3087E+00	2.4647E+00	2.3866E+00	3.4366E+00	2.8199E+00
11	2.2692E+00	2.2902E+00	2.4437E+00	2.3651E+00	3.4074E+00	2.7774E+00
12	2.2500E+00	2.2769E+00	2.4289E+00	2.3499E+00	3.3868E+00	2.7513E+00
13	2.2455E+00	2.2739E+00	2.4255E+00	2.3464E+00	3.3820E+00	2.7462E+00
14	2.2397E+00	2.2698E+00	2.4210E+00	2.3418E+00	3.3758E+00	2.7395E+00
15	2.2172E+00	2.2551E+00	2.4036E+00	2.3242E+00	3.3517E+00	2.7136E+00
16	2.1886E+00	2.2356E+00	2.3815E+00	2.3015E+00	3.3209E+00	2.6805E+00
17	2.1602E+00	2.2172E+00	2.3597E+00	2.2796E+00	3.2906E+00	2.6477E+00
18	2.1304E+00	2.2014E+00	2.3363E+00	2.2579E+00	3.2578E+00	2.6126E+00
19	2.1040E+00	2.1762E+00	2.3142E+00	2.2349E+00	3.2275E+00	2.5801E+00
20	2.0882E+00	2.1654E+00	2.3045E+00	2.2227E+00	3.2136E+00	2.5652E+00
21	2.0798E+00	2.1586E+00	2.2961E+00	2.2147E+00	3.2021E+00	2.5528E+00
22	2.0657E+00	2.1481E+00	2.2866E+00	2.2053E+00	3.1891E+00	2.5387E+00
23	2.0532E+00	2.1363E+00	2.2787E+00	2.1954E+00	3.1749E+00	2.5236E+00
24	2.0422E+00	2.1303E+00	2.2682E+00	2.1860E+00	3.1639E+00	2.5118E+00
25	2.0355E+00	2.1240E+00	2.2645E+00	2.1808E+00	3.1558E+00	2.5092E+00
26	2.0244E+00	2.1157E+00	2.2563E+00	2.1730E+00	3.1454E+00	2.5080E+00
27	2.0185E+00	2.1112E+00	2.2508E+00	2.1675E+00	3.1391E+00	2.5072E+00
28	2.0152E+00	2.1086E+00	2.2481E+00	2.1649E+00	3.1356E+00	2.5069E+00
29	2.0135E+00	2.1074E+00	2.2468E+00	2.1636E+00	3.1338E+00	2.5066E+00
30	2.0128E+00	2.1069E+00	2.2463E+00	2.1630E+00	3.1330E+00	2.5065E+00
31	2.0125E+00	2.1066E+00	2.2460E+00	2.1628E+00	3.1327E+00	2.5065E+00
32	2.0123E+00	2.1065E+00	2.2459E+00	2.1626E+00	3.1325E+00	2.5065E+00
33	2.0118E+00	2.1061E+00	2.2456E+00	2.1623E+00	3.1320E+00	2.5064E+00
34	2.0110E+00	2.1055E+00	2.2449E+00	2.1616E+00	3.1311E+00	2.5063E+00
35	2.0105E+00	2.1051E+00	2.2445E+00	2.1612E+00	3.1305E+00	2.5063E+00
36	2.0102E+00	0.0000E+00	2.2443E+00	2.1610E+00	3.1303E+00	2.5062E+00
37	2.0101E+00	0.0000E+00	2.2442E+00	2.1609E+00	3.1301E+00	2.5062E+00
38	2.0100E+00	0.0000E+00	2.2441E+00	2.1608E+00	3.1300E+00	2.5062E+00
39	0.0000E+00	0.0000E+00	2.2441E+00	2.1608E+00	3.1300E+00	2.5062E+00
40	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.5027E+00
41	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.4980E+00
42	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.4970E+00
43	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.4968E+00
44	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.4968E+00
45	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.4968E+00
46	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.4968E+00
47	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.4968E+00

TAB. D.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
Total Neutrons per Fission (ν) Collapsed Using Flat Weighting.

Grp	U-234	U-235	U-236	U-237	U-238	Np-237
1	4.3992E+00	4.6414E+00	4.6022E+00	4.8930E+00	4.6979E+00	4.9621E+00
2	4.0893E+00	4.2722E+00	4.2362E+00	4.5276E+00	4.3267E+00	4.6286E+00
3	3.8286E+00	3.9729E+00	3.9181E+00	4.2057E+00	3.9990E+00	4.3322E+00
4	3.6237E+00	3.7283E+00	3.6531E+00	3.9460E+00	3.7301E+00	4.0512E+00
5	3.4744E+00	3.5448E+00	3.4614E+00	3.7663E+00	3.5389E+00	3.8437E+00
6	3.3121E+00	3.3542E+00	3.2750E+00	3.5515E+00	3.3552E+00	3.6412E+00
7	3.1172E+00	3.1320E+00	3.0974E+00	3.2741E+00	3.1756E+00	3.4600E+00
8	2.9553E+00	2.9519E+00	2.9386E+00	3.0636E+00	2.9825E+00	3.2930E+00
9	2.8032E+00	2.8011E+00	2.7778E+00	2.9025E+00	2.7712E+00	3.1265E+00
10	2.7471E+00	2.7519E+00	2.7179E+00	2.8431E+00	2.6987E+00	3.0651E+00
11	2.7096E+00	2.7198E+00	2.6783E+00	2.8038E+00	2.6769E+00	3.0243E+00
12	2.6837E+00	2.6980E+00	2.6505E+00	2.7760E+00	2.6631E+00	2.9954E+00
13	2.6777E+00	2.6930E+00	2.6440E+00	2.7696E+00	2.6600E+00	2.9887E+00
14	2.6700E+00	2.6865E+00	2.6371E+00	2.7612E+00	2.6558E+00	2.9800E+00
15	2.6400E+00	2.6613E+00	2.6110E+00	2.7287E+00	2.6399E+00	2.9463E+00
16	2.6019E+00	2.6286E+00	2.5777E+00	2.6871E+00	2.6194E+00	2.9030E+00
17	2.5643E+00	2.5944E+00	2.5447E+00	2.6459E+00	2.6008E+00	2.8605E+00
18	2.5238E+00	2.5563E+00	2.5117E+00	2.6018E+00	2.5801E+00	2.8147E+00
19	2.4855E+00	2.5216E+00	2.4786E+00	2.5612E+00	2.5580E+00	2.7726E+00
20	2.4686E+00	2.5065E+00	2.4625E+00	2.5426E+00	2.5482E+00	2.7529E+00
21	2.4549E+00	2.4957E+00	2.4511E+00	2.5271E+00	2.5409E+00	2.7372E+00
22	2.4385E+00	2.4863E+00	2.4356E+00	2.5095E+00	2.5319E+00	2.7192E+00
23	2.4222E+00	2.4795E+00	2.4207E+00	2.4906E+00	2.5223E+00	2.7009E+00
24	2.4076E+00	2.4752E+00	2.4082E+00	2.4757E+00	2.5151E+00	2.6840E+00
25	2.3984E+00	2.4708E+00	2.3992E+00	2.4647E+00	2.5097E+00	2.6735E+00
26	2.3840E+00	2.4526E+00	2.3880E+00	2.4507E+00	2.5020E+00	2.6578E+00
27	2.3766E+00	2.4324E+00	2.3810E+00	2.4422E+00	2.4979E+00	2.6485E+00
28	2.3718E+00	2.4241E+00	2.3774E+00	2.4375E+00	2.4958E+00	2.6437E+00
29	2.3696E+00	2.4246E+00	2.3754E+00	2.4351E+00	2.4945E+00	2.6411E+00
30	2.3687E+00	2.4263E+00	2.3746E+00	2.4341E+00	2.4941E+00	2.6400E+00
31	2.3682E+00	2.4273E+00	2.3742E+00	2.4336E+00	2.4939E+00	2.6395E+00
32	2.3680E+00	2.4278E+00	2.3740E+00	2.4334E+00	2.4937E+00	2.6393E+00
33	2.3674E+00	2.4293E+00	2.3735E+00	2.4327E+00	2.4932E+00	2.6386E+00
34	2.3663E+00	2.4331E+00	2.3725E+00	2.4315E+00	2.4928E+00	2.6374E+00
35	2.3655E+00	2.4338E+00	2.3718E+00	2.4307E+00	2.4926E+00	2.6365E+00
36	2.3652E+00	2.4338E+00	2.3715E+00	2.4304E+00	2.4923E+00	2.6362E+00
37	2.3650E+00	2.4338E+00	2.3713E+00	2.4301E+00	2.4922E+00	2.6359E+00
38	2.3649E+00	2.4338E+00	2.3712E+00	2.4301E+00	2.4921E+00	2.6359E+00
39	2.3649E+00	2.4338E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
40	2.3649E+00	2.4351E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
41	2.3649E+00	2.4363E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
42	2.3649E+00	2.4367E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
43	2.3649E+00	2.4367E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
44	2.3649E+00	2.4367E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
45	2.3649E+00	2.4367E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
46	2.3649E+00	2.4367E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
47	2.3649E+00	2.4367E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00

TAB. D.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
Total Neutrons per Fission (ν) Collapsed Using Flat Weighting.

Grp	Np-238	Np-239	Pu-236	Pu-237	Pu-238	Pu-239
1	4.8323E+00	4.7805E+00	4.9502E+00	5.0527E+00	5.2176E+00	5.1592E+00
2	4.4648E+00	4.4530E+00	4.6039E+00	4.6939E+00	4.8426E+00	4.8241E+00
3	4.1456E+00	4.1779E+00	4.3157E+00	4.3934E+00	4.5290E+00	4.5313E+00
4	3.8828E+00	3.9461E+00	4.0749E+00	4.1409E+00	4.2661E+00	4.2763E+00
5	3.6932E+00	3.7791E+00	3.9005E+00	3.9579E+00	4.0752E+00	4.0750E+00
6	3.5094E+00	3.6125E+00	3.7253E+00	3.7755E+00	3.8841E+00	3.8802E+00
7	3.3281E+00	3.4551E+00	3.5610E+00	3.6041E+00	3.7084E+00	3.6867E+00
8	3.1651E+00	3.3108E+00	3.4112E+00	3.4445E+00	3.5436E+00	3.5142E+00
9	2.9999E+00	3.1657E+00	3.2596E+00	3.2886E+00	3.3802E+00	3.3525E+00
10	2.9391E+00	3.1124E+00	3.2037E+00	3.2308E+00	3.3192E+00	3.2989E+00
11	2.8988E+00	3.0770E+00	3.1667E+00	3.1921E+00	3.2788E+00	3.2586E+00
12	2.8702E+00	3.0520E+00	3.1404E+00	3.1648E+00	3.2502E+00	3.2311E+00
13	2.8636E+00	3.0462E+00	3.1343E+00	3.1585E+00	3.2436E+00	3.2251E+00
14	2.8550E+00	3.0387E+00	3.1265E+00	3.1502E+00	3.2350E+00	3.2175E+00
15	2.8216E+00	3.0093E+00	3.0958E+00	3.1181E+00	3.2017E+00	3.1875E+00
16	2.7789E+00	2.9719E+00	3.0566E+00	3.0772E+00	3.1590E+00	3.1450E+00
17	2.7367E+00	2.9348E+00	3.0178E+00	3.0368E+00	3.1168E+00	3.0973E+00
18	2.6914E+00	2.8952E+00	2.9762E+00	2.9936E+00	3.0716E+00	3.0458E+00
19	2.6497E+00	2.8585E+00	2.9379E+00	2.9534E+00	3.0299E+00	3.0027E+00
20	2.6306E+00	2.8416E+00	2.9202E+00	2.9349E+00	3.0107E+00	2.9846E+00
21	2.6145E+00	2.8277E+00	2.9056E+00	2.9196E+00	2.9948E+00	2.9694E+00
22	2.5965E+00	2.8118E+00	2.8890E+00	2.9023E+00	2.9768E+00	2.9524E+00
23	2.5771E+00	2.7954E+00	2.8712E+00	2.8835E+00	2.9575E+00	2.9354E+00
24	2.5619E+00	2.7817E+00	2.8571E+00	2.8689E+00	2.9420E+00	2.9226E+00
25	2.5507E+00	2.7734E+00	2.8467E+00	2.8584E+00	2.9312E+00	2.9131E+00
26	2.5363E+00	2.7592E+00	2.8335E+00	2.8444E+00	2.9165E+00	2.9012E+00
27	2.5275E+00	2.7511E+00	2.8254E+00	2.8360E+00	2.9076E+00	2.8963E+00
28	2.5227E+00	2.7468E+00	2.8211E+00	2.8314E+00	2.9027E+00	2.8958E+00
29	2.5202E+00	2.7448E+00	2.8188E+00	2.8290E+00	2.9002E+00	2.8954E+00
30	2.5192E+00	2.7441E+00	2.8178E+00	2.8280E+00	2.8992E+00	2.8951E+00
31	2.5187E+00	0.0000E+00	2.8174E+00	2.8275E+00	2.8987E+00	2.8950E+00
32	2.5185E+00	0.0000E+00	2.8172E+00	2.8273E+00	2.8984E+00	2.8950E+00
33	2.5178E+00	0.0000E+00	2.8165E+00	2.8267E+00	2.8978E+00	2.8948E+00
34	2.5165E+00	0.0000E+00	2.8154E+00	2.8255E+00	2.8965E+00	2.8946E+00
35	2.5157E+00	0.0000E+00	2.8146E+00	2.8247E+00	2.8957E+00	2.8944E+00
36	2.5154E+00	0.0000E+00	2.8143E+00	2.8243E+00	2.8954E+00	2.8943E+00
37	2.5151E+00	0.0000E+00	2.8141E+00	2.8241E+00	2.8951E+00	2.8861E+00
38	2.5151E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8627E+00
39	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8699E+00
40	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8690E+00
41	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8590E+00
42	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8571E+00
43	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8777E+00
44	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8777E+00
45	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8669E+00
46	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8602E+00
47	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8799E+00

TAB. D.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
Total Neutrons per Fission (ν) Collapsed Using Flat Weighting.

Grp	Pu-240	Pu-241	Pu-242	Pu-243	Pu-244	Am-241
1	5.1853E+00	5.3016E+00	5.2843E+00	6.0342E+00	5.7739E+00	5.3972E+00
2	4.8036E+00	4.9186E+00	4.8869E+00	5.5619E+00	5.3084E+00	5.0787E+00
3	4.4793E+00	4.5884E+00	4.5500E+00	5.1544E+00	4.9191E+00	4.7952E+00
4	4.2099E+00	4.3139E+00	4.2696E+00	4.7968E+00	4.5888E+00	4.5252E+00
5	4.0145E+00	4.1148E+00	4.0666E+00	4.5390E+00	4.3531E+00	4.3235E+00
6	3.8204E+00	3.9191E+00	3.8663E+00	4.2807E+00	4.1188E+00	4.1263E+00
7	3.6381E+00	3.7348E+00	3.6773E+00	4.0352E+00	3.8989E+00	3.9371E+00
8	3.4693E+00	3.5650E+00	3.5013E+00	3.8261E+00	3.6903E+00	3.7646E+00
9	3.3008E+00	3.3910E+00	3.3261E+00	3.6198E+00	3.4866E+00	3.5914E+00
10	3.2386E+00	3.3260E+00	3.2616E+00	3.5436E+00	3.4112E+00	3.5273E+00
11	3.1972E+00	3.2826E+00	3.2186E+00	3.4928E+00	3.3611E+00	3.4848E+00
12	3.1678E+00	3.2520E+00	3.1883E+00	3.4570E+00	3.3256E+00	3.4547E+00
13	3.1610E+00	3.2449E+00	3.1812E+00	3.4487E+00	3.3174E+00	3.4477E+00
14	3.1522E+00	3.2357E+00	3.1721E+00	3.4378E+00	3.3067E+00	3.4387E+00
15	3.1179E+00	3.1997E+00	3.1365E+00	3.3958E+00	3.2652E+00	3.4035E+00
16	3.0743E+00	3.1538E+00	3.0911E+00	3.3421E+00	3.2120E+00	3.3585E+00
17	3.0309E+00	3.1086E+00	3.0462E+00	3.2892E+00	3.1595E+00	3.3142E+00
18	2.9843E+00	3.0602E+00	2.9976E+00	3.2330E+00	3.1037E+00	3.2677E+00
19	2.9420E+00	3.0151E+00	2.9545E+00	3.1814E+00	3.0528E+00	3.2235E+00
20	2.9219E+00	2.9945E+00	2.9334E+00	3.1557E+00	3.0282E+00	3.2025E+00
21	2.9059E+00	2.9773E+00	2.9170E+00	3.1358E+00	3.0095E+00	3.1869E+00
22	2.8876E+00	2.9578E+00	2.8978E+00	3.1126E+00	2.9866E+00	3.1673E+00
23	2.8683E+00	2.9456E+00	2.8776E+00	3.0880E+00	2.9633E+00	3.1469E+00
24	2.8515E+00	2.9453E+00	2.8603E+00	3.0689E+00	2.9430E+00	3.1301E+00
25	2.8405E+00	2.9453E+00	2.8497E+00	3.0547E+00	2.9297E+00	3.1189E+00
26	2.8250E+00	2.9453E+00	2.8333E+00	3.0368E+00	2.9107E+00	3.1030E+00
27	2.8158E+00	2.9453E+00	2.8234E+00	3.0257E+00	2.8996E+00	3.0935E+00
28	2.8109E+00	2.9453E+00	2.8183E+00	3.0197E+00	2.8935E+00	3.0884E+00
29	2.8084E+00	2.9453E+00	2.8156E+00	3.0166E+00	2.8905E+00	3.0858E+00
30	2.8073E+00	2.9453E+00	2.8144E+00	3.0152E+00	2.8892E+00	3.0846E+00
31	2.8068E+00	2.9453E+00	2.8139E+00	3.0147E+00	2.8886E+00	3.0842E+00
32	2.8065E+00	2.9453E+00	2.8137E+00	3.0143E+00	2.8883E+00	3.0839E+00
33	2.8059E+00	2.9453E+00	2.8130E+00	3.0135E+00	2.8874E+00	3.0832E+00
34	2.8046E+00	2.9453E+00	2.8116E+00	3.0119E+00	2.8860E+00	3.0819E+00
35	2.8037E+00	2.9453E+00	2.8108E+00	3.0108E+00	2.8849E+00	3.0810E+00
36	2.8034E+00	2.9453E+00	2.8104E+00	3.0105E+00	2.8845E+00	3.0807E+00
37	2.8031E+00	2.9453E+00	2.8101E+00	3.0102E+00	2.8842E+00	3.0804E+00
38	2.8030E+00	2.9453E+00	2.8100E+00	3.0101E+00	2.8841E+00	3.0803E+00
39	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
40	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
41	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
42	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
43	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
44	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
45	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
46	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
47	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00

TAB. D.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
Total Neutrons per Fission (ν) Collapsed Using Flat Weighting.

Grp	Am-242	Am-242m	Am-243	Cm-241	Cm-242	Cm-243
1	5.3494E+00	5.3459E+00	5.3574E+00	5.3735E+00	6.1398E+00	5.3347E+00
2	5.0657E+00	5.0634E+00	5.0272E+00	4.9540E+00	5.7050E+00	5.0610E+00
3	4.8108E+00	4.8117E+00	4.7511E+00	4.6026E+00	5.3411E+00	4.8231E+00
4	4.5861E+00	4.5874E+00	4.5184E+00	4.3101E+00	5.0342E+00	4.6196E+00
5	4.4218E+00	4.4230E+00	4.3490E+00	4.0955E+00	4.8136E+00	4.4678E+00
6	4.2462E+00	4.2462E+00	4.1809E+00	3.8765E+00	4.5964E+00	4.3101E+00
7	4.0785E+00	4.0769E+00	4.0211E+00	3.6789E+00	4.3853E+00	4.1513E+00
8	3.9184E+00	3.9178E+00	3.8714E+00	3.5000E+00	4.1994E+00	4.0079E+00
9	3.7564E+00	3.7553E+00	3.7215E+00	3.3114E+00	4.0046E+00	3.8623E+00
10	3.6959E+00	3.6946E+00	3.6658E+00	3.2434E+00	3.9328E+00	3.8085E+00
11	3.6555E+00	3.6543E+00	3.6287E+00	3.1982E+00	3.8860E+00	3.7731E+00
12	3.6270E+00	3.6257E+00	3.6024E+00	3.1664E+00	3.8528E+00	3.7480E+00
13	3.6204E+00	3.6191E+00	3.5963E+00	3.1590E+00	3.8451E+00	3.7422E+00
14	3.6118E+00	3.6105E+00	3.5884E+00	3.1493E+00	3.8352E+00	3.7347E+00
15	3.5784E+00	3.5772E+00	3.5577E+00	3.1117E+00	3.7965E+00	3.7053E+00
16	3.5358E+00	3.5346E+00	3.5183E+00	3.0643E+00	3.7476E+00	3.6674E+00
17	3.4935E+00	3.4924E+00	3.4793E+00	3.0170E+00	3.6989E+00	3.6298E+00
18	3.4483E+00	3.4469E+00	3.4381E+00	2.9659E+00	3.6471E+00	3.5894E+00
19	3.4066E+00	3.4052E+00	3.4004E+00	2.9198E+00	3.5989E+00	3.5523E+00
20	3.3874E+00	3.3861E+00	3.3812E+00	2.8977E+00	3.5748E+00	3.5351E+00
21	3.3714E+00	3.3700E+00	3.3669E+00	2.8797E+00	3.5574E+00	3.5210E+00
22	3.3533E+00	3.3520E+00	3.3497E+00	2.8594E+00	3.5358E+00	3.5048E+00
23	3.3338E+00	3.3325E+00	3.3322E+00	2.8374E+00	3.5137E+00	3.4874E+00
24	3.3186E+00	3.3173E+00	3.3168E+00	2.8204E+00	3.4948E+00	3.4738E+00
25	3.3073E+00	3.3060E+00	3.3066E+00	2.8086E+00	3.4828E+00	3.4636E+00
26	3.2931E+00	3.2918E+00	3.2928E+00	2.7918E+00	3.4649E+00	3.4509E+00
27	3.2843E+00	3.2830E+00	3.2845E+00	2.7821E+00	3.4546E+00	3.4431E+00
28	3.2795E+00	3.2782E+00	3.2801E+00	2.7767E+00	3.4489E+00	3.4389E+00
29	3.2770E+00	3.2757E+00	3.2777E+00	2.7738E+00	3.4461E+00	3.4367E+00
30	3.2760E+00	3.2747E+00	3.2767E+00	2.7727E+00	3.4448E+00	3.4357E+00
31	3.2755E+00	3.2742E+00	3.2763E+00	2.7721E+00	3.4443E+00	3.4353E+00
32	3.2753E+00	3.2740E+00	3.2761E+00	2.7719E+00	3.4440E+00	3.4351E+00
33	3.2746E+00	3.2733E+00	3.2754E+00	2.7711E+00	3.4432E+00	3.4345E+00
34	3.2733E+00	3.2720E+00	3.2743E+00	2.7697E+00	3.4420E+00	3.4334E+00
35	3.2725E+00	3.2712E+00	3.2735E+00	2.7688E+00	0.0000E+00	3.4326E+00
36	3.2722E+00	3.2709E+00	3.2732E+00	2.7684E+00	0.0000E+00	3.4323E+00
37	3.2719E+00	3.2706E+00	3.2729E+00	2.7681E+00	0.0000E+00	3.4321E+00
38	3.2719E+00	3.2706E+00	3.2729E+00	2.7681E+00	3.4400E+00	3.4320E+00
39	3.2718E+00	3.2705E+00	3.2729E+00	2.7680E+00	3.4400E+00	3.4320E+00
40	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00
41	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00
42	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00
43	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00
44	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00
45	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00
46	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00
47	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00

TAB. D.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
Total Neutrons per Fission (ν) Collapsed Using Flat Weighting.

Grp	Cm-244	Cm-245	Cm-246	Cm-247	Cm-248
1	6.1322E+00	5.4192E+00	5.5186E+00	6.9691E+00	6.7547E+00
2	5.6723E+00	5.1549E+00	5.2522E+00	6.4594E+00	6.2290E+00
3	5.2761E+00	4.9212E+00	5.0107E+00	6.0348E+00	5.7889E+00
4	4.9497E+00	4.7218E+00	4.8083E+00	5.6723E+00	5.4190E+00
5	4.7137E+00	4.5731E+00	4.6612E+00	5.4131E+00	5.1507E+00
6	4.4763E+00	4.4281E+00	4.5158E+00	5.1581E+00	4.8868E+00
7	4.2554E+00	4.2861E+00	4.3575E+00	4.9135E+00	4.6330E+00
8	4.0524E+00	4.1405E+00	4.2078E+00	4.6895E+00	4.4041E+00
9	3.8473E+00	3.9938E+00	4.0592E+00	4.4650E+00	4.1716E+00
10	3.7717E+00	3.9394E+00	4.0041E+00	4.3822E+00	4.0861E+00
11	3.7215E+00	3.9028E+00	3.9673E+00	4.3271E+00	4.0294E+00
12	3.6860E+00	3.8770E+00	3.9412E+00	4.2882E+00	3.9893E+00
13	3.6778E+00	3.8710E+00	3.9351E+00	4.2792E+00	3.9799E+00
14	3.6671E+00	3.8632E+00	3.9273E+00	4.2675E+00	3.9679E+00
15	3.6258E+00	3.8329E+00	3.8969E+00	4.2216E+00	3.9210E+00
16	3.5726E+00	3.7943E+00	3.8577E+00	4.1638E+00	3.8610E+00
17	3.5198E+00	3.7564E+00	3.8187E+00	4.1064E+00	3.8017E+00
18	3.4634E+00	3.7156E+00	3.7778E+00	4.0447E+00	3.7382E+00
19	3.4122E+00	3.6775E+00	3.7398E+00	3.9874E+00	3.6806E+00
20	3.3883E+00	3.6603E+00	3.7214E+00	3.9612E+00	3.6531E+00
21	3.3693E+00	3.6459E+00	3.7074E+00	3.9393E+00	3.6318E+00
22	3.3470E+00	3.6297E+00	3.6905E+00	3.9147E+00	3.6065E+00
23	3.3234E+00	3.6122E+00	3.6729E+00	3.8880E+00	3.5783E+00
24	3.3032E+00	3.5987E+00	3.6580E+00	3.8673E+00	3.5568E+00
25	3.2904E+00	3.5964E+00	3.6482E+00	3.8519E+00	3.5406E+00
26	3.2712E+00	3.5964E+00	3.6341E+00	3.8324E+00	3.5204E+00
27	3.2598E+00	3.5964E+00	3.6257E+00	3.8204E+00	3.5079E+00
28	3.2539E+00	3.5964E+00	3.6212E+00	3.8139E+00	3.5010E+00
29	3.2508E+00	3.5964E+00	3.6190E+00	3.8105E+00	3.4973E+00
30	3.2495E+00	3.5964E+00	3.6180E+00	3.8091E+00	3.4959E+00
31	3.2489E+00	3.5964E+00	3.6176E+00	3.8084E+00	3.4952E+00
32	3.2486E+00	3.5964E+00	3.6174E+00	3.8081E+00	3.4948E+00
33	3.2478E+00	3.5964E+00	3.6167E+00	3.8072E+00	3.4939E+00
34	3.2463E+00	3.5964E+00	3.6156E+00	3.8055E+00	3.4922E+00
35	3.2452E+00	3.5964E+00	3.6148E+00	3.8043E+00	3.4910E+00
36	3.2448E+00	3.5964E+00	3.6145E+00	3.8039E+00	3.4905E+00
37	3.2445E+00	3.5964E+00	3.6143E+00	3.8036E+00	3.4902E+00
38	3.2444E+00	3.5964E+00	3.6142E+00	3.8035E+00	3.4901E+00
39	3.2444E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
40	3.2444E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
41	3.2443E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
42	3.2443E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
43	3.2443E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
44	3.2443E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
45	3.2443E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
46	3.2443E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
47	3.2443E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00

APPENDIX E

1/4 T PV Weighting Total Neutrons per Fission (*v*)

TAB. E.1

Response Functions Included with BUGENDF70.BOLIB.
Total Neutrons per Fission (ν) Collapsed Using 1/4 T PV Weighting.

Grp	Th-230	Th-232	Pa-231	Pa-233	U-232	U-233
1	3.5396E+00	4.1032E+00	3.9842E+00	4.2898E+00	4.7642E+00	4.4055E+00
2	3.3159E+00	3.7894E+00	3.6573E+00	3.7644E+00	4.5217E+00	4.1448E+00
3	3.0940E+00	3.5058E+00	3.5042E+00	3.5769E+00	4.2890E+00	3.8820E+00
4	2.9313E+00	3.3008E+00	3.4085E+00	3.4758E+00	4.1159E+00	3.6871E+00
5	2.8036E+00	3.1406E+00	3.3353E+00	3.4136E+00	3.9792E+00	3.5351E+00
6	2.6799E+00	2.9467E+00	3.1275E+00	3.1677E+00	3.8411E+00	3.3844E+00
7	2.5574E+00	2.6590E+00	2.6984E+00	2.5953E+00	3.7163E+00	3.2264E+00
8	2.4394E+00	2.4962E+00	2.5731E+00	2.4984E+00	3.5886E+00	3.0420E+00
9	2.3409E+00	2.3614E+00	2.4984E+00	2.4214E+00	3.4841E+00	2.8889E+00
10	2.2962E+00	2.3085E+00	2.4645E+00	2.3864E+00	3.4363E+00	2.8195E+00
11	2.2694E+00	2.2903E+00	2.4438E+00	2.3653E+00	3.4076E+00	2.7776E+00
12	2.2513E+00	2.2778E+00	2.4299E+00	2.3510E+00	3.3882E+00	2.7528E+00
13	2.2455E+00	2.2739E+00	2.4255E+00	2.3464E+00	3.3820E+00	2.7462E+00
14	2.2387E+00	2.2691E+00	2.4202E+00	2.3410E+00	3.3747E+00	2.7383E+00
15	2.2174E+00	2.2553E+00	2.4038E+00	2.3244E+00	3.3520E+00	2.7139E+00
16	2.1878E+00	2.2349E+00	2.3808E+00	2.3009E+00	3.3200E+00	2.6795E+00
17	2.1600E+00	2.2170E+00	2.3595E+00	2.2794E+00	3.2903E+00	2.6475E+00
18	2.1280E+00	2.2007E+00	2.3343E+00	2.2566E+00	3.2551E+00	2.6096E+00
19	2.1034E+00	2.1757E+00	2.3140E+00	2.2344E+00	3.2271E+00	2.5797E+00
20	2.0882E+00	2.1654E+00	2.3045E+00	2.2227E+00	3.2136E+00	2.5652E+00
21	2.0792E+00	2.1581E+00	2.2956E+00	2.2141E+00	3.2014E+00	2.5520E+00
22	2.0661E+00	2.1483E+00	2.2868E+00	2.2055E+00	3.1895E+00	2.5392E+00
23	2.0542E+00	2.1368E+00	2.2793E+00	2.1961E+00	3.1758E+00	2.5246E+00
24	2.0435E+00	2.1311E+00	2.2692E+00	2.1872E+00	3.1654E+00	2.5135E+00
25	2.0350E+00	2.1236E+00	2.2637E+00	2.1804E+00	3.1556E+00	2.5091E+00
26	2.0247E+00	2.1159E+00	2.2565E+00	2.1732E+00	3.1457E+00	2.5080E+00
27	2.0185E+00	2.1112E+00	2.2509E+00	2.1675E+00	3.1392E+00	2.5072E+00
28	2.0154E+00	2.1088E+00	2.2483E+00	2.1650E+00	3.1357E+00	2.5069E+00
29	2.0136E+00	2.1075E+00	2.2469E+00	2.1637E+00	3.1339E+00	2.5067E+00
30	2.0128E+00	2.1068E+00	2.2462E+00	2.1630E+00	3.1330E+00	2.5065E+00
31	2.0125E+00	2.1066E+00	2.2460E+00	2.1628E+00	3.1327E+00	2.5065E+00
32	2.0123E+00	2.1065E+00	2.2459E+00	2.1626E+00	3.1325E+00	2.5065E+00
33	2.0118E+00	2.1061E+00	2.2455E+00	2.1622E+00	3.1320E+00	2.5064E+00
34	2.0111E+00	2.1056E+00	2.2450E+00	2.1617E+00	3.1312E+00	2.5063E+00
35	2.0105E+00	2.1051E+00	2.2445E+00	2.1612E+00	3.1305E+00	2.5063E+00
36	2.0102E+00	0.0000E+00	2.2443E+00	2.1610E+00	3.1302E+00	2.5062E+00
37	2.0101E+00	0.0000E+00	2.2442E+00	2.1609E+00	3.1301E+00	2.5062E+00
38	2.0100E+00	0.0000E+00	2.2441E+00	2.1608E+00	3.1300E+00	2.5062E+00
39	0.0000E+00	0.0000E+00	2.2441E+00	2.1608E+00	3.1300E+00	2.5062E+00
40	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.5027E+00
41	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.4981E+00
42	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.4970E+00
43	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.4969E+00
44	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.4968E+00
45	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.4968E+00
46	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.4968E+00
47	0.0000E+00	0.0000E+00	2.2441E+00	0.0000E+00	3.1300E+00	2.4968E+00

TAB. E.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
Total Neutrons per Fission (ν) Collapsed Using 1/4 T PV Weighting.

Grp	U-234	U-235	U-236	U-237	U-238	Np-237
1	4.3512E+00	4.5795E+00	4.5461E+00	4.8357E+00	4.6403E+00	4.9064E+00
2	4.0686E+00	4.2477E+00	4.2116E+00	4.5151E+00	4.3011E+00	4.6071E+00
3	3.8041E+00	3.9443E+00	3.8865E+00	4.1693E+00	3.9670E+00	4.3004E+00
4	3.6165E+00	3.7196E+00	3.6440E+00	3.9382E+00	3.7209E+00	4.0413E+00
5	3.4687E+00	3.5379E+00	3.4541E+00	3.7590E+00	3.5317E+00	3.8358E+00
6	3.3046E+00	3.3456E+00	3.2683E+00	3.5415E+00	3.3487E+00	3.6338E+00
7	3.1139E+00	3.1283E+00	3.0944E+00	3.2689E+00	3.1725E+00	3.4568E+00
8	2.9421E+00	2.9373E+00	2.9247E+00	3.0495E+00	2.9642E+00	3.2785E+00
9	2.8077E+00	2.8050E+00	2.7826E+00	2.9072E+00	2.7775E+00	3.1313E+00
10	2.7467E+00	2.7516E+00	2.7175E+00	2.8427E+00	2.6984E+00	3.0647E+00
11	2.7098E+00	2.7200E+00	2.6785E+00	2.8041E+00	2.6770E+00	3.0246E+00
12	2.6854E+00	2.6995E+00	2.6524E+00	2.7779E+00	2.6641E+00	2.9974E+00
13	2.6777E+00	2.6930E+00	2.6440E+00	2.7696E+00	2.6600E+00	2.9887E+00
14	2.6686E+00	2.6854E+00	2.6359E+00	2.7597E+00	2.6551E+00	2.9784E+00
15	2.6403E+00	2.6616E+00	2.6113E+00	2.7291E+00	2.6401E+00	2.9466E+00
16	2.6008E+00	2.6276E+00	2.5768E+00	2.6859E+00	2.6188E+00	2.9018E+00
17	2.5640E+00	2.5941E+00	2.5444E+00	2.6456E+00	2.6006E+00	2.8601E+00
18	2.5204E+00	2.5531E+00	2.5090E+00	2.5980E+00	2.5785E+00	2.8109E+00
19	2.4851E+00	2.5211E+00	2.4781E+00	2.5607E+00	2.5577E+00	2.7720E+00
20	2.4686E+00	2.5065E+00	2.4624E+00	2.5425E+00	2.5481E+00	2.7529E+00
21	2.4540E+00	2.4950E+00	2.4503E+00	2.5260E+00	2.5404E+00	2.7361E+00
22	2.4390E+00	2.4865E+00	2.4360E+00	2.5100E+00	2.5321E+00	2.7197E+00
23	2.4235E+00	2.4800E+00	2.4219E+00	2.4918E+00	2.5230E+00	2.7024E+00
24	2.4094E+00	2.4758E+00	2.4098E+00	2.4778E+00	2.5160E+00	2.6860E+00
25	2.3977E+00	2.4707E+00	2.3990E+00	2.4645E+00	2.5094E+00	2.6728E+00
26	2.3843E+00	2.4535E+00	2.3883E+00	2.4511E+00	2.5021E+00	2.6582E+00
27	2.3767E+00	2.4325E+00	2.3810E+00	2.4422E+00	2.4979E+00	2.6486E+00
28	2.3720E+00	2.4243E+00	2.3775E+00	2.4377E+00	2.4959E+00	2.6439E+00
29	2.3698E+00	2.4244E+00	2.3756E+00	2.4352E+00	2.4945E+00	2.6413E+00
30	2.3686E+00	2.4264E+00	2.3745E+00	2.4340E+00	2.4941E+00	2.6400E+00
31	2.3682E+00	2.4272E+00	2.3742E+00	2.4336E+00	2.4939E+00	2.6396E+00
32	2.3680E+00	2.4279E+00	2.3740E+00	2.4333E+00	2.4937E+00	2.6393E+00
33	2.3674E+00	2.4294E+00	2.3734E+00	2.4327E+00	2.4932E+00	2.6386E+00
34	2.3664E+00	2.4327E+00	2.3725E+00	2.4316E+00	2.4928E+00	2.6375E+00
35	2.3656E+00	2.4338E+00	2.3718E+00	2.4308E+00	2.4926E+00	2.6365E+00
36	2.3652E+00	2.4338E+00	2.3715E+00	2.4304E+00	2.4923E+00	2.6362E+00
37	2.3650E+00	2.4338E+00	2.3713E+00	2.4301E+00	2.4922E+00	2.6359E+00
38	2.3649E+00	2.4338E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6359E+00
39	2.3649E+00	2.4338E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
40	2.3649E+00	2.4351E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
41	2.3649E+00	2.4363E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
42	2.3649E+00	2.4367E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
43	2.3649E+00	2.4367E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
44	2.3649E+00	2.4367E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
45	2.3649E+00	2.4367E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
46	2.3649E+00	2.4367E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00
47	2.3649E+00	2.4367E+00	2.3712E+00	2.4300E+00	2.4921E+00	2.6358E+00

TAB. E.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
Total Neutrons per Fission (ν) Collapsed Using 1/4 T PV Weighting.

Grp	Np-238	Np-239	Pu-236	Pu-237	Pu-238	Pu-239
1	4.7701E+00	4.7255E+00	4.8921E+00	4.9927E+00	5.1547E+00	5.1047E+00
2	4.4403E+00	4.4319E+00	4.5821E+00	4.6711E+00	4.8189E+00	4.8016E+00
3	4.1143E+00	4.1501E+00	4.2870E+00	4.3633E+00	4.4976E+00	4.5027E+00
4	3.8737E+00	3.9382E+00	4.0666E+00	4.1322E+00	4.2570E+00	4.2668E+00
5	3.6861E+00	3.7727E+00	3.8938E+00	3.9509E+00	4.0680E+00	4.0676E+00
6	3.5019E+00	3.6060E+00	3.7182E+00	3.7678E+00	3.8764E+00	3.8719E+00
7	3.3250E+00	3.4524E+00	3.5581E+00	3.6011E+00	3.7053E+00	3.6834E+00
8	3.1506E+00	3.2981E+00	3.3980E+00	3.4309E+00	3.5292E+00	3.4991E+00
9	3.0047E+00	3.1699E+00	3.2640E+00	3.2932E+00	3.3850E+00	3.3566E+00
10	2.9387E+00	3.1121E+00	3.2033E+00	3.2304E+00	3.3188E+00	3.2985E+00
11	2.8990E+00	3.0772E+00	3.1669E+00	3.1923E+00	3.2790E+00	3.2589E+00
12	2.8722E+00	3.0537E+00	3.1422E+00	3.1667E+00	3.2522E+00	3.2329E+00
13	2.8636E+00	3.0462E+00	3.1343E+00	3.1585E+00	3.2436E+00	3.2251E+00
14	2.8535E+00	3.0373E+00	3.1250E+00	3.1488E+00	3.2335E+00	3.2162E+00
15	2.8220E+00	3.0097E+00	3.0962E+00	3.1185E+00	3.2021E+00	3.1879E+00
16	2.7777E+00	2.9708E+00	3.0554E+00	3.0760E+00	3.1577E+00	3.1437E+00
17	2.7364E+00	2.9346E+00	3.0175E+00	3.0365E+00	3.1165E+00	3.0969E+00
18	2.6876E+00	2.8918E+00	2.9727E+00	2.9899E+00	3.0677E+00	3.0416E+00
19	2.6491E+00	2.8580E+00	2.9374E+00	2.9528E+00	3.0293E+00	3.0022E+00
20	2.6305E+00	2.8416E+00	2.9202E+00	2.9349E+00	3.0106E+00	2.9845E+00
21	2.6135E+00	2.8268E+00	2.9047E+00	2.9186E+00	2.9937E+00	2.9684E+00
22	2.5970E+00	2.8123E+00	2.8895E+00	2.9028E+00	2.9773E+00	2.9529E+00
23	2.5784E+00	2.7966E+00	2.8723E+00	2.8847E+00	2.9589E+00	2.9365E+00
24	2.5640E+00	2.7835E+00	2.8591E+00	2.8710E+00	2.9441E+00	2.9244E+00
25	2.5504E+00	2.7726E+00	2.8465E+00	2.8581E+00	2.9308E+00	2.9129E+00
26	2.5367E+00	2.7597E+00	2.8339E+00	2.8448E+00	2.9169E+00	2.9015E+00
27	2.5276E+00	2.7512E+00	2.8255E+00	2.8361E+00	2.9077E+00	2.8963E+00
28	2.5230E+00	2.7471E+00	2.8213E+00	2.8316E+00	2.9029E+00	2.8958E+00
29	2.5204E+00	2.7449E+00	2.8189E+00	2.8292E+00	2.9004E+00	2.8954E+00
30	2.5191E+00	2.7441E+00	2.8178E+00	2.8280E+00	2.8991E+00	2.8951E+00
31	2.5187E+00	0.0000E+00	2.8174E+00	2.8276E+00	2.8987E+00	2.8950E+00
32	2.5184E+00	0.0000E+00	2.8171E+00	2.8273E+00	2.8984E+00	2.8950E+00
33	2.5177E+00	0.0000E+00	2.8165E+00	2.8266E+00	2.8978E+00	2.8948E+00
34	2.5166E+00	0.0000E+00	2.8155E+00	2.8256E+00	2.8965E+00	2.8946E+00
35	2.5157E+00	0.0000E+00	2.8147E+00	2.8247E+00	2.8957E+00	2.8944E+00
36	2.5153E+00	0.0000E+00	2.8143E+00	2.8243E+00	2.8954E+00	2.8943E+00
37	2.5151E+00	0.0000E+00	2.8141E+00	2.8241E+00	2.8951E+00	2.8866E+00
38	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8619E+00
39	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8699E+00
40	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8690E+00
41	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8590E+00
42	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8571E+00
43	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8776E+00
44	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8780E+00
45	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8664E+00
46	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8606E+00
47	2.5150E+00	0.0000E+00	2.8140E+00	2.8240E+00	2.8950E+00	2.8761E+00

TAB. E.1 Continued.

Response Functions Included with BUGENDF70.BOLIB.
Total Neutrons per Fission (ν) Collapsed Using 1/4 T PV Weighting.

Grp	Pu-240	Pu-241	Pu-242	Pu-243	Pu-244	Am-241
1	5.1221E+00	5.2376E+00	5.2173E+00	5.9553E+00	5.6955E+00	5.3446E+00
2	4.7787E+00	4.8933E+00	4.8610E+00	5.5316E+00	5.2786E+00	5.0574E+00
3	4.4472E+00	4.5556E+00	4.5166E+00	5.1118E+00	4.8799E+00	4.7649E+00
4	4.2006E+00	4.3044E+00	4.2600E+00	4.7845E+00	4.5776E+00	4.5157E+00
5	4.0070E+00	4.1072E+00	4.0589E+00	4.5291E+00	4.3441E+00	4.3158E+00
6	3.8129E+00	3.9118E+00	3.8586E+00	4.2706E+00	4.1097E+00	4.1187E+00
7	3.6349E+00	3.7316E+00	3.6740E+00	4.0312E+00	3.8950E+00	3.9339E+00
8	3.4546E+00	3.5502E+00	3.4860E+00	3.8081E+00	3.6723E+00	3.7495E+00
9	3.3057E+00	3.3961E+00	3.3312E+00	3.6258E+00	3.4926E+00	3.5965E+00
10	3.2382E+00	3.3255E+00	3.2611E+00	3.5431E+00	3.4107E+00	3.5268E+00
11	3.1974E+00	3.2829E+00	3.2189E+00	3.4931E+00	3.3614E+00	3.4850E+00
12	3.1698E+00	3.2541E+00	3.1904E+00	3.4594E+00	3.3281E+00	3.4568E+00
13	3.1610E+00	3.2449E+00	3.1812E+00	3.4487E+00	3.3174E+00	3.4477E+00
14	3.1506E+00	3.2340E+00	3.1705E+00	3.4359E+00	3.3048E+00	3.4371E+00
15	3.1183E+00	3.2001E+00	3.1369E+00	3.3963E+00	3.2656E+00	3.4039E+00
16	3.0730E+00	3.1525E+00	3.0898E+00	3.3405E+00	3.2105E+00	3.3572E+00
17	3.0306E+00	3.1083E+00	3.0458E+00	3.2888E+00	3.1591E+00	3.3138E+00
18	2.9804E+00	3.0561E+00	2.9936E+00	3.2283E+00	3.0990E+00	3.2638E+00
19	2.9414E+00	3.0145E+00	2.9539E+00	3.1806E+00	3.0520E+00	3.2229E+00
20	2.9219E+00	2.9944E+00	2.9333E+00	3.1556E+00	3.0281E+00	3.2025E+00
21	2.9047E+00	2.9762E+00	2.9159E+00	3.1344E+00	3.0082E+00	3.1858E+00
22	2.8881E+00	2.9584E+00	2.8983E+00	3.1133E+00	2.9873E+00	3.1679E+00
23	2.8698E+00	2.9457E+00	2.8792E+00	3.0896E+00	2.9651E+00	3.1484E+00
24	2.8536E+00	2.9453E+00	2.8626E+00	3.0717E+00	2.9455E+00	3.1323E+00
25	2.8400E+00	2.9453E+00	2.8490E+00	3.0544E+00	2.9291E+00	3.1183E+00
26	2.8254E+00	2.9453E+00	2.8337E+00	3.0372E+00	2.9112E+00	3.1034E+00
27	2.8158E+00	2.9453E+00	2.8235E+00	3.0257E+00	2.8997E+00	3.0936E+00
28	2.8112E+00	2.9453E+00	2.8185E+00	3.0200E+00	2.8938E+00	3.0887E+00
29	2.8085E+00	2.9453E+00	2.8157E+00	3.0168E+00	2.8907E+00	3.0859E+00
30	2.8072E+00	2.9453E+00	2.8144E+00	3.0152E+00	2.8891E+00	3.0846E+00
31	2.8068E+00	2.9453E+00	2.8140E+00	3.0147E+00	2.8886E+00	3.0842E+00
32	2.8065E+00	2.9453E+00	2.8136E+00	3.0143E+00	2.8882E+00	3.0839E+00
33	2.8058E+00	2.9453E+00	2.8129E+00	3.0134E+00	2.8874E+00	3.0832E+00
34	2.8047E+00	2.9453E+00	2.8117E+00	3.0121E+00	2.8861E+00	3.0820E+00
35	2.8038E+00	2.9453E+00	2.8108E+00	3.0109E+00	2.8850E+00	3.0810E+00
36	2.8033E+00	2.9453E+00	2.8104E+00	3.0104E+00	2.8845E+00	3.0806E+00
37	2.8032E+00	2.9453E+00	2.8101E+00	3.0102E+00	2.8842E+00	3.0804E+00
38	2.8030E+00	2.9453E+00	2.8100E+00	3.0101E+00	2.8841E+00	3.0803E+00
39	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
40	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
41	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
42	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
43	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
44	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
45	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
46	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00
47	2.8030E+00	2.9453E+00	2.8100E+00	3.0100E+00	0.0000E+00	3.0803E+00

TAB. E.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
Total Neutrons per Fission (ν) Collapsed Using 1/4 T PV Weighting.

Grp	Am-242	Am-242m	Am-243	Cm-241	Cm-242	Cm-243
1	5.3023E+00	5.3001E+00	5.3014E+00	5.3032E+00	6.0670E+00	5.2893E+00
2	5.0468E+00	5.0450E+00	5.0063E+00	4.9273E+00	5.6774E+00	5.0434E+00
3	4.7840E+00	4.7850E+00	4.7235E+00	4.5674E+00	5.3045E+00	4.7990E+00
4	4.5784E+00	4.5797E+00	4.5104E+00	4.2999E+00	5.0237E+00	4.6125E+00
5	4.4153E+00	4.4166E+00	4.3426E+00	4.0873E+00	4.8051E+00	4.4620E+00
6	4.2392E+00	4.2390E+00	4.1745E+00	3.8673E+00	4.5880E+00	4.3033E+00
7	4.0755E+00	4.0739E+00	4.0183E+00	3.6755E+00	4.3817E+00	4.1486E+00
8	3.9045E+00	3.9038E+00	3.8584E+00	3.4836E+00	4.1825E+00	3.9952E+00
9	3.7612E+00	3.7601E+00	3.7259E+00	3.3168E+00	4.0104E+00	3.8666E+00
10	3.6954E+00	3.6942E+00	3.6654E+00	3.2429E+00	3.9324E+00	3.8082E+00
11	3.6558E+00	3.6545E+00	3.6289E+00	3.1984E+00	3.8862E+00	3.7733E+00
12	3.6290E+00	3.6277E+00	3.6042E+00	3.1686E+00	3.8551E+00	3.7498E+00
13	3.6204E+00	3.6191E+00	3.5963E+00	3.1590E+00	3.8451E+00	3.7422E+00
14	3.6103E+00	3.6090E+00	3.5870E+00	3.1476E+00	3.8334E+00	3.7333E+00
15	3.5788E+00	3.5776E+00	3.5580E+00	3.1122E+00	3.7969E+00	3.7057E+00
16	3.5345E+00	3.5333E+00	3.5171E+00	3.0629E+00	3.7461E+00	3.6663E+00
17	3.4932E+00	3.4920E+00	3.4790E+00	3.0166E+00	3.6985E+00	3.6295E+00
18	3.4445E+00	3.4430E+00	3.4347E+00	2.9615E+00	3.6429E+00	3.5860E+00
19	3.4060E+00	3.4047E+00	3.3997E+00	2.9191E+00	3.5981E+00	3.5518E+00
20	3.3874E+00	3.3860E+00	3.3812E+00	2.8977E+00	3.5748E+00	3.5351E+00
21	3.3704E+00	3.3690E+00	3.3659E+00	2.8785E+00	3.5561E+00	3.5200E+00
22	3.3539E+00	3.3525E+00	3.3502E+00	2.8601E+00	3.5364E+00	3.5053E+00
23	3.3351E+00	3.3337E+00	3.3335E+00	2.8388E+00	3.5154E+00	3.4886E+00
24	3.3208E+00	3.3195E+00	3.3188E+00	2.8229E+00	3.4973E+00	3.4757E+00
25	3.3071E+00	3.3058E+00	3.3062E+00	2.8081E+00	3.4822E+00	3.4634E+00
26	3.2935E+00	3.2922E+00	3.2932E+00	2.7923E+00	3.4654E+00	3.4512E+00
27	3.2843E+00	3.2830E+00	3.2846E+00	2.7821E+00	3.4547E+00	3.4432E+00
28	3.2797E+00	3.2784E+00	3.2803E+00	2.7769E+00	3.4492E+00	3.4391E+00
29	3.2772E+00	3.2759E+00	3.2778E+00	2.7740E+00	3.4463E+00	3.4368E+00
30	3.2759E+00	3.2746E+00	3.2767E+00	2.7726E+00	3.4448E+00	3.4357E+00
31	3.2755E+00	3.2742E+00	3.2763E+00	2.7722E+00	3.4443E+00	3.4353E+00
32	3.2752E+00	3.2739E+00	3.2760E+00	2.7718E+00	3.4440E+00	3.4350E+00
33	3.2745E+00	3.2732E+00	3.2754E+00	2.7711E+00	3.4432E+00	3.4345E+00
34	3.2734E+00	3.2721E+00	3.2744E+00	2.7698E+00	3.4421E+00	3.4335E+00
35	3.2725E+00	3.2712E+00	3.2735E+00	2.7688E+00	0.0000E+00	3.4327E+00
36	3.2721E+00	3.2708E+00	3.2731E+00	2.7684E+00	0.0000E+00	3.4323E+00
37	3.2719E+00	3.2706E+00	3.2729E+00	2.7681E+00	0.0000E+00	3.4321E+00
38	3.2718E+00	3.2705E+00	3.2729E+00	2.7681E+00	3.4400E+00	3.4320E+00
39	3.2718E+00	3.2705E+00	3.2729E+00	2.7680E+00	3.4400E+00	3.4320E+00
40	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00
41	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00
42	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00
43	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00
44	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00
45	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00
46	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00
47	3.2718E+00	3.2705E+00	3.2728E+00	2.7680E+00	3.4400E+00	3.4320E+00

TAB. E.1 Continued.

Response Functions Included with BUGENDF70.BOLIB.
Total Neutrons per Fission (ν) Collapsed Using 1/4 T PV Weighting.

Grp	Cm-244	Cm-245	Cm-246	Cm-247	Cm-248
1	6.0542E+00	5.3759E+00	5.4756E+00	6.8839E+00	6.6667E+00
2	5.6416E+00	5.1373E+00	5.2340E+00	6.4270E+00	6.1956E+00
3	5.2371E+00	4.8976E+00	4.9865E+00	5.9915E+00	5.7447E+00
4	4.9384E+00	4.7149E+00	4.8013E+00	5.6600E+00	5.4063E+00
5	4.7047E+00	4.5675E+00	4.6557E+00	5.4033E+00	5.1405E+00
6	4.4666E+00	4.4228E+00	4.5096E+00	5.1483E+00	4.8769E+00
7	4.2516E+00	4.2834E+00	4.3547E+00	4.9092E+00	4.6286E+00
8	4.0344E+00	4.1277E+00	4.1947E+00	4.6698E+00	4.3838E+00
9	3.8533E+00	3.9981E+00	4.0635E+00	4.4716E+00	4.1784E+00
10	3.7712E+00	3.9390E+00	4.0037E+00	4.3816E+00	4.0855E+00
11	3.7218E+00	3.9031E+00	3.9675E+00	4.3274E+00	4.0297E+00
12	3.6885E+00	3.8788E+00	3.9430E+00	4.2909E+00	3.9920E+00
13	3.6778E+00	3.8710E+00	3.9351E+00	4.2792E+00	3.9799E+00
14	3.6652E+00	3.8619E+00	3.9259E+00	4.2654E+00	3.9657E+00
15	3.6263E+00	3.8333E+00	3.8973E+00	4.2222E+00	3.9216E+00
16	3.5710E+00	3.7931E+00	3.8565E+00	4.1621E+00	3.8592E+00
17	3.5194E+00	3.7561E+00	3.8184E+00	4.1060E+00	3.8012E+00
18	3.4586E+00	3.7121E+00	3.7744E+00	4.0395E+00	3.7328E+00
19	3.4115E+00	3.6770E+00	3.7392E+00	3.9866E+00	3.6798E+00
20	3.3882E+00	3.6603E+00	3.7214E+00	3.9611E+00	3.6531E+00
21	3.3679E+00	3.6450E+00	3.7064E+00	3.9379E+00	3.6303E+00
22	3.3476E+00	3.6302E+00	3.6909E+00	3.9154E+00	3.6073E+00
23	3.3253E+00	3.6133E+00	3.6742E+00	3.8897E+00	3.5802E+00
24	3.3058E+00	3.6005E+00	3.6599E+00	3.8703E+00	3.5596E+00
25	3.2896E+00	3.5964E+00	3.6477E+00	3.8515E+00	3.5401E+00
26	3.2717E+00	3.5964E+00	3.6345E+00	3.8330E+00	3.5209E+00
27	3.2599E+00	3.5964E+00	3.6257E+00	3.8205E+00	3.5081E+00
28	3.2542E+00	3.5964E+00	3.6214E+00	3.8142E+00	3.5013E+00
29	3.2510E+00	3.5964E+00	3.6191E+00	3.8108E+00	3.4975E+00
30	3.2495E+00	3.5964E+00	3.6180E+00	3.8090E+00	3.4958E+00
31	3.2490E+00	3.5964E+00	3.6176E+00	3.8085E+00	3.4952E+00
32	3.2486E+00	3.5964E+00	3.6173E+00	3.8081E+00	3.4948E+00
33	3.2477E+00	3.5964E+00	3.6167E+00	3.8071E+00	3.4938E+00
34	3.2464E+00	3.5964E+00	3.6157E+00	3.8056E+00	3.4924E+00
35	3.2452E+00	3.5964E+00	3.6148E+00	3.8044E+00	3.4911E+00
36	3.2448E+00	3.5964E+00	3.6145E+00	3.8039E+00	3.4905E+00
37	3.2445E+00	3.5964E+00	3.6143E+00	3.8036E+00	3.4902E+00
38	3.2444E+00	3.5964E+00	3.6142E+00	3.8035E+00	3.4901E+00
39	3.2444E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
40	3.2444E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
41	3.2444E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
42	3.2443E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
43	3.2443E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
44	3.2443E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
45	3.2443E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
46	3.2443E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00
47	3.2443E+00	3.5964E+00	3.6142E+00	3.8034E+00	3.4900E+00

APPENDIX F

Flat Weighting IRDF-2002 Neutron Dosimeter Cross Sections

TAB. F.1

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using Flat Weighting.

Grp	Li-6 (n,t)	B-10 (n,α)	F-19 (n,2n)	Na-23 (n,γ)	Na-23 (n,2n)	Mg-24 (n,p)
1	2.2777E-02	4.2100E-02	5.9196E-02	2.3083E-04	5.8229E-02	1.5739E-01
2	2.7637E-02	4.2646E-02	2.1943E-02	2.0023E-04	5.4069E-03	1.9926E-01
3	3.4007E-02	4.4324E-02	1.3158E-03	1.7993E-04	0.	1.6662E-01
4	4.1358E-02	5.7456E-02	0.	1.7401E-04	0.	1.2969E-01
5	4.9173E-02	8.0159E-02	0.	1.7011E-04	0.	1.0097E-01
6	5.9777E-02	1.1258E-01	0.	1.6679E-04	0.	3.1786E-02
7	7.5092E-02	1.2185E-01	0.	1.6336E-04	0.	4.7656E-04
8	1.0073E-01	2.6213E-01	0.	1.6076E-04	0.	6.8276E-07
9	1.4154E-01	3.0472E-01	0.	1.6786E-04	0.	0.
10	1.7205E-01	3.6904E-01	0.	1.7468E-04	0.	0.
11	1.8800E-01	3.1921E-01	0.	1.7930E-04	0.	0.
12	1.9791E-01	2.8462E-01	0.	1.8274E-04	0.	0.
13	2.0101E-01	2.9290E-01	0.	1.8396E-04	0.	0.
14	2.0436E-01	3.0917E-01	0.	1.8533E-04	0.	0.
15	2.1340E-01	4.1153E-01	0.	1.9015E-04	0.	0.
16	2.1588E-01	5.0294E-01	0.	1.9772E-04	0.	0.
17	2.1292E-01	3.1006E-01	0.	2.0694E-04	0.	0.
18	2.2018E-01	2.1853E-01	0.	2.2073E-04	0.	0.
19	2.3861E-01	2.5393E-01	0.	2.4004E-04	0.	0.
20	2.5657E-01	3.2093E-01	0.	2.8924E-04	0.	0.
21	2.8263E-01	4.3022E-01	0.	3.4130E-04	0.	0.
22	3.3812E-01	6.6644E-01	0.	3.1344E-04	0.	0.
23	4.9337E-01	8.4377E-01	0.	4.9414E-04	0.	0.
24	1.0095E+00	9.5106E-01	0.	5.8311E-04	0.	0.
25	2.4365E+00	1.2644E+00	0.	8.4542E-04	0.	0.
26	9.2228E-01	1.6948E+00	0.	1.0148E-03	0.	0.
27	6.5355E-01	2.0949E+00	0.	1.8310E-05	0.	0.
28	7.1558E-01	2.6015E+00	0.	1.4479E-03	0.	0.
29	8.2600E-01	3.1190E+00	0.	4.5664E-03	0.	0.
30	9.0927E-01	3.4791E+00	0.	2.1505E-05	0.	0.
31	9.6789E-01	3.7255E+00	0.	2.1190E-05	0.	0.
32	1.0069E+00	3.8902E+00	0.	2.2602E-05	0.	0.
33	1.1208E+00	4.3656E+00	0.	3.2116E-05	0.	0.
34	1.4508E+00	5.7276E+00	0.	4.0597E-04	0.	0.
35	2.0992E+00	8.3894E+00	0.	4.2696E-03	0.	0.
36	3.0519E+00	1.2284E+01	0.	1.0200E-01	0.	0.
37	4.8832E+00	1.9767E+01	0.	7.0711E-03	0.	0.
38	8.3105E+00	3.3757E+01	0.	6.0097E-03	0.	0.
39	1.2100E+01	4.9226E+01	0.	7.6283E-03	0.	0.
40	1.8501E+01	7.5318E+01	0.	1.0928E-02	0.	0.
41	3.1940E+01	1.3014E+02	0.	1.8305E-02	0.	0.
42	5.4316E+01	2.2146E+02	0.	3.0689E-02	0.	0.
43	8.3018E+01	3.3852E+02	0.	4.6774E-02	0.	0.
44	1.3032E+02	5.3147E+02	0.	7.3235E-02	0.	0.
45	1.8963E+02	7.7341E+02	0.	1.0646E-01	0.	0.
46	3.1212E+02	1.2733E+03	0.	1.7521E-01	0.	0.
47	9.3790E+02	3.8268E+03	0.	5.2630E-01	0.	0.

TAB. F.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using Flat Weighting.

Grp	Al-27 (n,p)	Al-27 (n,α)	P-31 (n,p)	S-32 (n,p)	Sc-45 (n,γ)	Ti-46 (n,2n)
1	5.6858E-02	9.7222E-02	7.6858E-02	1.7639E-01	5.9600E-04	9.9836E-02
2	8.6898E-02	1.2398E-01	1.0397E-01	3.0787E-01	7.5136E-04	2.2688E-03
3	1.0529E-01	1.0678E-01	1.3395E-01	3.8051E-01	8.6724E-04	0.
4	9.8594E-02	7.7279E-02	1.4093E-01	3.3979E-01	8.7225E-04	0.
5	8.2480E-02	4.1475E-02	1.3519E-01	3.2202E-01	7.4391E-04	0.
6	6.2377E-02	1.1792E-02	1.3771E-01	3.1380E-01	5.7438E-04	0.
7	3.8514E-02	4.6528E-04	1.2798E-01	2.4629E-01	5.6169E-04	0.
8	1.1580E-02	2.6020E-06	9.9590E-02	2.7592E-01	8.8165E-04	0.
9	3.5831E-03	1.5730E-07	6.6977E-02	1.9108E-01	1.3874E-03	0.
10	5.0348E-04	0.	6.1958E-02	1.0053E-01	1.7113E-03	0.
11	7.4050E-05	0.	5.1396E-02	6.8686E-02	1.9530E-03	0.
12	7.1563E-06	0.	3.5904E-02	7.2417E-02	2.1124E-03	0.
13	5.0878E-06	0.	3.2199E-02	6.6472E-02	2.1656E-03	0.
14	4.3455E-06	0.	2.7388E-02	6.1245E-02	2.2249E-03	0.
15	1.9899E-06	0.	1.1947E-02	2.1377E-02	2.4195E-03	0.
16	1.2122E-08	0.	6.6123E-03	4.1502E-03	2.8018E-03	0.
17	0.	0.	1.2130E-03	5.9998E-04	3.2745E-03	0.
18	0.	0.	1.4023E-04	6.0917E-05	4.1266E-03	0.
19	0.	0.	1.2603E-05	1.6785E-06	5.0270E-03	0.
20	0.	0.	2.6438E-06	0.	5.7282E-03	0.
21	0.	0.	1.9832E-08	0.	6.5081E-03	0.
22	0.	0.	0.	0.	7.7226E-03	0.
23	0.	0.	0.	0.	9.5306E-03	0.
24	0.	0.	0.	0.	1.1884E-02	0.
25	0.	0.	0.	0.	1.5827E-02	0.
26	0.	0.	0.	0.	2.3259E-02	0.
27	0.	0.	0.	0.	3.5784E-02	0.
28	0.	0.	0.	0.	4.5936E-02	0.
29	0.	0.	0.	0.	7.2189E-02	0.
30	0.	0.	0.	0.	6.2929E-02	0.
31	0.	0.	0.	0.	2.8741E-02	0.
32	0.	0.	0.	0.	5.1032E-02	0.
33	0.	0.	0.	0.	1.0586E-01	0.
34	0.	0.	0.	0.	1.5537E-01	0.
35	0.	0.	0.	0.	2.3491E-01	0.
36	0.	0.	0.	0.	2.2423E-01	0.
37	0.	0.	0.	0.	5.6155E-02	0.
38	0.	0.	0.	0.	7.8206E-02	0.
39	0.	0.	0.	0.	1.8019E-01	0.
40	0.	0.	0.	0.	3.8603E-01	0.
41	0.	0.	0.	0.	8.1908E-01	0.
42	0.	0.	0.	0.	1.5057E+00	0.
43	0.	0.	0.	0.	2.3568E+00	0.
44	0.	0.	0.	0.	3.7408E+00	0.
45	0.	0.	0.	0.	5.4643E+00	0.
46	0.	0.	0.	0.	9.0139E+00	0.
47	0.	0.	0.	0.	2.7114E+01	0.

TAB. F.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using Flat Weighting.

Grp	Ti-46 (n,p)	Ti-47 (n,x)	Ti-47 (n,p)	Ti-48 (n,x)	Ti-48 (n,p)	Ti-49 (n,x)
1	1.9614E-01	1.0271E-01	9.6117E-02	2.7399E-02	5.5979E-02	1.7497E-02
2	2.8124E-01	1.7610E-02	1.2010E-01	2.5853E-03	5.9345E-02	1.4810E-03
3	2.9804E-01	1.0843E-03	1.3133E-01	6.4089E-05	4.1795E-02	4.2431E-05
4	2.6868E-01	2.9187E-06	1.2432E-01	6.9853E-07	2.4843E-02	5.4493E-09
5	2.3930E-01	4.3596E-09	1.0771E-01	0.	1.5128E-02	0.
6	1.8477E-01	0.	8.8011E-02	0.	5.9033E-03	0.
7	9.7292E-02	0.	7.1204E-02	0.	6.8636E-04	0.
8	4.0127E-02	0.	5.4883E-02	0.	1.6848E-05	0.
9	5.3275E-03	0.	3.9387E-02	0.	2.4724E-07	0.
10	5.7867E-04	0.	3.1401E-02	0.	0.	0.
11	7.3689E-05	0.	2.5982E-02	0.	0.	0.
12	2.1604E-05	0.	2.2131E-02	0.	0.	0.
13	1.7487E-05	0.	2.0778E-02	0.	0.	0.
14	1.2901E-05	0.	1.9271E-02	0.	0.	0.
15	1.8982E-06	0.	1.4534E-02	0.	0.	0.
16	0.	0.	8.9501E-03	0.	0.	0.
17	0.	0.	4.3051E-03	0.	0.	0.
18	0.	0.	2.0440E-03	0.	0.	0.
19	0.	0.	9.4212E-04	0.	0.	0.
20	0.	0.	6.1664E-04	0.	0.	0.
21	0.	0.	5.2585E-05	0.	0.	0.
22	0.	0.	0.	0.	0.	0.
23	0.	0.	0.	0.	0.	0.
24	0.	0.	0.	0.	0.	0.
25	0.	0.	0.	0.	0.	0.
26	0.	0.	0.	0.	0.	0.
27	0.	0.	0.	0.	0.	0.
28	0.	0.	0.	0.	0.	0.
29	0.	0.	0.	0.	0.	0.
30	0.	0.	0.	0.	0.	0.
31	0.	0.	0.	0.	0.	0.
32	0.	0.	0.	0.	0.	0.
33	0.	0.	0.	0.	0.	0.
34	0.	0.	0.	0.	0.	0.
35	0.	0.	0.	0.	0.	0.
36	0.	0.	0.	0.	0.	0.
37	0.	0.	0.	0.	0.	0.
38	0.	0.	0.	0.	0.	0.
39	0.	0.	0.	0.	0.	0.
40	0.	0.	0.	0.	0.	0.
41	0.	0.	0.	0.	0.	0.
42	0.	0.	0.	0.	0.	0.
43	0.	0.	0.	0.	0.	0.
44	0.	0.	0.	0.	0.	0.
45	0.	0.	0.	0.	0.	0.
46	0.	0.	0.	0.	0.	0.
47	0.	0.	0.	0.	0.	0.

TAB. F.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using Flat Weighting.

Grp	V-51 (n,α)	Cr-52 (n,2n)	Mn-55 (n,γ)	Fe-54 (n,2n)	Fe-54 (n,α)	Fe-54 (n,p)
1	1.8507E-02	4.8834E-01	6.3791E-04	3.0419E-02	8.7935E-02	2.3634E-01
2	1.2653E-02	1.2004E-01	6.7022E-04	2.4921E-04	8.1207E-02	3.9357E-01
3	7.0244E-03	0.	6.8110E-04	0.	6.1212E-02	4.6624E-01
4	3.2718E-03	0.	6.9011E-04	0.	4.5032E-02	4.8169E-01
5	9.8080E-04	0.	7.0725E-04	0.	3.3503E-02	4.8240E-01
6	9.2138E-05	0.	7.8681E-04	0.	1.8517E-02	4.7885E-01
7	2.1828E-06	0.	9.4026E-04	0.	5.1740E-03	4.3669E-01
8	2.1185E-07	0.	1.1705E-03	0.	6.1965E-04	3.1544E-01
9	3.3140E-08	0.	1.4238E-03	0.	1.7939E-05	1.9564E-01
10	1.8582E-08	0.	1.6227E-03	0.	1.5512E-06	1.3414E-01
11	1.1997E-08	0.	1.7147E-03	0.	4.1165E-07	7.8808E-02
12	7.6513E-09	0.	1.7435E-03	0.	0.	5.6762E-02
13	6.1991E-09	0.	1.7531E-03	0.	0.	5.1203E-02
14	4.5814E-09	0.	1.7638E-03	0.	0.	4.5047E-02
15	6.8246E-10	0.	1.8013E-03	0.	0.	2.9403E-02
16	0.	0.	1.9008E-03	0.	0.	9.5530E-03
17	0.	0.	2.0756E-03	0.	0.	2.9684E-03
18	0.	0.	2.4662E-03	0.	0.	7.8514E-04
19	0.	0.	3.0418E-03	0.	0.	9.0118E-05
20	0.	0.	3.2800E-03	0.	0.	6.6796E-06
21	0.	0.	3.7084E-03	0.	0.	3.9326E-07
22	0.	0.	4.2165E-03	0.	0.	0.
23	0.	0.	4.9192E-03	0.	0.	0.
24	0.	0.	5.3634E-03	0.	0.	0.
25	0.	0.	6.0842E-03	0.	0.	0.
26	0.	0.	1.0518E-02	0.	0.	0.
27	0.	0.	1.4957E-02	0.	0.	0.
28	0.	0.	1.8140E-02	0.	0.	0.
29	0.	0.	2.6705E-02	0.	0.	0.
30	0.	0.	3.0915E-02	0.	0.	0.
31	0.	0.	3.6360E-02	0.	0.	0.
32	0.	0.	3.4224E-02	0.	0.	0.
33	0.	0.	4.6091E-02	0.	0.	0.
34	0.	0.	6.0186E-02	0.	0.	0.
35	0.	0.	8.3963E-02	0.	0.	0.
36	0.	0.	2.9476E-01	0.	0.	0.
37	0.	0.	6.7619E-01	0.	0.	0.
38	0.	0.	6.3782E+00	0.	0.	0.
39	0.	0.	3.7445E-01	0.	0.	0.
40	0.	0.	3.2645E-01	0.	0.	0.
41	0.	0.	4.8332E-01	0.	0.	0.
42	0.	0.	7.8907E-01	0.	0.	0.
43	0.	0.	1.1929E+00	0.	0.	0.
44	0.	0.	1.8637E+00	0.	0.	0.
45	0.	0.	2.7073E+00	0.	0.	0.
46	0.	0.	4.4516E+00	0.	0.	0.
47	0.	0.	1.3368E+01	0.	0.	0.

TAB. F.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using Flat Weighting.

Grp	Fe-56 (n,p)	Fe-58 (n,γ)	Co-59 (n,2n)	Co-59 (n,α)	Co-59 (n,γ)	Ni-58 (n,2n)
1	9.1210E-02	1.1881E-03	7.9100E-01	2.7901E-02	6.8291E-04	5.2093E-02
2	1.1355E-01	7.2113E-04	5.5635E-01	2.8997E-02	8.5139E-04	9.7220E-03
3	8.9709E-02	2.7648E-04	9.8593E-02	2.0584E-02	6.8329E-04	0.
4	6.4310E-02	2.8020E-04	0.	1.3549E-02	6.3483E-04	0.
5	4.6200E-02	3.7292E-04	0.	7.8868E-03	7.1788E-04	0.
6	2.6202E-02	5.0855E-04	0.	2.8138E-03	9.2107E-04	0.
7	5.9377E-03	6.5369E-04	0.	4.5438E-04	1.1713E-03	0.
8	1.7723E-04	7.8981E-04	0.	1.6565E-05	1.5472E-03	0.
9	3.0477E-09	8.8318E-04	0.	4.8548E-08	2.0428E-03	0.
10	3.7637E-12	9.2559E-04	0.	2.5493E-08	2.3420E-03	0.
11	0.	9.5848E-04	0.	2.2747E-08	2.5339E-03	0.
12	0.	9.8474E-04	0.	2.0937E-08	2.6928E-03	0.
13	0.	9.9370E-04	0.	2.0332E-08	2.7590E-03	0.
14	0.	1.0192E-03	0.	1.9658E-08	2.8327E-03	0.
15	0.	1.2195E-03	0.	1.7520E-08	3.0744E-03	0.
16	0.	1.3257E-03	0.	1.4612E-08	3.5499E-03	0.
17	0.	1.2807E-03	0.	1.1760E-08	4.2977E-03	0.
18	0.	1.3156E-03	0.	8.4888E-09	5.9983E-03	0.
19	0.	2.0414E-03	0.	5.8106E-09	6.4531E-03	0.
20	0.	3.3972E-03	0.	4.5038E-09	6.5396E-03	0.
21	0.	3.2644E-03	0.	3.4338E-09	6.7244E-03	0.
22	0.	3.1003E-03	0.	2.2024E-09	7.1213E-03	0.
23	0.	3.1117E-03	0.	9.9910E-10	7.8522E-03	0.
24	0.	3.0661E-03	0.	8.5105E-11	8.9120E-03	0.
25	0.	4.2618E-03	0.	0.	1.0951E-02	0.
26	0.	6.1805E-03	0.	0.	1.5504E-02	0.
27	0.	7.7548E-03	0.	0.	1.7403E-02	0.
28	0.	9.9371E-03	0.	0.	1.8056E-02	0.
29	0.	9.7221E-03	0.	0.	2.6961E-02	0.
30	0.	9.0995E-03	0.	0.	3.0404E-02	0.
31	0.	5.5775E-03	0.	0.	7.4254E-02	0.
32	0.	8.0596E-04	0.	0.	2.5570E-02	0.
33	0.	8.7272E-03	0.	0.	4.2568E-02	0.
34	0.	6.1231E-02	0.	0.	6.8809E-02	0.
35	0.	2.8327E-02	0.	0.	2.1591E-01	0.
36	0.	4.0568E-03	0.	0.	7.8656E-02	0.
37	0.	6.7433E-03	0.	0.	4.4682E-02	0.
38	0.	9.7292E-01	0.	0.	2.4101E-01	0.
39	0.	1.7725E-02	0.	0.	6.5969E+01	0.
40	0.	2.5653E-02	0.	0.	3.3228E+00	0.
41	0.	4.4174E-02	0.	0.	1.7336E+00	0.
42	0.	7.5082E-02	0.	0.	2.3587E+00	0.
43	0.	1.1473E-01	0.	0.	3.4122E+00	0.
44	0.	1.8010E-01	0.	0.	5.2296E+00	0.
45	0.	2.6205E-01	0.	0.	7.5457E+00	0.
46	0.	4.3132E-01	0.	0.	1.2362E+01	0.
47	0.	1.2959E+00	0.	0.	3.7052E+01	0.

TAB. F.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using Flat Weighting.

Grp	Ni-58 (n,p)	Ni-60 (n,p)	Cu-63 (n,2n)	Cu-63 (n,γ)	Cu-63 (n,α)	Cu-65 (n,2n)
1	2.2132E-01	1.0911E-01	6.3934E-01	2.5606E-03	3.4460E-02	1.0168E+00
2	4.4874E-01	1.4482E-01	2.9996E-01	2.8157E-03	4.6786E-02	7.5575E-01
3	6.0075E-01	1.1976E-01	1.9265E-02	3.0672E-03	3.8885E-02	2.0553E-01
4	6.3861E-01	8.8754E-02	0.	3.3393E-03	2.6939E-02	0.
5	6.4484E-01	6.5405E-02	0.	3.6082E-03	1.8860E-02	0.
6	6.2243E-01	3.8913E-02	0.	3.9369E-03	1.0688E-02	0.
7	5.4560E-01	1.5631E-02	0.	4.2627E-03	3.6481E-03	0.
8	4.0492E-01	2.3113E-03	0.	4.8066E-03	6.1452E-04	0.
9	2.5204E-01	3.1712E-05	0.	5.3928E-03	4.2940E-05	0.
10	1.6817E-01	5.4326E-08	0.	5.7183E-03	4.6929E-06	0.
11	1.2162E-01	3.1422E-10	0.	5.9639E-03	1.3682E-06	0.
12	9.3475E-02	0.	0.	6.1260E-03	7.4165E-07	0.
13	8.4827E-02	0.	0.	6.1801E-03	4.7235E-07	0.
14	7.5697E-02	0.	0.	6.2404E-03	1.7922E-07	0.
15	5.0849E-02	0.	0.	6.4734E-03	0.	0.
16	2.5952E-02	0.	0.	7.5663E-03	0.	0.
17	1.1677E-02	0.	0.	9.1275E-03	0.	0.
18	3.7988E-03	0.	0.	1.1898E-02	0.	0.
19	4.8761E-04	0.	0.	1.3722E-02	0.	0.
20	1.5596E-04	0.	0.	1.3875E-02	0.	0.
21	4.1887E-05	0.	0.	1.3948E-02	0.	0.
22	3.3813E-06	0.	0.	1.4476E-02	0.	0.
23	2.9075E-07	0.	0.	1.6372E-02	0.	0.
24	0.	0.	0.	1.9011E-02	0.	0.
25	0.	0.	0.	2.3563E-02	0.	0.
26	0.	0.	0.	2.8104E-02	0.	0.
27	0.	0.	0.	3.2191E-02	0.	0.
28	0.	0.	0.	3.8833E-02	0.	0.
29	0.	0.	0.	5.9392E-02	0.	0.
30	0.	0.	0.	5.7082E-02	0.	0.
31	0.	0.	0.	8.8324E-02	0.	0.
32	0.	0.	0.	9.8525E-02	0.	0.
33	0.	0.	0.	1.1393E-01	0.	0.
34	0.	0.	0.	1.7574E-01	0.	0.
35	0.	0.	0.	2.8259E-01	0.	0.
36	0.	0.	0.	5.1632E-01	0.	0.
37	0.	0.	0.	1.1570E+00	0.	0.
38	0.	0.	0.	2.8624E-02	0.	0.
39	0.	0.	0.	3.3284E-02	0.	0.
40	0.	0.	0.	6.5539E-02	0.	0.
41	0.	0.	0.	1.3557E-01	0.	0.
42	0.	0.	0.	2.4784E-01	0.	0.
43	0.	0.	0.	3.8746E-01	0.	0.
44	0.	0.	0.	6.1465E-01	0.	0.
45	0.	0.	0.	8.9781E-01	0.	0.
46	0.	0.	0.	1.4809E+00	0.	0.
47	0.	0.	0.	4.4539E+00	0.	0.

TAB. F.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using Flat Weighting.

Grp	Zn-64 (n,p)	As-75 (n,2n)	Y-89 (n,2n)	Zr-90 (n,2n)	Nb-93 (n,2n)	Nb-93 (n,n')
1	1.4070E-01	1.0991E+00	1.1083E+00	9.5752E-01	4.4092E-01	3.2562E-02
2	2.1830E-01	7.7292E-01	5.3617E-01	3.1976E-01	4.4078E-01	4.7167E-02
3	2.5873E-01	1.6154E-01	2.0036E-02	3.2443E-04	2.8267E-01	1.0291E-01
4	2.3088E-01	0.	0.	0.	2.9299E-02	2.0851E-01
5	2.1633E-01	0.	0.	0.	0.	2.4422E-01
6	2.0282E-01	0.	0.	0.	0.	2.4804E-01
7	1.8545E-01	0.	0.	0.	0.	2.5457E-01
8	1.5281E-01	0.	0.	0.	0.	2.6349E-01
9	1.1147E-01	0.	0.	0.	0.	2.6104E-01
10	6.8891E-02	0.	0.	0.	0.	2.4935E-01
11	3.8620E-02	0.	0.	0.	0.	2.3699E-01
12	2.4187E-02	0.	0.	0.	0.	2.2629E-01
13	2.0267E-02	0.	0.	0.	0.	2.2215E-01
14	1.7440E-02	0.	0.	0.	0.	2.1720E-01
15	9.5983E-03	0.	0.	0.	0.	1.9901E-01
16	3.3376E-03	0.	0.	0.	0.	1.6909E-01
17	1.0695E-03	0.	0.	0.	0.	1.3378E-01
18	2.3909E-04	0.	0.	0.	0.	8.8615E-02
19	9.2970E-05	0.	0.	0.	0.	5.1959E-02
20	6.3631E-05	0.	0.	0.	0.	3.6094E-02
21	3.9611E-05	0.	0.	0.	0.	2.5332E-02
22	1.1969E-05	0.	0.	0.	0.	1.5631E-02
23	0.	0.	0.	0.	0.	9.3805E-03
24	0.	0.	0.	0.	0.	6.0051E-03
25	0.	0.	0.	0.	0.	3.7445E-03
26	0.	0.	0.	0.	0.	1.7897E-03
27	0.	0.	0.	0.	0.	7.2459E-04
28	0.	0.	0.	0.	0.	2.5826E-04
29	0.	0.	0.	0.	0.	5.7500E-05
30	0.	0.	0.	0.	0.	1.8645E-06
31	0.	0.	0.	0.	0.	0.
32	0.	0.	0.	0.	0.	0.
33	0.	0.	0.	0.	0.	0.
34	0.	0.	0.	0.	0.	0.
35	0.	0.	0.	0.	0.	0.
36	0.	0.	0.	0.	0.	0.
37	0.	0.	0.	0.	0.	0.
38	0.	0.	0.	0.	0.	0.
39	0.	0.	0.	0.	0.	0.
40	0.	0.	0.	0.	0.	0.
41	0.	0.	0.	0.	0.	0.
42	0.	0.	0.	0.	0.	0.
43	0.	0.	0.	0.	0.	0.
44	0.	0.	0.	0.	0.	0.
45	0.	0.	0.	0.	0.	0.
46	0.	0.	0.	0.	0.	0.
47	0.	0.	0.	0.	0.	0.

TAB. F.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using Flat Weighting.

Grp	Nb-93 (n,γ)	Rh-103 (n,n')	Ag-109 (n,γ)	In-115 (n,2n)	In-115 (n,n')	In-115 (n,γ)
1	4.7162E-04	2.3466E-01	1.6223E-04	1.2949E+00	5.8133E-02	8.7776E-04
2	6.0785E-04	3.8742E-01	2.1051E-04	1.1435E+00	9.1680E-02	9.3900E-04
3	7.3359E-04	8.1816E-01	2.4636E-04	5.8357E-01	2.0239E-01	9.9789E-04
4	9.7412E-04	1.2190E+00	2.4325E-04	2.6919E-02	2.7381E-01	1.4314E-03
5	1.3492E-03	1.2753E+00	1.9587E-04	0.	3.0859E-01	2.2530E-03
6	2.0580E-03	1.2305E+00	1.4467E-04	0.	3.3417E-01	4.0781E-03
7	2.8880E-03	1.1732E+00	1.9254E-04	0.	3.3897E-01	9.1806E-03
8	4.0177E-03	1.1091E+00	7.7253E-04	0.	3.2538E-01	2.0615E-02
9	5.6093E-03	1.0370E+00	2.2547E-03	0.	3.3341E-01	4.0975E-02
10	6.8261E-03	9.8735E-01	3.5534E-03	0.	3.5259E-01	6.0726E-02
11	7.9205E-03	9.5185E-01	4.1434E-03	0.	3.5385E-01	7.6946E-02
12	8.7555E-03	9.2536E-01	4.6208E-03	0.	3.4311E-01	9.0786E-02
13	9.0803E-03	9.1570E-01	4.8017E-03	0.	3.3654E-01	9.5144E-02
14	9.4421E-03	9.0494E-01	5.0229E-03	0.	3.2759E-01	9.9665E-02
15	1.0612E-02	8.6764E-01	5.8339E-03	0.	2.9012E-01	1.1657E-01
16	1.2902E-02	8.0955E-01	6.8741E-03	0.	2.3022E-01	1.3940E-01
17	1.7251E-02	7.4452E-01	8.0739E-03	0.	1.7096E-01	1.6303E-01
18	2.7326E-02	6.6193E-01	9.9839E-03	0.	1.0336E-01	1.8055E-01
19	4.2767E-02	5.8044E-01	1.2209E-02	0.	4.9401E-02	1.8045E-01
20	4.8247E-02	4.9269E-01	1.3602E-02	0.	2.7891E-02	1.7196E-01
21	5.0055E-02	3.3867E-01	1.5111E-02	0.	1.5489E-02	1.6172E-01
22	5.2287E-02	1.9319E-01	1.7365E-02	0.	6.3893E-03	1.5226E-01
23	5.5640E-02	1.3801E-01	2.0173E-02	0.	2.2558E-03	1.4532E-01
24	5.9916E-02	1.0374E-01	2.3171E-02	0.	1.6310E-04	1.5180E-01
25	6.8399E-02	6.6383E-02	2.7195E-02	0.	0.	1.8203E-01
26	9.1615E-02	2.3291E-02	3.3116E-02	0.	0.	2.4572E-01
27	1.2616E-01	5.8219E-03	4.0653E-02	0.	0.	3.2914E-01
28	1.8389E-01	1.6581E-03	5.0587E-02	0.	0.	4.4074E-01
29	2.4509E-01	3.4572E-06	5.6106E-02	0.	0.	5.4395E-01
30	2.7771E-01	0.	5.8614E-02	0.	0.	6.0985E-01
31	3.1736E-01	0.	6.0234E-02	0.	0.	6.4757E-01
32	3.4834E-01	0.	6.1263E-02	0.	0.	6.6554E-01
33	3.7587E-01	0.	6.3627E-02	0.	0.	7.0912E-01
34	5.1687E-01	0.	6.8907E-02	0.	0.	8.4414E-01
35	8.2688E-01	0.	7.5980E-02	0.	0.	1.0312E+00
36	1.2772E+00	0.	1.2226E-01	0.	0.	1.2987E+00
37	2.2973E+00	0.	2.0246E-01	0.	0.	1.9588E+00
38	2.1939E+00	0.	3.3396E-01	0.	0.	4.9564E+00
39	3.5527E+00	0.	8.7730E-01	0.	0.	5.8668E+00
40	2.2849E-01	0.	1.5709E+00	0.	0.	6.6038E+00
41	2.9131E-01	0.	1.2705E+00	0.	0.	6.7612E+00
42	7.9497E-02	0.	4.7634E+01	0.	0.	5.4802E+01
43	1.1450E-01	0.	1.6877E+01	0.	0.	6.6343E+01
44	1.7187E-01	0.	1.0616E+00	0.	0.	3.5224E+03
45	2.4359E-01	0.	1.0926E+00	0.	0.	1.0682E+02
46	3.9189E-01	0.	1.5281E+00	0.	0.	7.6861E+01
47	1.1518E+00	0.	4.2291E+00	0.	0.	1.6806E+02

TAB. F.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using Flat Weighting.

Grp	I-127 (n,2n)	La-139 (n,γ)	Pr-141 (n,2n)	Tm-169 (n,2n)	Ta-181 (n,γ)	W-186 (n,γ)
1	1.6707E+00	7.9402E-04	1.7453E+00	1.9733E+00	1.0238E-07	7.9663E-04
2	1.5567E+00	1.1238E-03	1.4913E+00	1.9708E+00	3.2578E-07	1.2875E-03
3	8.5375E-01	1.0134E-03	7.5478E-01	1.5277E+00	1.5792E-06	1.4217E-03
4	2.7369E-02	9.4235E-04	1.7252E-02	7.1588E-01	9.5174E-06	1.2682E-03
5	0.	1.1494E-03	0.	2.3608E-02	4.4482E-05	1.2213E-03
6	0.	1.4962E-03	0.	0.	2.2720E-04	1.6815E-03
7	0.	1.9701E-03	0.	0.	1.4548E-03	3.2710E-03
8	0.	2.7281E-03	0.	0.	8.0875E-03	7.2370E-03
9	0.	3.7681E-03	0.	0.	2.0702E-02	1.3350E-02
10	0.	4.5494E-03	0.	0.	3.1347E-02	1.8216E-02
11	0.	5.1280E-03	0.	0.	3.8586E-02	2.1607E-02
12	0.	5.5675E-03	0.	0.	4.4057E-02	2.3850E-02
13	0.	5.7284E-03	0.	0.	4.5912E-02	2.4565E-02
14	0.	5.9159E-03	0.	0.	4.8155E-02	2.5336E-02
15	0.	6.5810E-03	0.	0.	5.6623E-02	2.7617E-02
16	0.	7.6884E-03	0.	0.	6.9983E-02	3.0440E-02
17	0.	9.0798E-03	0.	0.	8.6426E-02	3.3112E-02
18	0.	8.5682E-03	0.	0.	1.0029E-01	3.7581E-02
19	0.	7.1979E-03	0.	0.	1.0507E-01	4.4506E-02
20	0.	6.8375E-03	0.	0.	1.0489E-01	4.7118E-02
21	0.	6.6447E-03	0.	0.	1.1096E-01	4.6564E-02
22	0.	6.5630E-03	0.	0.	1.2389E-01	4.5509E-02
23	0.	6.7837E-03	0.	0.	1.4673E-01	4.5700E-02
24	0.	7.3760E-03	0.	0.	1.6849E-01	4.7590E-02
25	0.	8.9141E-03	0.	0.	2.0995E-01	5.3061E-02
26	0.	1.3604E-02	0.	0.	3.0057E-01	7.9406E-02
27	0.	1.9118E-02	0.	0.	3.9797E-01	1.2962E-01
28	0.	2.7679E-02	0.	0.	5.1385E-01	1.4331E-01
29	0.	3.6655E-02	0.	0.	6.4121E-01	1.6093E-01
30	0.	4.2956E-02	0.	0.	7.3595E-01	1.7622E-01
31	0.	4.7212E-02	0.	0.	7.9960E-01	1.8751E-01
32	0.	5.0033E-02	0.	0.	8.4072E-01	1.9540E-01
33	0.	5.4188E-02	0.	0.	9.6664E-01	2.1901E-01
34	0.	9.2236E-02	0.	0.	1.4008E+00	2.8510E-01
35	0.	1.6280E-01	0.	0.	2.5590E+00	4.2887E-01
36	0.	2.1032E-01	0.	0.	4.3265E+00	7.0396E-01
37	0.	2.5894E-01	0.	0.	9.7754E+00	2.6351E+00
38	0.	1.4704E-02	0.	0.	2.2433E+01	6.8228E+00
39	0.	1.0111E-02	0.	0.	3.2131E+01	4.3728E+00
40	0.	8.8675E+00	0.	0.	5.3243E+01	2.0938E-01
41	0.	1.5561E-01	0.	0.	8.6537E+01	3.1297E+02
42	0.	3.9491E-01	0.	0.	1.1564E+02	6.3830E+00
43	0.	7.0380E-01	0.	0.	5.0859E+02	4.8928E+00
44	0.	1.2003E+00	0.	0.	4.4972E+00	6.0835E+00
45	0.	2.1899E+00	0.	0.	4.8997E+00	8.2294E+00
46	0.	2.9782E+00	0.	0.	7.2160E+00	1.3031E+01
47	0.	8.9854E+00	0.	0.	2.0645E+01	3.8377E+01

TAB. F.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using Flat Weighting.

Grp	Au-197 (n,2n)	Au-197 (n,γ)	Hg-199 (n,n')	Pb-204 (n,n')	Th-232 (n,γ)	Th-232 (n,f)
1	2.0688E+00	6.8137E-04	1.0657E-01	5.3647E-02	1.4706E-03	4.3231E-01
2	2.0873E+00	1.6063E-03	1.9565E-01	9.8759E-02	1.6007E-03	3.2705E-01
3	1.6162E+00	1.6769E-03	4.2169E-01	1.9413E-01	1.8616E-03	3.0823E-01
4	5.4021E-01	1.1792E-03	5.8687E-01	2.6870E-01	2.6268E-03	3.2376E-01
5	2.1309E-02	9.1672E-04	6.3344E-01	2.3533E-01	3.9684E-03	3.6014E-01
6	0.	2.9909E-03	6.3733E-01	1.7162E-01	6.4529E-03	3.0504E-01
7	0.	6.0620E-03	6.1096E-01	1.1816E-01	1.0428E-02	1.4368E-01
8	0.	1.0673E-02	5.6775E-01	7.3281E-02	1.6669E-02	1.4236E-01
9	0.	1.9494E-02	4.8876E-01	3.5480E-02	2.5856E-02	1.3971E-01
10	0.	2.7032E-02	4.3154E-01	1.6057E-02	3.3157E-02	1.2707E-01
11	0.	3.1752E-02	3.8426E-01	6.1301E-03	3.9749E-02	1.2048E-01
12	0.	3.5656E-02	3.5319E-01	6.3880E-04	4.5712E-02	1.1508E-01
13	0.	3.7624E-02	3.4477E-01	1.8718E-04	4.8358E-02	1.1585E-01
14	0.	4.0070E-02	3.3530E-01	7.9038E-05	5.1306E-02	1.2606E-01
15	0.	4.9391E-02	2.9518E-01	7.8500E-07	6.1027E-02	1.2270E-01
16	0.	6.0152E-02	2.3594E-01	0.	8.0708E-02	9.1357E-02
17	0.	6.7842E-02	1.6247E-01	0.	9.8665E-02	8.1384E-02
18	0.	7.5077E-02	9.1685E-02	0.	1.1982E-01	8.6871E-03
19	0.	8.4031E-02	5.2525E-02	0.	1.4626E-01	7.1286E-04
20	0.	9.0899E-02	3.5628E-02	0.	1.6068E-01	1.2621E-04
21	0.	9.8907E-02	2.0639E-02	0.	1.6465E-01	3.4520E-05
22	0.	1.2061E-01	3.6573E-03	0.	1.5792E-01	5.8663E-06
23	0.	1.5148E-01	0.	0.	1.4508E-01	0.
24	0.	1.8519E-01	0.	0.	1.4401E-01	0.
25	0.	2.3554E-01	0.	0.	1.5522E-01	0.
26	0.	2.7527E-01	0.	0.	1.9093E-01	0.
27	0.	3.2852E-01	0.	0.	2.4684E-01	0.
28	0.	4.2145E-01	0.	0.	3.6513E-01	0.
29	0.	5.3159E-01	0.	0.	4.2986E-01	0.
30	0.	5.9625E-01	0.	0.	4.6115E-01	0.
31	0.	6.3013E-01	0.	0.	4.9099E-01	0.
32	0.	6.5349E-01	0.	0.	5.1156E-01	0.
33	0.	7.6340E-01	0.	0.	5.6768E-01	0.
34	0.	1.1562E+00	0.	0.	7.1186E-01	0.
35	0.	1.8674E+00	0.	0.	9.8524E-01	0.
36	0.	3.1240E+00	0.	0.	1.7557E+00	0.
37	0.	9.2471E+00	0.	0.	2.8446E+00	0.
38	0.	1.5716E+01	0.	0.	7.6409E+00	0.
39	0.	1.1968E+01	0.	0.	1.4077E+01	0.
40	0.	3.6253E+01	0.	0.	1.7619E+01	0.
41	0.	6.5784E-01	0.	0.	3.5886E+01	0.
42	0.	1.8349E+02	0.	0.	2.6684E-01	0.
43	0.	1.8888E+03	0.	0.	4.2268E-01	0.
44	0.	2.5049E+01	0.	0.	8.0571E-01	0.
45	0.	2.5506E+01	0.	0.	1.3141E+00	0.
46	0.	3.5602E+01	0.	0.	2.3401E+00	0.
47	0.	9.8716E+01	0.	0.	7.3769E+00	0.

TAB. F.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using Flat Weighting.

Grp	U-235 (n,f)	U-238 (n,f)	U-238 (n,γ)	Np-237 (n,f)	Pu-239 (n,f)	Am-241 (n,f)
1	2.0730E+00	1.2190E+00	5.2804E-04	2.2071E+00	2.3811E+00	2.5474E+00
2	1.9295E+00	1.0428E+00	1.4006E-03	2.1063E+00	2.2863E+00	2.5661E+00
3	1.7302E+00	9.7830E-01	1.5887E-03	2.0841E+00	2.2219E+00	2.4080E+00
4	1.7608E+00	9.8793E-01	1.0525E-03	2.1774E+00	2.2362E+00	2.3680E+00
5	1.7469E+00	9.9636E-01	1.3561E-03	2.2235E+00	2.2530E+00	2.4052E+00
6	1.4258E+00	8.6164E-01	2.4120E-03	1.9547E+00	2.0416E+00	2.1624E+00
7	1.0555E+00	5.4549E-01	4.6625E-03	1.5224E+00	1.6870E+00	1.7483E+00
8	1.1104E+00	5.5498E-01	8.8709E-03	1.5290E+00	1.7493E+00	1.7907E+00
9	1.1703E+00	5.3415E-01	1.5594E-02	1.6280E+00	1.8433E+00	1.8464E+00
10	1.2152E+00	5.3271E-01	2.1367E-02	1.6660E+00	1.8943E+00	1.8711E+00
11	1.2432E+00	5.4403E-01	2.6310E-02	1.6804E+00	1.9261E+00	1.8845E+00
12	1.2560E+00	5.4972E-01	3.0705E-02	1.6855E+00	1.9478E+00	1.8917E+00
13	1.2599E+00	5.5067E-01	3.2634E-02	1.6868E+00	1.9550E+00	1.8940E+00
14	1.2642E+00	5.5129E-01	3.4782E-02	1.6875E+00	1.9631E+00	1.8966E+00
15	1.2697E+00	5.4443E-01	4.2970E-02	1.6832E+00	1.9865E+00	1.8989E+00
16	1.2595E+00	4.8376E-01	5.6272E-02	1.6656E+00	1.9946E+00	1.8787E+00
17	1.2319E+00	3.2313E-01	7.1111E-02	1.6260E+00	1.9500E+00	1.8354E+00
18	1.2004E+00	4.4018E-02	1.0330E-01	1.5299E+00	1.8195E+00	1.6196E+00
19	1.1521E+00	1.2809E-02	1.2356E-01	1.4011E+00	1.7008E+00	9.8844E-01
20	1.1110E+00	3.7983E-03	1.1999E-01	1.2445E+00	1.6731E+00	5.7760E-01
21	1.1159E+00	1.4955E-03	1.1687E-01	1.0224E+00	1.5988E+00	3.1873E-01
22	1.1282E+00	5.9988E-04	1.1018E-01	6.5194E-01	1.5742E+00	1.4473E-01
23	1.1795E+00	2.8527E-04	1.1109E-01	2.8178E-01	1.5375E+00	6.9285E-02
24	1.2256E+00	1.5015E-04	1.1408E-01	9.8144E-02	1.5213E+00	4.0678E-02
25	1.2985E+00	9.3264E-05	1.2074E-01	4.5283E-02	1.4838E+00	2.6927E-02
26	1.4376E+00	6.8791E-05	1.4576E-01	2.6086E-02	1.4709E+00	1.8174E-02
27	1.6021E+00	3.8553E-05	1.9605E-01	1.6780E-02	1.5140E+00	1.4839E-02
28	1.8091E+00	8.4376E-05	3.0616E-01	1.5320E-02	1.5780E+00	1.3970E-02
29	1.9679E+00	3.6056E-05	4.0537E-01	1.5819E-02	1.6389E+00	1.4727E-02
30	2.1196E+00	5.9821E-05	4.5020E-01	1.6263E-02	1.6509E+00	1.5544E-02
31	2.1834E+00	8.4858E-05	4.8043E-01	1.6493E-02	1.6783E+00	1.6008E-02
32	2.2794E+00	9.5280E-05	5.0027E-01	1.6618E-02	1.7028E+00	1.6176E-02
33	2.3943E+00	1.1676E-04	5.5527E-01	1.7580E-02	1.7732E+00	1.6945E-02
34	2.8839E+00	1.1372E-04	6.5568E-01	2.0739E-02	1.9613E+00	2.0123E-02
35	3.9884E+00	1.6682E-05	9.0286E-01	2.5089E-02	2.4423E+00	2.8710E-02
36	5.5263E+00	4.4890E-09	1.4466E+00	3.3672E-02	3.3808E+00	3.8856E-02
37	9.0932E+00	1.0089E-03	2.6131E+00	4.5093E-02	6.4377E+00	6.5820E-02
38	1.5820E+01	1.0793E-05	4.3113E+00	6.2187E-02	1.1612E+01	1.3073E-01
39	2.0238E+01	1.9529E-05	1.7530E+01	1.0463E-01	1.9109E+01	1.7537E-01
40	3.1495E+01	3.0217E-05	1.3687E+01	1.6104E-01	5.0156E+01	1.6855E-01
41	4.9598E+01	1.4925E-04	1.0974E+02	9.3968E-02	5.0935E+01	3.4396E-01
42	5.3431E+01	6.7098E-05	1.5355E+02	2.0404E-02	3.8889E+01	1.2041E+00
43	1.6209E+01	1.7685E-06	7.5269E-01	8.9537E-03	9.7792E+00	6.3718E-01
44	3.8748E+01	1.8697E-06	4.8536E-01	1.9759E-02	2.5693E+01	4.1120E+00
45	6.8071E+01	2.5136E-06	6.0828E-01	1.1841E-02	1.3152E+02	2.1459E+00
46	1.7642E+02	3.9826E-06	9.3369E-01	6.6293E-03	1.3911E+03	8.1617E+00
47	5.7426E+02	1.1744E-05	2.7118E+00	2.1237E-02	7.7036E+02	3.0462E+00

TAB. F.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using Flat Weighting.

Grp	B-nat (n,disap)	B-nat (n,γ)	B-nat (n,α)	Cd-nat (n,γ)	Gd-nat (n,γ)
1	7.6407E-02	1.5053E-07	3.0976E-02	2.1208E-03	2.4886E-03
2	7.9905E-02	1.7882E-07	3.4531E-02	2.4866E-03	2.1315E-03
3	6.7897E-02	2.0181E-07	2.9561E-02	3.1593E-03	1.8832E-03
4	5.9129E-02	3.0440E-07	1.6773E-02	4.0179E-03	1.9789E-03
5	6.1562E-02	4.4787E-07	1.6332E-02	4.8948E-03	2.4722E-03
6	7.0444E-02	4.5805E-07	2.2291E-02	6.0266E-03	3.9065E-03
7	7.4394E-02	8.9872E-07	2.4125E-02	7.5310E-03	1.0301E-02
8	8.8001E-02	7.6184E-07	5.1905E-02	1.0114E-02	2.3317E-02
9	7.8770E-02	6.0166E-07	6.0311E-02	1.3751E-02	3.8045E-02
10	8.6600E-02	6.3317E-07	7.3065E-02	1.6510E-02	4.9168E-02
11	7.5143E-02	3.5801E-06	6.3200E-02	1.8957E-02	5.6045E-02
12	6.7661E-02	2.4064E-06	5.6367E-02	2.0413E-02	6.0551E-02
13	6.9202E-02	1.7273E-06	5.7995E-02	2.1056E-02	6.2195E-02
14	7.2764E-02	1.1874E-06	6.1218E-02	2.1894E-02	6.4171E-02
15	9.5772E-02	7.3335E-07	8.1458E-02	2.4860E-02	7.1655E-02
16	1.1029E-01	2.6339E-06	9.9585E-02	2.9861E-02	8.3987E-02
17	6.5625E-02	9.9816E-07	6.1386E-02	3.6923E-02	1.0073E-01
18	4.3984E-02	1.6517E-06	4.3272E-02	4.9077E-02	1.2970E-01
19	5.0416E-02	4.7950E-06	5.0265E-02	6.3907E-02	1.4935E-01
20	6.3629E-02	1.0366E-05	6.3537E-02	7.4044E-02	1.5755E-01
21	8.5233E-02	1.5158E-05	8.5162E-02	8.1967E-02	1.6237E-01
22	1.3202E-01	2.1241E-05	1.3197E-01	9.1030E-02	1.7035E-01
23	1.6711E-01	3.4589E-05	1.6706E-01	9.8453E-02	1.7966E-01
24	1.8835E-01	2.9291E-05	1.8830E-01	1.0307E-01	1.9332E-01
25	2.5039E-01	3.3349E-05	2.5035E-01	1.0982E-01	2.2459E-01
26	3.3561E-01	4.2366E-05	3.3556E-01	1.2859E-01	2.9239E-01
27	4.1486E-01	5.4351E-05	4.1480E-01	1.6142E-01	4.2484E-01
28	5.1518E-01	6.9816E-05	5.1511E-01	2.1628E-01	6.4900E-01
29	6.1767E-01	8.4860E-05	6.1759E-01	2.8302E-01	7.9155E-01
30	6.8898E-01	9.6661E-05	6.8888E-01	3.4033E-01	8.7470E-01
31	7.3781E-01	1.1354E-04	7.3769E-01	3.7978E-01	9.3163E-01
32	7.7043E-01	1.3678E-04	7.7029E-01	4.1229E-01	9.6873E-01
33	8.6513E-01	7.2494E-04	8.6441E-01	5.1441E-01	1.0863E+00
34	1.1343E+00	1.6425E-04	1.1341E+00	7.9893E-01	1.4524E+00
35	1.6614E+00	2.2833E-04	1.6611E+00	1.1811E+00	2.3379E+00
36	2.4327E+00	3.3246E-04	2.4323E+00	1.4583E+00	3.9830E+00
37	3.9145E+00	5.3181E-04	3.9139E+00	1.1875E+00	7.8896E+00
38	6.6849E+00	9.0509E-04	6.6840E+00	4.4112E+00	1.4126E+01
39	9.7482E+00	1.3181E-03	9.7468E+00	5.9393E+00	2.7441E+01
40	1.4915E+01	2.0162E-03	1.4913E+01	1.4101E+01	2.7700E+01
41	2.5771E+01	3.4836E-03	2.5768E+01	3.0126E+00	7.2990E+01
42	4.3856E+01	5.9299E-03	4.3850E+01	1.0992E-01	3.2388E+01
43	6.7037E+01	9.0708E-03	6.7028E+01	8.1739E-01	1.0767E+02
44	1.0525E+02	1.4255E-02	1.0523E+02	8.2288E+00	2.4449E+01
45	1.5316E+02	2.0761E-02	1.5314E+02	8.5875E+01	9.5676E+01
46	2.5215E+02	3.4217E-02	2.5211E+02	3.0114E+03	1.7105E+03
47	7.5782E+02	1.0302E-01	7.5771E+02	3.0226E+03	3.9574E+04

APPENDIX G

1/4 T PV Weighting IRDF-2002 Neutron Dosimeter Cross Sections

TAB. G.1

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using 1/4 T PV Weighting.

Grp	Li-6 (n,t)	B-10 (n,α)	F-19 (n,2n)	Na-23 (n,γ)	Na-23 (n,2n)	Mg-24 (n,p)
1	2.3601E-02	4.3221E-02	5.3385E-02	2.2804E-04	4.9429E-02	1.6845E-01
2	2.8018E-02	4.2306E-02	1.9138E-02	1.9777E-04	3.6249E-03	1.9908E-01
3	3.4827E-02	4.5361E-02	7.3987E-04	1.7883E-04	0.	1.6293E-01
4	4.1774E-02	5.8355E-02	0.	1.7378E-04	0.	1.2908E-01
5	4.9594E-02	8.1732E-02	0.	1.6993E-04	0.	9.7360E-02
6	6.0791E-02	1.1404E-01	0.	1.6654E-04	0.	2.7466E-02
7	7.5703E-02	1.2288E-01	0.	1.6323E-04	0.	4.1389E-04
8	1.0122E-01	2.6556E-01	0.	1.6075E-04	0.	6.2025E-07
9	1.4368E-01	3.0405E-01	0.	1.6828E-04	0.	0.
10	1.7231E-01	3.7017E-01	0.	1.7476E-04	0.	0.
11	1.8800E-01	3.1918E-01	0.	1.7930E-04	0.	0.
12	1.9801E-01	2.8485E-01	0.	1.8278E-04	0.	0.
13	2.0101E-01	2.9290E-01	0.	1.8396E-04	0.	0.
14	2.0442E-01	3.0946E-01	0.	1.8536E-04	0.	0.
15	2.1342E-01	4.1198E-01	0.	1.9017E-04	0.	0.
16	2.1567E-01	5.0068E-01	0.	1.9806E-04	0.	0.
17	2.1293E-01	3.0596E-01	0.	2.0719E-04	0.	0.
18	2.2061E-01	2.1748E-01	0.	2.2127E-04	0.	0.
19	2.3910E-01	2.5564E-01	0.	2.4071E-04	0.	0.
20	2.5671E-01	3.2150E-01	0.	2.8995E-04	0.	0.
21	2.8544E-01	4.4260E-01	0.	3.4345E-04	0.	0.
22	3.3683E-01	6.6153E-01	0.	3.1503E-04	0.	0.
23	5.0975E-01	8.4610E-01	0.	4.7257E-04	0.	0.
24	1.0263E+00	9.5459E-01	0.	6.3904E-04	0.	0.
25	2.4746E+00	1.2683E+00	0.	9.5281E-04	0.	0.
26	9.2632E-01	1.6968E+00	0.	8.2188E-04	0.	0.
27	6.5436E-01	2.1375E+00	0.	2.0684E-05	0.	0.
28	7.1654E-01	2.6064E+00	0.	1.2590E-03	0.	0.
29	8.2489E-01	3.1142E+00	0.	4.7238E-03	0.	0.
30	9.2454E-01	3.5434E+00	0.	2.1211E-05	0.	0.
31	9.6786E-01	3.7254E+00	0.	2.1189E-05	0.	0.
32	1.0057E+00	3.8856E+00	0.	2.2553E-05	0.	0.
33	1.1171E+00	4.3502E+00	0.	3.1731E-05	0.	0.
34	1.4329E+00	5.6538E+00	0.	3.3042E-04	0.	0.
35	2.1433E+00	8.5701E+00	0.	5.3427E-03	0.	0.
36	3.1199E+00	1.2562E+01	0.	8.9639E-02	0.	0.
37	5.1208E+00	2.0737E+01	0.	6.7656E-03	0.	0.
38	8.6357E+00	3.5084E+01	0.	6.1246E-03	0.	0.
39	1.2362E+01	5.0292E+01	0.	7.7557E-03	0.	0.
40	1.9210E+01	7.8208E+01	0.	1.1307E-02	0.	0.
41	3.3738E+01	1.3748E+02	0.	1.9298E-02	0.	0.
42	5.5202E+01	2.2508E+02	0.	3.1187E-02	0.	0.
43	8.5187E+01	3.4737E+02	0.	4.7989E-02	0.	0.
44	1.3193E+02	5.3804E+02	0.	7.4134E-02	0.	0.
45	1.9250E+02	7.8514E+02	0.	1.0807E-01	0.	0.
46	3.3989E+02	1.3866E+03	0.	1.9080E-01	0.	0.
47	6.4079E+02	2.6144E+03	0.	3.5960E-01	0.	0.

TAB. G.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using 1/4 T PV Weighting.

Grp	Al-27 (n,p)	Al-27 (n,α)	P-31 (n,p)	S-32 (n,p)	Sc-45 (n,γ)	Ti-46 (n,2n)
1	6.1301E-02	1.0443E-01	8.0746E-02	1.9504E-01	6.2226E-04	7.5700E-02
2	8.9262E-02	1.2391E-01	1.0667E-01	3.1939E-01	7.6280E-04	1.3745E-03
3	1.0587E-01	1.0308E-01	1.3674E-01	3.7877E-01	8.7687E-04	0.
4	9.7815E-02	7.5654E-02	1.4065E-01	3.3756E-01	8.6841E-04	0.
5	8.1584E-02	3.9795E-02	1.3489E-01	3.2279E-01	7.3538E-04	0.
6	6.0626E-02	9.9044E-03	1.3799E-01	3.1134E-01	5.6461E-04	0.
7	3.7311E-02	4.0980E-04	1.2768E-01	2.4087E-01	5.6580E-04	0.
8	1.1340E-02	2.4639E-06	9.8805E-02	2.7597E-01	8.8883E-04	0.
9	3.2844E-03	1.3485E-07	6.6404E-02	1.8679E-01	1.4065E-03	0.
10	4.9037E-04	0.	6.1983E-02	9.9856E-02	1.7153E-03	0.
11	7.4010E-05	0.	5.1387E-02	6.8690E-02	1.9530E-03	0.
12	7.0679E-06	0.	3.5785E-02	7.2357E-02	2.1141E-03	0.
13	5.0878E-06	0.	3.2199E-02	6.6472E-02	2.1656E-03	0.
14	4.3333E-06	0.	2.7301E-02	6.1120E-02	2.2259E-03	0.
15	1.9814E-06	0.	1.1922E-02	2.1288E-02	2.4203E-03	0.
16	9.6985E-09	0.	6.4660E-03	3.8128E-03	2.8190E-03	0.
17	0.	0.	1.1581E-03	5.7288E-04	3.2893E-03	0.
18	0.	0.	1.2572E-04	5.5136E-05	4.1580E-03	0.
19	0.	0.	1.1945E-05	1.3091E-06	5.0486E-03	0.
20	0.	0.	2.6011E-06	0.	5.7334E-03	0.
21	0.	0.	1.3665E-08	0.	6.5848E-03	0.
22	0.	0.	0.	0.	7.6990E-03	0.
23	0.	0.	0.	0.	9.6422E-03	0.
24	0.	0.	0.	0.	1.1924E-02	0.
25	0.	0.	0.	0.	1.5887E-02	0.
26	0.	0.	0.	0.	2.3322E-02	0.
27	0.	0.	0.	0.	3.6950E-02	0.
28	0.	0.	0.	0.	4.6466E-02	0.
29	0.	0.	0.	0.	7.2227E-02	0.
30	0.	0.	0.	0.	6.8957E-02	0.
31	0.	0.	0.	0.	2.8593E-02	0.
32	0.	0.	0.	0.	5.8347E-02	0.
33	0.	0.	0.	0.	1.0631E-01	0.
34	0.	0.	0.	0.	1.5927E-01	0.
35	0.	0.	0.	0.	2.5944E-01	0.
36	0.	0.	0.	0.	2.0725E-01	0.
37	0.	0.	0.	0.	6.5986E-02	0.
38	0.	0.	0.	0.	8.5330E-02	0.
39	0.	0.	0.	0.	1.8817E-01	0.
40	0.	0.	0.	0.	4.0937E-01	0.
41	0.	0.	0.	0.	8.7544E-01	0.
42	0.	0.	0.	0.	1.5322E+00	0.
43	0.	0.	0.	0.	2.4206E+00	0.
44	0.	0.	0.	0.	3.7878E+00	0.
45	0.	0.	0.	0.	5.5479E+00	0.
46	0.	0.	0.	0.	9.8182E+00	0.
47	0.	0.	0.	0.	1.8524E+01	0.

TAB. G.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using 1/4 T PV Weighting.

Grp	Ti-46 (n,p)	Ti-47 (n,x)	Ti-47 (n,p)	Ti-48 (n,x)	Ti-48 (n,p)	Ti-49 (n,x)
1	2.1225E-01	8.4933E-02	1.0084E-01	2.0288E-02	5.8384E-02	1.2626E-02
2	2.8573E-01	1.4110E-02	1.2147E-01	1.9882E-03	5.8442E-02	1.1719E-03
3	2.9576E-01	7.1954E-04	1.3166E-01	4.0775E-05	3.9261E-02	2.4653E-05
4	2.6701E-01	2.2161E-06	1.2361E-01	5.2583E-07	2.4222E-02	4.1882E-09
5	2.3770E-01	3.3524E-09	1.0658E-01	0.	1.4676E-02	0.
6	1.7813E-01	0.	8.6714E-02	0.	5.2375E-03	0.
7	9.5243E-02	0.	7.0676E-02	0.	6.2659E-04	0.
8	3.9193E-02	0.	5.4657E-02	0.	1.5794E-05	0.
9	4.8484E-03	0.	3.8881E-02	0.	2.1124E-07	0.
10	5.6383E-04	0.	3.1313E-02	0.	0.	0.
11	7.3637E-05	0.	2.5980E-02	0.	0.	0.
12	2.1471E-05	0.	2.2087E-02	0.	0.	0.
13	1.7487E-05	0.	2.0778E-02	0.	0.	0.
14	1.2826E-05	0.	1.9246E-02	0.	0.	0.
15	1.8835E-06	0.	1.4518E-02	0.	0.	0.
16	0.	0.	8.7452E-03	0.	0.	0.
17	0.	0.	4.2257E-03	0.	0.	0.
18	0.	0.	1.9822E-03	0.	0.	0.
19	0.	0.	9.3529E-04	0.	0.	0.
20	0.	0.	6.1119E-04	0.	0.	0.
21	0.	0.	3.6441E-05	0.	0.	0.
22	0.	0.	0.	0.	0.	0.
23	0.	0.	0.	0.	0.	0.
24	0.	0.	0.	0.	0.	0.
25	0.	0.	0.	0.	0.	0.
26	0.	0.	0.	0.	0.	0.
27	0.	0.	0.	0.	0.	0.
28	0.	0.	0.	0.	0.	0.
29	0.	0.	0.	0.	0.	0.
30	0.	0.	0.	0.	0.	0.
31	0.	0.	0.	0.	0.	0.
32	0.	0.	0.	0.	0.	0.
33	0.	0.	0.	0.	0.	0.
34	0.	0.	0.	0.	0.	0.
35	0.	0.	0.	0.	0.	0.
36	0.	0.	0.	0.	0.	0.
37	0.	0.	0.	0.	0.	0.
38	0.	0.	0.	0.	0.	0.
39	0.	0.	0.	0.	0.	0.
40	0.	0.	0.	0.	0.	0.
41	0.	0.	0.	0.	0.	0.
42	0.	0.	0.	0.	0.	0.
43	0.	0.	0.	0.	0.	0.
44	0.	0.	0.	0.	0.	0.
45	0.	0.	0.	0.	0.	0.
46	0.	0.	0.	0.	0.	0.
47	0.	0.	0.	0.	0.	0.

TAB. G.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using 1/4 T PV Weighting.

Grp	V-51 (n,α)	Cr-52 (n,2n)	Mn-55 (n,γ)	Fe-54 (n,2n)	Fe-54 (n,α)	Fe-54 (n,p)
1	1.7787E-02	4.4320E-01	6.4720E-04	2.0624E-02	8.9352E-02	2.5964E-01
2	1.2136E-02	9.2498E-02	6.7144E-04	1.4999E-04	7.9644E-02	4.0367E-01
3	6.4426E-03	0.	6.8198E-04	0.	5.8838E-02	4.6945E-01
4	3.1263E-03	0.	6.9067E-04	0.	4.4388E-02	4.8210E-01
5	8.9399E-04	0.	7.0893E-04	0.	3.2882E-02	4.8231E-01
6	7.3259E-05	0.	7.9637E-04	0.	1.7220E-02	4.7821E-01
7	1.9981E-06	0.	9.4640E-04	0.	4.8957E-03	4.3399E-01
8	2.0623E-07	0.	1.1743E-03	0.	5.9365E-04	3.1340E-01
9	3.1519E-08	0.	1.4349E-03	0.	1.5888E-05	1.9366E-01
10	1.8473E-08	0.	1.6252E-03	0.	1.5322E-06	1.3291E-01
11	1.1995E-08	0.	1.7147E-03	0.	4.1128E-07	7.8794E-02
12	7.6048E-09	0.	1.7438E-03	0.	0.	5.6584E-02
13	6.1991E-09	0.	1.7531E-03	0.	0.	5.1203E-02
14	4.5550E-09	0.	1.7640E-03	0.	0.	4.4947E-02
15	6.7720E-10	0.	1.8015E-03	0.	0.	2.9350E-02
16	0.	0.	1.9046E-03	0.	0.	8.9058E-03
17	0.	0.	2.0848E-03	0.	0.	2.9029E-03
18	0.	0.	2.4785E-03	0.	0.	7.3600E-04
19	0.	0.	3.0576E-03	0.	0.	8.6667E-05
20	0.	0.	3.2808E-03	0.	0.	6.5570E-06
21	0.	0.	3.7278E-03	0.	0.	2.7253E-07
22	0.	0.	4.2226E-03	0.	0.	0.
23	0.	0.	4.8042E-03	0.	0.	0.
24	0.	0.	5.3828E-03	0.	0.	0.
25	0.	0.	6.0679E-03	0.	0.	0.
26	0.	0.	1.0890E-02	0.	0.	0.
27	0.	0.	1.5607E-02	0.	0.	0.
28	0.	0.	1.7807E-02	0.	0.	0.
29	0.	0.	2.6798E-02	0.	0.	0.
30	0.	0.	5.0032E-02	0.	0.	0.
31	0.	0.	3.6331E-02	0.	0.	0.
32	0.	0.	3.4806E-02	0.	0.	0.
33	0.	0.	4.5655E-02	0.	0.	0.
34	0.	0.	5.3162E-02	0.	0.	0.
35	0.	0.	7.6934E-02	0.	0.	0.
36	0.	0.	3.2216E-01	0.	0.	0.
37	0.	0.	5.7565E-01	0.	0.	0.
38	0.	0.	6.6843E+00	0.	0.	0.
39	0.	0.	3.6271E-01	0.	0.	0.
40	0.	0.	3.3263E-01	0.	0.	0.
41	0.	0.	5.0721E-01	0.	0.	0.
42	0.	0.	8.0145E-01	0.	0.	0.
43	0.	0.	1.2236E+00	0.	0.	0.
44	0.	0.	1.8866E+00	0.	0.	0.
45	0.	0.	2.7483E+00	0.	0.	0.
46	0.	0.	4.8474E+00	0.	0.	0.
47	0.	0.	9.1348E+00	0.	0.	0.

TAB. G.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using 1/4 T PV Weighting.

Grp	Fe-56 (n,p)	Fe-58 (n,γ)	Co-59 (n,2n)	Co-59 (n,α)	Co-59 (n,γ)	Ni-58 (n,2n)
1	9.7979E-02	1.1867E-03	7.7766E-01	2.9440E-02	7.6103E-04	4.5791E-02
2	1.1292E-01	6.5808E-04	5.2283E-01	2.8398E-02	8.3930E-04	7.2306E-03
3	8.5842E-02	2.6182E-04	6.0420E-02	1.9572E-02	6.6460E-04	0.
4	6.3359E-02	2.8379E-04	0.	1.3248E-02	6.3867E-04	0.
5	4.5181E-02	3.7871E-04	0.	7.5922E-03	7.2483E-04	0.
6	2.4629E-02	5.2000E-04	0.	2.5216E-03	9.3839E-04	0.
7	5.4463E-03	6.5816E-04	0.	4.1928E-04	1.1808E-03	0.
8	1.6576E-04	7.9178E-04	0.	1.5536E-05	1.5540E-03	0.
9	2.6211E-09	8.8596E-04	0.	4.5525E-08	2.0619E-03	0.
10	3.5316E-12	9.2616E-04	0.	2.5447E-08	2.3452E-03	0.
11	0.	9.5849E-04	0.	2.2747E-08	2.5340E-03	0.
12	0.	9.8503E-04	0.	2.0918E-08	2.6949E-03	0.
13	0.	9.9370E-04	0.	2.0332E-08	2.7590E-03	0.
14	0.	1.0198E-03	0.	1.9647E-08	2.8339E-03	0.
15	0.	1.2199E-03	0.	1.7512E-08	3.0754E-03	0.
16	0.	1.3315E-03	0.	1.4497E-08	3.5712E-03	0.
17	0.	1.2781E-03	0.	1.1691E-08	4.3250E-03	0.
18	0.	1.3218E-03	0.	8.3748E-09	6.0619E-03	0.
19	0.	2.0831E-03	0.	5.7647E-09	6.4541E-03	0.
20	0.	3.3963E-03	0.	4.4958E-09	6.5407E-03	0.
21	0.	3.2526E-03	0.	3.3411E-09	6.7459E-03	0.
22	0.	3.1032E-03	0.	2.2222E-09	7.1125E-03	0.
23	0.	3.1192E-03	0.	9.4910E-10	7.8991E-03	0.
24	0.	2.9850E-03	0.	7.8392E-11	8.9334E-03	0.
25	0.	4.2361E-03	0.	0.	1.0980E-02	0.
26	0.	5.0987E-03	0.	0.	1.5564E-02	0.
27	0.	6.9170E-03	0.	0.	1.6406E-02	0.
28	0.	9.5977E-03	0.	0.	1.8249E-02	0.
29	0.	1.0012E-02	0.	0.	2.6449E-02	0.
30	0.	1.6248E-02	0.	0.	1.8996E-02	0.
31	0.	5.5907E-03	0.	0.	7.4297E-02	0.
32	0.	8.0600E-04	0.	0.	2.4773E-02	0.
33	0.	8.4205E-03	0.	0.	4.1615E-02	0.
34	0.	6.3408E-02	0.	0.	6.8261E-02	0.
35	0.	2.3026E-02	0.	0.	2.2311E-01	0.
36	0.	4.1492E-03	0.	0.	7.0549E-02	0.
37	0.	7.1365E-03	0.	0.	4.6362E-02	0.
38	0.	6.8814E-01	0.	0.	2.9975E-01	0.
39	0.	1.7926E-02	0.	0.	7.3252E+01	0.
40	0.	2.6628E-02	0.	0.	2.9624E+00	0.
41	0.	4.6656E-02	0.	0.	1.7663E+00	0.
42	0.	7.6307E-02	0.	0.	2.3898E+00	0.
43	0.	1.1773E-01	0.	0.	3.4944E+00	0.
44	0.	1.8233E-01	0.	0.	5.2923E+00	0.
45	0.	2.6603E-01	0.	0.	7.6586E+00	0.
46	0.	4.6970E-01	0.	0.	1.3457E+01	0.
47	0.	8.8546E-01	0.	0.	2.5324E+01	0.

TAB. G.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using 1/4 T PV Weighting.

Grp	Ni-58 (n,p)	Ni-60 (n,p)	Cu-63 (n,2n)	Cu-63 (n,γ)	Cu-63 (n,α)	Cu-65 (n,2n)
1	2.5226E-01	1.1989E-01	5.9885E-01	2.5940E-03	3.7744E-02	9.9301E-01
2	4.6741E-01	1.4396E-01	2.6626E-01	2.8373E-03	4.6793E-02	7.2093E-01
3	6.0974E-01	1.1558E-01	1.0770E-02	3.0969E-03	3.7273E-02	1.4136E-01
4	6.3931E-01	8.7380E-02	0.	3.3547E-03	2.6446E-02	0.
5	6.4475E-01	6.4216E-02	0.	3.6227E-03	1.8474E-02	0.
6	6.1875E-01	3.6827E-02	0.	3.9633E-03	1.0035E-02	0.
7	5.4211E-01	1.5034E-02	0.	4.2729E-03	3.4830E-03	0.
8	4.0243E-01	2.2003E-03	0.	4.8173E-03	5.9294E-04	0.
9	2.4648E-01	2.7925E-05	0.	5.4119E-03	3.8984E-05	0.
10	1.6737E-01	5.2470E-08	0.	5.7224E-03	4.5885E-06	0.
11	1.2160E-01	3.1379E-10	0.	5.9640E-03	1.3679E-06	0.
12	9.3194E-02	0.	0.	6.1277E-03	7.3302E-07	0.
13	8.4827E-02	0.	0.	6.1801E-03	4.7235E-07	0.
14	7.5551E-02	0.	0.	6.2414E-03	1.7451E-07	0.
15	5.0770E-02	0.	0.	6.4748E-03	0.	0.
16	2.5188E-02	0.	0.	7.6233E-03	0.	0.
17	1.1434E-02	0.	0.	9.1759E-03	0.	0.
18	3.6037E-03	0.	0.	1.2000E-02	0.	0.
19	4.7232E-04	0.	0.	1.3732E-02	0.	0.
20	1.5479E-04	0.	0.	1.3875E-02	0.	0.
21	3.6573E-05	0.	0.	1.3955E-02	0.	0.
22	3.5593E-06	0.	0.	1.4457E-02	0.	0.
23	2.7259E-07	0.	0.	1.6498E-02	0.	0.
24	0.	0.	0.	1.9055E-02	0.	0.
25	0.	0.	0.	2.3620E-02	0.	0.
26	0.	0.	0.	2.8116E-02	0.	0.
27	0.	0.	0.	3.1952E-02	0.	0.
28	0.	0.	0.	3.8869E-02	0.	0.
29	0.	0.	0.	5.9377E-02	0.	0.
30	0.	0.	0.	6.6645E-02	0.	0.
31	0.	0.	0.	8.8491E-02	0.	0.
32	0.	0.	0.	9.9662E-02	0.	0.
33	0.	0.	0.	1.1551E-01	0.	0.
34	0.	0.	0.	1.6488E-01	0.	0.
35	0.	0.	0.	2.7624E-01	0.	0.
36	0.	0.	0.	5.0926E-01	0.	0.
37	0.	0.	0.	1.7919E+00	0.	0.
38	0.	0.	0.	2.5613E-02	0.	0.
39	0.	0.	0.	3.4496E-02	0.	0.
40	0.	0.	0.	6.9272E-02	0.	0.
41	0.	0.	0.	1.4477E-01	0.	0.
42	0.	0.	0.	2.5220E-01	0.	0.
43	0.	0.	0.	3.9794E-01	0.	0.
44	0.	0.	0.	6.2235E-01	0.	0.
45	0.	0.	0.	9.1150E-01	0.	0.
46	0.	0.	0.	1.6130E+00	0.	0.
47	0.	0.	0.	3.0429E+00	0.	0.

TAB. G.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using 1/4 T PV Weighting.

Grp	Zn-64 (n,p)	As-75 (n,2n)	Y-89 (n,2n)	Zr-90 (n,2n)	Nb-93 (n,2n)	Nb-93 (n,n')
1	1.4987E-01	1.0713E+00	1.0594E+00	8.8480E-01	4.4876E-01	3.4098E-02
2	2.2737E-01	7.2943E-01	4.6612E-01	2.5685E-01	4.3542E-01	4.9125E-02
3	2.5388E-01	1.0234E-01	1.0823E-02	1.7523E-04	2.5047E-01	1.1503E-01
4	2.3029E-01	0.	0.	0.	2.2834E-02	2.1253E-01
5	2.1529E-01	0.	0.	0.	0.	2.4464E-01
6	2.0184E-01	0.	0.	0.	0.	2.4831E-01
7	1.8534E-01	0.	0.	0.	0.	2.5488E-01
8	1.5205E-01	0.	0.	0.	0.	2.6359E-01
9	1.0956E-01	0.	0.	0.	0.	2.6049E-01
10	6.8342E-02	0.	0.	0.	0.	2.4918E-01
11	3.8612E-02	0.	0.	0.	0.	2.3698E-01
12	2.4044E-02	0.	0.	0.	0.	2.2616E-01
13	2.0267E-02	0.	0.	0.	0.	2.2215E-01
14	1.7394E-02	0.	0.	0.	0.	2.1712E-01
15	9.5764E-03	0.	0.	0.	0.	1.9894E-01
16	3.2083E-03	0.	0.	0.	0.	1.6778E-01
17	1.0380E-03	0.	0.	0.	0.	1.3286E-01
18	2.2507E-04	0.	0.	0.	0.	8.7005E-02
19	9.1940E-05	0.	0.	0.	0.	5.1368E-02
20	6.3452E-05	0.	0.	0.	0.	3.6008E-02
21	3.7530E-05	0.	0.	0.	0.	2.4491E-02
22	1.2414E-05	0.	0.	0.	0.	1.5759E-02
23	0.	0.	0.	0.	0.	9.2059E-03
24	0.	0.	0.	0.	0.	5.9643E-03
25	0.	0.	0.	0.	0.	3.7219E-03
26	0.	0.	0.	0.	0.	1.7841E-03
27	0.	0.	0.	0.	0.	6.6840E-04
28	0.	0.	0.	0.	0.	2.5562E-04
29	0.	0.	0.	0.	0.	5.8588E-05
30	0.	0.	0.	0.	0.	1.0991E-06
31	0.	0.	0.	0.	0.	0.
32	0.	0.	0.	0.	0.	0.
33	0.	0.	0.	0.	0.	0.
34	0.	0.	0.	0.	0.	0.
35	0.	0.	0.	0.	0.	0.
36	0.	0.	0.	0.	0.	0.
37	0.	0.	0.	0.	0.	0.
38	0.	0.	0.	0.	0.	0.
39	0.	0.	0.	0.	0.	0.
40	0.	0.	0.	0.	0.	0.
41	0.	0.	0.	0.	0.	0.
42	0.	0.	0.	0.	0.	0.
43	0.	0.	0.	0.	0.	0.
44	0.	0.	0.	0.	0.	0.
45	0.	0.	0.	0.	0.	0.
46	0.	0.	0.	0.	0.	0.
47	0.	0.	0.	0.	0.	0.

TAB. G.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using 1/4 T PV Weighting.

Grp	Nb-93 (n,γ)	Rh-103 (n,n')	Ag-109 (n,γ)	In-115 (n,2n)	In-115 (n,n')	In-115 (n,γ)
1	4.9204E-04	2.4736E-01	1.7042E-04	1.2998E+00	5.9174E-02	9.2495E-04
2	6.1865E-04	4.0943E-01	2.1398E-04	1.1131E+00	9.8495E-02	9.3250E-04
3	7.4843E-04	8.8628E-01	2.4950E-04	4.9213E-01	2.1529E-01	1.0266E-03
4	9.9333E-04	1.2280E+00	2.4146E-04	2.0572E-02	2.7602E-01	1.4670E-03
5	1.3755E-03	1.2743E+00	1.9297E-04	0.	3.1014E-01	2.3062E-03
6	2.1173E-03	1.2259E+00	1.4243E-04	0.	3.3553E-01	4.3129E-03
7	2.9184E-03	1.1713E+00	1.9795E-04	0.	3.3870E-01	9.4364E-03
8	4.0376E-03	1.1080E+00	7.8889E-04	0.	3.2519E-01	2.0840E-02
9	5.6729E-03	1.0340E+00	2.3404E-03	0.	3.3463E-01	4.2098E-02
10	6.8442E-03	9.8681E-01	3.5621E-03	0.	3.5274E-01	6.0953E-02
11	7.9209E-03	9.5184E-01	4.1436E-03	0.	3.5384E-01	7.6953E-02
12	8.7659E-03	9.2505E-01	4.6265E-03	0.	3.4291E-01	9.0933E-02
13	9.0803E-03	9.1570E-01	4.8017E-03	0.	3.3654E-01	9.5144E-02
14	9.4480E-03	9.0476E-01	5.0265E-03	0.	3.2744E-01	9.9741E-02
15	1.0616E-02	8.6750E-01	5.8370E-03	0.	2.8997E-01	1.1663E-01
16	1.3035E-02	8.0709E-01	6.9150E-03	0.	2.2780E-01	1.4033E-01
17	1.7376E-02	7.4285E-01	8.1070E-03	0.	1.6955E-01	1.6355E-01
18	2.7799E-02	6.5898E-01	1.0062E-02	0.	1.0098E-01	1.8074E-01
19	4.3044E-02	5.7853E-01	1.2252E-02	0.	4.8555E-02	1.8032E-01
20	4.8263E-02	4.9180E-01	1.3612E-02	0.	2.7780E-02	1.7188E-01
21	5.0198E-02	3.2413E-01	1.5256E-02	0.	1.4595E-02	1.6099E-01
22	5.2242E-02	1.9458E-01	1.7323E-02	0.	6.4916E-03	1.5242E-01
23	5.5843E-02	1.3637E-01	2.0313E-02	0.	2.1464E-03	1.4552E-01
24	5.9989E-02	1.0321E-01	2.3224E-02	0.	1.5024E-04	1.5201E-01
25	6.8569E-02	6.5947E-02	2.7239E-02	0.	0.	1.8224E-01
26	9.1717E-02	2.3120E-02	3.3168E-02	0.	0.	2.4632E-01
27	1.3180E-01	5.3290E-03	4.1495E-02	0.	0.	3.3879E-01
28	1.8454E-01	1.6314E-03	5.0685E-02	0.	0.	4.4179E-01
29	2.4404E-01	3.5772E-06	5.6071E-02	0.	0.	5.4302E-01
30	2.9352E-01	0.	5.9039E-02	0.	0.	6.1988E-01
31	3.1738E-01	0.	6.0233E-02	0.	0.	6.4755E-01
32	3.4694E-01	0.	6.1233E-02	0.	0.	6.6515E-01
33	3.7470E-01	0.	6.3563E-02	0.	0.	7.0752E-01
34	5.1329E-01	0.	6.8547E-02	0.	0.	8.3795E-01
35	8.5780E-01	0.	7.6971E-02	0.	0.	1.0426E+00
36	1.2995E+00	0.	1.2228E-01	0.	0.	1.3209E+00
37	2.0710E+00	0.	2.1973E-01	0.	0.	2.0372E+00
38	1.6000E+00	0.	3.1460E-01	0.	0.	5.3145E+00
39	3.1694E+00	0.	8.0721E-01	0.	0.	5.4166E+00
40	2.2898E-01	0.	1.7078E+00	0.	0.	7.0429E+00
41	2.1666E-01	0.	8.8488E-01	0.	0.	6.4472E+00
42	8.0580E-02	0.	5.8811E+01	0.	0.	4.6345E+01
43	1.1714E-01	0.	1.3916E+01	0.	0.	6.8981E+01
44	1.7382E-01	0.	1.0570E+00	0.	0.	3.2928E+03
45	2.4707E-01	0.	1.1004E+00	0.	0.	1.0374E+02
46	4.2555E-01	0.	1.6392E+00	0.	0.	7.8599E+01
47	7.9058E-01	0.	2.9188E+00	0.	0.	1.1864E+02

TAB. G.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using 1/4 T PV Weighting.

Grp	I-127 (n,2n)	La-139 (n,γ)	Pr-141 (n,2n)	Tm-169 (n,2n)	Ta-181 (n,γ)	W-186 (n,γ)
1	1.6657E+00	8.7994E-04	1.7222E+00	1.9872E+00	1.2345E-07	8.8654E-04
2	1.5365E+00	1.1289E-03	1.4574E+00	1.9562E+00	3.5772E-07	1.3155E-03
3	7.0591E-01	9.8629E-04	6.2044E-01	1.4584E+00	1.9164E-06	1.4125E-03
4	2.0644E-02	9.4864E-04	1.2847E-02	6.5959E-01	1.0335E-05	1.2597E-03
5	0.	1.1635E-03	0.	1.8268E-02	4.7668E-05	1.2265E-03
6	0.	1.5280E-03	0.	0.	2.5723E-04	1.7520E-03
7	0.	1.9882E-03	0.	0.	1.5415E-03	3.3496E-03
8	0.	2.7424E-03	0.	0.	8.2510E-03	7.3181E-03
9	0.	3.8147E-03	0.	0.	2.1345E-02	1.3635E-02
10	0.	4.5583E-03	0.	0.	3.1450E-02	1.8271E-02
11	0.	5.1282E-03	0.	0.	3.8589E-02	2.1608E-02
12	0.	5.5726E-03	0.	0.	4.4116E-02	2.3874E-02
13	0.	5.7284E-03	0.	0.	4.5912E-02	2.4565E-02
14	0.	5.9190E-03	0.	0.	4.8193E-02	2.5349E-02
15	0.	6.5836E-03	0.	0.	5.6657E-02	2.7625E-02
16	0.	7.7390E-03	0.	0.	7.0577E-02	3.0546E-02
17	0.	9.1104E-03	0.	0.	8.6784E-02	3.3180E-02
18	0.	8.4935E-03	0.	0.	1.0078E-01	3.7779E-02
19	0.	7.1828E-03	0.	0.	1.0504E-01	4.4643E-02
20	0.	6.8357E-03	0.	0.	1.0490E-01	4.7121E-02
21	0.	6.6317E-03	0.	0.	1.1182E-01	4.6471E-02
22	0.	6.5626E-03	0.	0.	1.2366E-01	4.5521E-02
23	0.	6.8086E-03	0.	0.	1.4759E-01	4.5772E-02
24	0.	7.3896E-03	0.	0.	1.6904E-01	4.7637E-02
25	0.	8.9368E-03	0.	0.	2.1049E-01	5.3137E-02
26	0.	1.3608E-02	0.	0.	3.0148E-01	8.0220E-02
27	0.	1.9811E-02	0.	0.	4.0747E-01	1.3074E-01
28	0.	2.7762E-02	0.	0.	5.1501E-01	1.4345E-01
29	0.	3.6571E-02	0.	0.	6.3995E-01	1.6073E-01
30	0.	4.4065E-02	0.	0.	7.5253E-01	1.7911E-01
31	0.	4.7209E-02	0.	0.	7.9957E-01	1.8751E-01
32	0.	4.9954E-02	0.	0.	8.3956E-01	1.9518E-01
33	0.	5.4110E-02	0.	0.	9.6242E-01	2.1823E-01
34	0.	8.9273E-02	0.	0.	1.3737E+00	2.8226E-01
35	0.	1.6574E-01	0.	0.	2.6374E+00	4.3949E-01
36	0.	1.9863E-01	0.	0.	4.4952E+00	6.9471E-01
37	0.	2.5366E-01	0.	0.	1.0510E+01	3.0704E+00
38	0.	1.9148E-02	0.	0.	2.4604E+01	8.6115E+00
39	0.	1.1065E-02	0.	0.	3.2960E+01	3.4605E+00
40	0.	8.1860E+00	0.	0.	5.8808E+01	2.5995E-01
41	0.	1.7370E-01	0.	0.	6.8338E+01	3.3760E+02
42	0.	4.0466E-01	0.	0.	9.7935E+01	6.2118E+00
43	0.	7.2665E-01	0.	0.	4.1054E+02	4.9206E+00
44	0.	1.2178E+00	0.	0.	4.4839E+00	6.1380E+00
45	0.	2.1764E+00	0.	0.	4.9455E+00	8.3388E+00
46	0.	3.2467E+00	0.	0.	7.7845E+00	1.4139E+01
47	0.	6.1481E+00	0.	0.	1.4181E+01	2.6283E+01

TAB. G.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using 1/4 T PV Weighting.

Grp	Au-197 (n,2n)	Au-197 (n,γ)	Hg-199 (n,n')	Pb-204 (n,n')	Th-232 (n,γ)	Th-232 (n,f)
1	2.1013E+00	8.3464E-04	1.1393E-01	5.8667E-02	1.4784E-03	4.1418E-01
2	2.0782E+00	1.6633E-03	2.0837E-01	1.0415E-01	1.6228E-03	3.2318E-01
3	1.5102E+00	1.6248E-03	4.5057E-01	2.0872E-01	1.8926E-03	3.1013E-01
4	4.8736E-01	1.1551E-03	5.9181E-01	2.6900E-01	2.6960E-03	3.2436E-01
5	1.6467E-02	9.2211E-04	6.3474E-01	2.3231E-01	4.0613E-03	3.6311E-01
6	0.	3.2300E-03	6.3640E-01	1.6683E-01	6.6607E-03	2.8615E-01
7	0.	6.1614E-03	6.0993E-01	1.1667E-01	1.0617E-02	1.4372E-01
8	0.	1.0768E-02	5.6679E-01	7.2617E-02	1.6772E-02	1.4233E-01
9	0.	1.9964E-02	4.8567E-01	3.4173E-02	2.6238E-02	1.3949E-01
10	0.	2.7109E-02	4.3090E-01	1.5880E-02	3.3265E-02	1.2698E-01
11	0.	3.1753E-02	3.8425E-01	6.1270E-03	3.9751E-02	1.2048E-01
12	0.	3.5716E-02	3.5292E-01	6.1982E-04	4.5796E-02	1.1508E-01
13	0.	3.7624E-02	3.4477E-01	1.8718E-04	4.8358E-02	1.1585E-01
14	0.	4.0110E-02	3.3514E-01	7.7510E-05	5.1354E-02	1.2623E-01
15	0.	4.9426E-02	2.9503E-01	7.7892E-07	6.1066E-02	1.2265E-01
16	0.	6.0528E-02	2.3341E-01	0.	8.1557E-02	9.0192E-02
17	0.	6.7956E-02	1.6071E-01	0.	9.9063E-02	7.9587E-02
18	0.	7.5386E-02	8.9724E-02	0.	1.2072E-01	7.5049E-03
19	0.	8.4235E-02	5.1931E-02	0.	1.4684E-01	6.8876E-04
20	0.	9.0947E-02	3.5525E-02	0.	1.6072E-01	1.2563E-04
21	0.	9.9769E-02	1.9297E-02	0.	1.6476E-01	3.0505E-05
22	0.	1.2016E-01	3.8803E-03	0.	1.5821E-01	6.0866E-06
23	0.	1.5301E-01	0.	0.	1.4498E-01	0.
24	0.	1.8581E-01	0.	0.	1.4406E-01	0.
25	0.	2.3623E-01	0.	0.	1.5540E-01	0.
26	0.	2.7556E-01	0.	0.	1.9124E-01	0.
27	0.	3.3608E-01	0.	0.	2.5525E-01	0.
28	0.	4.2233E-01	0.	0.	3.6649E-01	0.
29	0.	5.3060E-01	0.	0.	4.2952E-01	0.
30	0.	6.0556E-01	0.	0.	4.6867E-01	0.
31	0.	6.3011E-01	0.	0.	4.9097E-01	0.
32	0.	6.5271E-01	0.	0.	5.1096E-01	0.
33	0.	7.5890E-01	0.	0.	5.6592E-01	0.
34	0.	1.1324E+00	0.	0.	7.0433E-01	0.
35	0.	1.9028E+00	0.	0.	9.9824E-01	0.
36	0.	3.3272E+00	0.	0.	1.8149E+00	0.
37	0.	9.9661E+00	0.	0.	3.0829E+00	0.
38	0.	1.5713E+01	0.	0.	8.8470E+00	0.
39	0.	1.2173E+01	0.	0.	1.4878E+01	0.
40	0.	4.1246E+01	0.	0.	1.7161E+01	0.
41	0.	8.3842E-01	0.	0.	3.3580E+01	0.
42	0.	2.2447E+02	0.	0.	2.6817E-01	0.
43	0.	1.5248E+03	0.	0.	4.3898E-01	0.
44	0.	2.4923E+01	0.	0.	8.1946E-01	0.
45	0.	2.5685E+01	0.	0.	1.3387E+00	0.
46	0.	3.8192E+01	0.	0.	2.5691E+00	0.
47	0.	6.8065E+01	0.	0.	5.0035E+00	0.

TAB. G.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using 1/4 T PV Weighting.

Grp	U-235 (n,f)	U-238 (n,f)	U-238 (n,γ)	Np-237 (n,f)	Pu-239 (n,f)	Am-241 (n,f)
1	2.0789E+00	1.2041E+00	6.3465E-04	2.2057E+00	2.3868E+00	2.5895E+00
2	1.9008E+00	1.0249E+00	1.4898E-03	2.0959E+00	2.2751E+00	2.5465E+00
3	1.7310E+00	9.7912E-01	1.4819E-03	2.0930E+00	2.2226E+00	2.3991E+00
4	1.7626E+00	9.8941E-01	1.0501E-03	2.1821E+00	2.2369E+00	2.3685E+00
5	1.7417E+00	9.9636E-01	1.3871E-03	2.2221E+00	2.2528E+00	2.4069E+00
6	1.3850E+00	8.3240E-01	2.5362E-03	1.9137E+00	2.0045E+00	2.1134E+00
7	1.0549E+00	5.4324E-01	4.7612E-03	1.5162E+00	1.6837E+00	1.7471E+00
8	1.1115E+00	5.5542E-01	8.9504E-03	1.5307E+00	1.7515E+00	1.7917E+00
9	1.1731E+00	5.3336E-01	1.5905E-02	1.6309E+00	1.8466E+00	1.8481E+00
10	1.2157E+00	5.3287E-01	2.1449E-02	1.6662E+00	1.8948E+00	1.8714E+00
11	1.2432E+00	5.4404E-01	2.6311E-02	1.6804E+00	1.9261E+00	1.8845E+00
12	1.2562E+00	5.4975E-01	3.0767E-02	1.6855E+00	1.9480E+00	1.8917E+00
13	1.2599E+00	5.5067E-01	3.2634E-02	1.6868E+00	1.9550E+00	1.8940E+00
14	1.2643E+00	5.5130E-01	3.4817E-02	1.6875E+00	1.9632E+00	1.8966E+00
15	1.2697E+00	5.4438E-01	4.3004E-02	1.6832E+00	1.9865E+00	1.8989E+00
16	1.2587E+00	4.7915E-01	5.6777E-02	1.6645E+00	1.9940E+00	1.8775E+00
17	1.2311E+00	3.1602E-01	7.1596E-02	1.6247E+00	1.9482E+00	1.8339E+00
18	1.1996E+00	4.1377E-02	1.0476E-01	1.5260E+00	1.8137E+00	1.6080E+00
19	1.1495E+00	1.2627E-02	1.2345E-01	1.3969E+00	1.6996E+00	9.7355E-01
20	1.1110E+00	3.7654E-03	1.1997E-01	1.2432E+00	1.6726E+00	5.7529E-01
21	1.1163E+00	1.3948E-03	1.1636E-01	9.9908E-01	1.5947E+00	3.0049E-01
22	1.1277E+00	6.0794E-04	1.1022E-01	6.5869E-01	1.5744E+00	1.4658E-01
23	1.1820E+00	2.8343E-04	1.1126E-01	2.7198E-01	1.5360E+00	6.7836E-02
24	1.2259E+00	1.4694E-04	1.1411E-01	9.6701E-02	1.5214E+00	4.0412E-02
25	1.3003E+00	9.3331E-05	1.2082E-01	4.4873E-02	1.4832E+00	2.6790E-02
26	1.4390E+00	6.7674E-05	1.4594E-01	2.5998E-02	1.4720E+00	1.8176E-02
27	1.6231E+00	4.1712E-05	2.0336E-01	1.6557E-02	1.5173E+00	1.4667E-02
28	1.8060E+00	8.4455E-05	3.0752E-01	1.5312E-02	1.5789E+00	1.3971E-02
29	1.9641E+00	3.6185E-05	4.0476E-01	1.5812E-02	1.6388E+00	1.4716E-02
30	2.1163E+00	6.6806E-05	4.5806E-01	1.6327E-02	1.6540E+00	1.5671E-02
31	2.1832E+00	8.4848E-05	4.8041E-01	1.6493E-02	1.6782E+00	1.6008E-02
32	2.2765E+00	9.5018E-05	4.9972E-01	1.6615E-02	1.7021E+00	1.6172E-02
33	2.3879E+00	1.1614E-04	5.5352E-01	1.7532E-02	1.7714E+00	1.6911E-02
34	2.8702E+00	1.0791E-04	6.5555E-01	2.0630E-02	1.9378E+00	1.9965E-02
35	4.0823E+00	1.3159E-05	9.1224E-01	2.5535E-02	2.5025E+00	2.9554E-02
36	5.6345E+00	5.9153E-09	1.4873E+00	3.4411E-02	3.3859E+00	3.9272E-02
37	9.7433E+00	1.1373E-03	2.7940E+00	4.3349E-02	7.0564E+00	7.0167E-02
38	1.7426E+01	1.2443E-05	4.9941E+00	6.7059E-02	1.3026E+01	1.3239E-01
39	2.0410E+01	1.7220E-05	1.9321E+01	1.0150E-01	1.9291E+01	1.7884E-01
40	3.3460E+01	2.7199E-05	1.2518E+01	2.3481E-01	4.8337E+01	1.7580E-01
41	4.8117E+01	1.4996E-04	9.5008E+01	6.8124E-02	6.7740E+01	3.4747E-01
42	4.8592E+01	6.9040E-05	1.5759E+02	2.2430E-02	3.8989E+01	1.3255E+00
43	1.6628E+01	1.7404E-06	7.0503E-01	8.5297E-03	1.0229E+01	6.3394E-01
44	4.1549E+01	1.8856E-06	4.8777E-01	1.9210E-02	2.6569E+01	4.2766E+00
45	7.0120E+01	2.5470E-06	6.1540E-01	1.3676E-02	1.4506E+02	2.2311E+00
46	1.8680E+02	4.3218E-06	1.0103E+00	6.9513E-03	1.1930E+03	6.2518E+00
47	3.7172E+02	8.0413E-06	1.8600E+00	1.4056E-02	5.5396E+02	2.0131E+00

TAB. G.1 Continued

Response Functions Included with BUGENDF70.BOLIB.
IRDF-2002 Neutron Dosimeter Cross Sections [barns] Collapsed Using 1/4 T PV Weighting.

Grp	B-nat (n,disap)	B-nat (n,γ)	B-nat (n,α)	Cd-nat (n,γ)	Gd-nat (n,γ)
1	7.8683E-02	1.5744E-07	3.2170E-02	2.1395E-03	2.4199E-03
2	7.8816E-02	1.8027E-07	3.4495E-02	2.5343E-03	2.1078E-03
3	6.6922E-02	2.0716E-07	2.8103E-02	3.2573E-03	1.8592E-03
4	5.8994E-02	2.8647E-07	1.6440E-02	4.0640E-03	2.0020E-03
5	6.1861E-02	4.7178E-07	1.6561E-02	4.9446E-03	2.5031E-03
6	7.1156E-02	4.7695E-07	2.2579E-02	6.1324E-03	4.1408E-03
7	7.4450E-02	9.1762E-07	2.4327E-02	7.5913E-03	1.0671E-02
8	8.8343E-02	7.3650E-07	5.2586E-02	1.0164E-02	2.3513E-02
9	7.8252E-02	5.8311E-07	6.0182E-02	1.3906E-02	3.8719E-02
10	8.6791E-02	6.4144E-07	7.3289E-02	1.6546E-02	4.9288E-02
11	7.5137E-02	3.5807E-06	6.3194E-02	1.8958E-02	5.6047E-02
12	6.7703E-02	2.3840E-06	5.6411E-02	2.0432E-02	6.0603E-02
13	6.9202E-02	1.7273E-06	5.7995E-02	2.1056E-02	6.2195E-02
14	7.2828E-02	1.1795E-06	6.1275E-02	2.1907E-02	6.4204E-02
15	9.5867E-02	7.3387E-07	8.1548E-02	2.4872E-02	7.1685E-02
16	1.0949E-01	2.6598E-06	9.9136E-02	3.0090E-02	8.4547E-02
17	6.4716E-02	1.0064E-06	6.0574E-02	3.7129E-02	1.0120E-01
18	4.3707E-02	1.5826E-06	4.3063E-02	4.9570E-02	1.3094E-01
19	5.0753E-02	4.9707E-06	5.0605E-02	6.4253E-02	1.4963E-01
20	6.3742E-02	1.0401E-05	6.3649E-02	7.4095E-02	1.5759E-01
21	8.7680E-02	1.5582E-05	8.7612E-02	8.2704E-02	1.6294E-01
22	1.3105E-01	2.1128E-05	1.3100E-01	9.0895E-02	1.7024E-01
23	1.6758E-01	3.3246E-05	1.6752E-01	9.8657E-02	1.8015E-01
24	1.8904E-01	2.9310E-05	1.8900E-01	1.0315E-01	1.9360E-01
25	2.5116E-01	3.3406E-05	2.5112E-01	1.0988E-01	2.2499E-01
26	3.3599E-01	4.2430E-05	3.3594E-01	1.2872E-01	2.9321E-01
27	4.2330E-01	5.5689E-05	4.2324E-01	1.6549E-01	4.4483E-01
28	5.1615E-01	6.9959E-05	5.1607E-01	2.1686E-01	6.5055E-01
29	6.1672E-01	8.4721E-05	6.1663E-01	2.8226E-01	7.9043E-01
30	7.0172E-01	1.0020E-04	7.0162E-01	3.5048E-01	8.8975E-01
31	7.3778E-01	1.1353E-04	7.3767E-01	3.7975E-01	9.3160E-01
32	7.6952E-01	1.3598E-04	7.6938E-01	4.1137E-01	9.6769E-01
33	8.6213E-01	7.8109E-04	8.6135E-01	5.1098E-01	1.0822E+00
34	1.1196E+00	1.6285E-04	1.1195E+00	7.8534E-01	1.4314E+00
35	1.6972E+00	2.3318E-04	1.6969E+00	1.2006E+00	2.4056E+00
36	2.4876E+00	3.3985E-04	2.4873E+00	1.4546E+00	4.1155E+00
37	4.1065E+00	5.5766E-04	4.1060E+00	1.2898E+00	8.4765E+00
38	6.9477E+00	9.4050E-04	6.9468E+00	4.2188E+00	1.4569E+01
39	9.9593E+00	1.3466E-03	9.9580E+00	7.1122E+00	2.7266E+01
40	1.5488E+01	2.0936E-03	1.5486E+01	1.2229E+01	2.7757E+01
41	2.7224E+01	3.6800E-03	2.7221E+01	2.6092E+00	7.1962E+01
42	4.4572E+01	6.0268E-03	4.4566E+01	1.1657E-01	3.6901E+01
43	6.8790E+01	9.3086E-03	6.8780E+01	9.2593E-01	1.2415E+02
44	1.0655E+02	1.4432E-02	1.0653E+02	8.8051E+00	2.4662E+01
45	1.5548E+02	2.1077E-02	1.5546E+02	9.5070E+01	1.0304E+02
46	2.7459E+02	3.7268E-02	2.7455E+02	3.9756E+03	2.4696E+03
47	5.1773E+02	7.0355E-02	5.1766E+02	2.7331E+03	2.4936E+04