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NUCLEAR DATA AND LOW ENERGY NUCLEAR RESEARCH IN ISRAEL

Progress Report

compiled by

SHIMON YIFTAH

Israel Atomic Energy Commission

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The Israel Nuclear Data and Low Energy Nuclear Research relevant to the International NUclear Data Committee was continued in the various institutions listed in previous Progress Reports [Report for 1974, INDC (SEC) - 42/L, December 1974, Report for 1975, INDC (SEC) - 50/L, January 1976].

The major experimental facilities consist of:

1. A 5 Megawatt swimming pool enriched uranium reactor at the Soreq Nuclear Research Centre.
2. A 26 Megawatt heavy water tank-type natural uranium reactor at the Negev Research Centre.
3. A 6-million volt EN tandem accelerator at the Weizmann Institute of Science, Rehovot.
4. The new most modern high energy 14 UD pelletron accelerator manufactured by the National Electrostatic Corporation of Middleton, Wisconsin, installed inside the Koffler Accelerator Tower at the Weizmann Institute of Science, Rehovot.

Brief abstracts of the research work, both published and unpublished, listed according to the various laboratories, are reported in the following pages.

Isral Atomic Energy Commission Laboratories

ETGAR - INFORMATION SYSTEM FOR ABNORMAL OCCURRENCES IN NUCLEAR POWER PLANTS[1]

J. Baram, M. Nagar and G. Pultorak

Extensive information on the systems and components of a nuclear power plant are needed early in the planning stage, during the building of the plant, and during the licensing process. Another type of information helps preventive maintenance during the operating life of the plant. In the case of abnormal occurrences additional information on the possible consequences and ways of handling them is essential.

To cover these needs, the ETGAR system collects and evaluates information on abnormal occurrences in nuclear power plants. The information is coded, using a three-level coding scheme for systems and components, and put on magnetic tape. A search program permits the retrieval of any pertinent information from the data base. At present the ETGAR scope covers mostly PWR and BWR type nuclear power plants.

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1. Baram, J., Nagar, M. and Pultorak, G., Trans. Joint Annual Meeting of the Israel Nucl. Soc. and Israel Health Phys. Soc., Haifa, Nov. 30, 1975. 3, p. III-15, 1975.

PRACTICAL FORMALISMS FOR NUCLEAR DATA REPRESENTATION IN EVALUATED NUCLEAR DATA FILES IN THE UNRESOLVED RESONANCE ENERGY REGION[1]

Y. Gur and S. Yiftah

The currently used formalism for basic data representation in the unresolved resonances energy range is based on the statistical parameters of the population of Breit-Wigner resonances. The present work suggests new practical formalisms, based on parametric representation of the shielding factor curves, by which the values of effective cross sections may be obtained simply and quickly in the unresolved range. These formalisms were shown to be useful in existing codes to improve the accuracy and efficiency of the calculation of effective cross sections. Observed spatially dependent self-shielding factors were transformed into pseudo-background cross-section-dependent self-shielding factors, enabling the evaluator to use this experimental information. Numerical values of the transformation for U-235 and Pu-239 self-shielding factors were determined. Finally, as an example, Pu-239 basic data were represented in the unresolved range by one of the new formalisms.

REFERENCE:

1. Gur, Y. and Yiftah, S., Nucl. Sci. Eng., in press.

A SURVEY OF CROSS SECTION EVALUATION METHODS FOR HEAVY ISOTOPES[1]

M. Caner and S. Yiftah

Evaluation methods and neutron nuclear reaction theories applicable to heavy isotopes were considered. A compilation was made, in tabular form, of the most recent evaluations of the transactinium isotopes, with the exception of the main fissile and fertile isotopes (^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{238}Pu). The evaluations were examined in terms of the formalisms and models used in the different energy ranges.

Of the 18 isotopes for which evaluations are available, 11 were evaluated using optical model and statistical theory calculations (^{231}Pa, ^{233}Pa, ^{232}U, ^{236}U, ^{237}U, ^{238}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{244}Cm, ^{252}Cf) and the other 7 (mentioned below) were evaluated using systematic methods of varying degrees of sophistication. We checked the experimental data now available on these 7 isotopes in order to ascertain for which of them an optical model and statistical theory calculation has become feasible. The data needed are: excited energy levels, their spins and parities; fission cross sections; average capture widths and average level spacings. Of the 7 isotopes, enough data for the above calculations are available for ^{234}U, ^{237}Np, ^{241}Am, ^{243}Am and ^{245}Cm, but not enough for ^{238}Np and ^{236}Pu.

The next point considered was which isotopes, for which there are requests in WRENDA[2], have not been evaluated. These isotopes are: ^{239}U (σ_γ, σ_f requested), ^{239}Np (σ_γ requested) and ^{237}Pu (σ_γ, σ_f requested). However, there are not enough data available for any of these to perform optical model and statistical theory calculations. Only preliminary evaluations, based on systematics and the limited experimental data, are possible.

REFERENCES:

1. Caner, M. and Yiftah, S., "A Survey of Cross Section Evaluation Methods for Heavy Isotopes", contributed paper for the IAEA Advisory Group Meeting on Transactinium Isotope Nuclear Data, Karlsruhe, 3-7 November 1975.
2. WRENDA 74, INDC (SEC)-38/U (1974)

NEUTRON CROSS SECTIONS FOR PLUTONIUM-238[1,2]

M. Caner and S. Yiftah

An evaluation of ^{238}Pu neutron data was performed. All significant cross sections in the neutron energy range 10^{-3} to 15×10^{6} eV were considered.

The experimental data were complemented by spherical optical model and statistical theory calculations, and by systematics. The evaluated data were compared with those in the ENDF/B-IV file.

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1. Caner, M. and Yiftah, S., IA-1301, 1975, and IAEA distribution code INDC (ISL) - 2/L (July 1974, rev. Jan. 1975)
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STATUS OF TRANSACTINIUM ISOTOPE EVALUATED NEUTRON DATA IN THE ENERGY RANGE 10^{-3} eV TO 15 MeV(1)

S. Yiftah, Y. Gur and M. Caner

Large amounts of transactinium elements will be produced in the next 25 years in thermal power reactors, fast breeders, fast reactors, special purpose reactors, thermonuclear explosions and improved heavy-ion accelerators. In order to be able to evaluate, predict, compute and judge the effects and uses of these elements, the nuclear community needs fully evaluated nuclear data to be used as nuclear input for all computations and evaluations.

The sixteen transactinium elements and two hundred isotopes known to data were divided into three groups, and eight main areas of application were considered from which needs can be derived for new measurements and evaluations. Existing evaluations were tabulated and analyzed. From "WRENDA minus CINDA" descriptive equation, nine main conclusions and recommendations were presented, including a "world transactinium nuclear data evaluation program" and other specific items for future IAEA action in this field.

REFERENCE:

1. Yiftah, S., Gur, Y. and Caner, M., "Status of Transactinium Isotope Evaluated Neutron Data in the Energy Range 10^{-3} eV to 15 MeV", Invited paper presented at the IAEA Advisory Group Meeting on "Transactinium Isotope Nuclear Data", Karlsruhe, Germany, November 3-7, 1975.

SENSITIVITY OF THE LUMPED FISSION-PRODUCT MODEL TO DIFFERENT FISSION-PRODUCT CROSS SECTION LIBRARIES(1)

D. Ilberg, D. Saphier and S. Yiftah

The extent to which the use of different fission-product cross sections (FPCS) affects the neutron multiplication factor k_{eff} in high burnup cores of fast reactors was evaluated. It was found that discrepancies of the order of 2.5% exist when different FPCS are used to calculate k_{eff} in the same core. These discrepancies are due to the absence of data on a number of fission-product isotopes present in some of the nuclear data libraries on the one hand, and large differences in the capture cross sections of some of the isotopes on the other. A list of fission-product isotopes was proposed that, when used, reduces discrepancies in k_{eff} to <1%. The important isotopes for fast-reactor burnup and k_{eff} calculations in which large discrepancies exist were identified, and it is suggested that they be subjected to further evaluation to close the discrepancy gap.

REFERENCE:

1. Ilberg, D., Saphier, D. and Yiftah, S., Nucl. Sci. Eng. 58, 445 (1975)

ENERGY SPECTRA OF SECONDARY NEUTRONS FROM THE ^{238}U (n,2n) AND (n,3n) REACTIONS

M. Caner, M. Segev* and S. Yiftah

A consistent compound nucleus theory of (n,2n) and (n,3n) neutron emission was applied to ^{238}U in order to obtain the energy spectra of the second and third secondary neutrons. The evaluation was based on inelastic level excitation and evaporation data for ^{238}U, ^{237}U and ^{236}U. The ^{238}U and ^{236}U data were retrieved from ENDF/B-IV files; the ^{237}U data were evaluated by us using experimental information and statistical reaction theory codes.

At reaction energies E_o just above the (n,2n) threshold energy B_2, the energy E of the second inelastic neutron has a spectrum of (E_o-B_2-E); just above the (n,3n) threshold B_3, the third neutron energy has a spectrum of $(E_o-B_2-E)^3$. At energies E_o well above the thresholds the second and third neutron spectra approach the evaporation form.

A secondary neutron spectrum for any given reaction energy E_o is approximated by the composite form

$$P_i(E_o\to E) = \left[\frac{2(i-1)\beta_i(E_o)}{E_o-B_i}\right]\left(1-\frac{E}{E_o-B_i}\right)^{2i-3} + \left[\frac{1-\beta_i(E_o)}{T_i^2(E_o)}\right]E\exp\left(-\frac{E}{T_i(E_o)}\right)$$

where $i=2,3$ for the second and third neutrons, respectively. The temperatures T_i and blending coefficients β_i were calculated for several energies in the range from threshold up to 15 MeV for $i=2,3$; in addition, $\beta_1\equiv 0$ and $T_1=T_{U-238}$ (ENDF/B-IV).

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We compared the experimental effective temperature for all secondary neutrons from ^{238}U (n,n'), (n,2n) and (n,3n) at 14 MeV with the value we calculated (by performing a least squares fit to our data). Our results were found to be consistent with the experimental data.

COMPARISON OF METHODS FOR THERMAL REACTOR LATTICE CALCULATIONS

A. Schneider, W. Rothenstein* and E. Greenspan

The computer codes HAMMER[1] and WIMS-D[2] for solving the transport equation in thermal lattice cells were investigated. The values of the effective multiplication constant (k_{eff}) calculated by HAMMER are about 2% below the experimental values for both heavy and light water-moderated lattices. The k_{eff} calculated by the WIMS-D code are about 1.0% below and less than 0.5% above the experimental values for heavy water and light water lattices, respectively.

The approximations to the transport equation used by the two codes were investigated. Detailed comparisons of the group cross sections used by the codes and of the values of the average lattice parameters calculated by them were performed.

It was found that the discrepancies between calculated and experimental results are due mainly to inaccuracies in the cross section data, especially in the energy region of the resonances.

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1. Suich, J.E. and Honech, H.C., DP-1064, 1967.
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STUDY OF A TEMPERATURE DISTRIBUTION IN THE COOLANT MEDIUM OF A LIQUID METAL FAST BREEDER REACTOR[1]

M. Tilman

An analytical method of determination of a temperature distribution in the coolant medium in a fuel assembly of a liquid-metal fast-breeder reactor (LMFBR) was developed. The temperature field obtained was applied to a constant velocity (slug flow) fluid flowing parallel to the fuel pins of a square and hexagonal array assembly. The coolant subchannels contain irregular boundaries. The geometry of the channel due to the rod adjacent to the wall (edge rod) differs from the geometry of the other channels.

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The governing energy equation was solved analytically, assuming se .e solutions for the Poisson and diffusion equations, and the total solution w.s a superposition of the two solutions. The boundary conditions were specified by symmetry considerations, assembly wall insulation and a continuity of the temperature field and heat fluxes. The initial condition is arbitrary.

The method satisfies the boundary conditions on the irregular boundaries and the initial condition by a least squares technique.

Computed results were given for various geometrical forms, with the ratio of rod pitch-to-diameter typical for LMFBR cores. These results are applicable to various fast-reactors, and thus the influence of the transient solution (which solves the diffusion equation) on the total depends on the core parameters.

REFERENCE:

1. Tilman, M., NRCN-387, 1975.

THE EFFECT OF FUEL BURNUP ON THE DYNAMIC BEHAVIOR OF FAST REACTORS

D. Ilberg, D. Saphier and S. Yiftah

Performance of an accident analysis taking burnup (BU) changes into account requires fission product (FP) nuclear data of relatively small uncertainty, suitable BU calculation models, and point kinetics and dynamic computer programs. These were prepared and used in the present study[1] with the following results:

a) Significant changes in static and dynamic parameters were observed when investigating the effect of BU. These changes were found to be larger than differences introduced by the uncertainty of the FP nuclear data.
b) A one-dimensional BU computer program was prepared. It was found that a BU model based on the generalized radioactive decay scheme is most suitable for accurate fast reactor calculations.
c) Kinetic and space-time dynamic calculations of fast reactors having different BU levels were performed. The stability difference between "clean" and high BU cores is greater when local rather than uniform perturbations are inserted along the entire core length. The magnitude by which the "end-of-life" core increases the transient excursion over that of the "clean" core depends on the particular region in which the perturbation is inserted. The "end-of-life" core will magnify the transient excursion more than the "clean" core whenever the perturbation is inserted into a

region having a higher adjoint flux level than that of the "clean" core. It is suggested that the analysis of local perturbations be performed for "end-of-life" cores as well as for "clean" cores in the safety evaluation of fast reactors.

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1. Ilberg, D., Saphier, D. and Yiftah, S., Nucl. Sci. Eng. (1976), in press.

METHODS FOR THE OPTIMIZATION OF NUCLEAR SYSTEMS

D. Gilai, E. Greenspan, P. Levin and T. Tabak*

The maximum principle of Pontryagin for the optimization of one-dimensional systems has been extended to multidimensional systems(1). Table I summarizes the generalized formalism and compares it with the conventional one-dimensional principle. In the generalized formulation, $\underline{\underline{E}}$ is a matrix operator of first order partial derivatives. The resulting optimality conditions are found to be the same as those obtained from the perturbation theory approach(2). Using the generalized maximum principle, an algorithm for the optimization of the enrichment distribution in power reactors was developed and successfully tested using the diffusion approximation. The solution of a two-dimensional shield optimization problem was also illustrated.

TABLE I

Comparison between the generalized and one-dimensional maximum principle

	One-dimensional	Generalized
State equation	$\frac{d\underline{\phi}}{dx} = \underline{F} = \frac{\partial H}{\partial \underline{P}}$	$\underline{\underline{E}}\,\underline{\phi} = \underline{F} = \frac{\partial H}{\partial \underline{P}}$
Adjoint equation	$\frac{d\underline{P}}{dx} = -\frac{\partial H}{\partial \phi}$	$\underline{\underline{E}}^T \underline{P} = -\frac{\partial H}{\partial \underline{\phi}}$
Hamiltonian	$H = \pm g + \underline{P}\cdot\underline{F}$	$H = \pm g + \underline{P}\cdot\underline{F}$
Optimum condition	$H(\underline{U}^*, \underline{\phi}^*, \underline{P}^*) \geq H(\underline{U}, \underline{\phi}^*, \underline{P}^*)$	$H(\underline{U}^*, \phi^*, \underline{P}^*) \geq H(\underline{U}, \underline{\phi}^*, \underline{P}^*)$

*Ben-Gurion University of the Negev, Beer-Sheva

Efforts are currently being made to extend the range of applicability and efficiency of the perturbation theory-based transport code SWAN designed for the optimization of source driven systems.

REFERENCES:

1. Gilai, D. and Tabak, T., An extension of the maximum principle to multidimensional systems and its application in nuclear engineering problems, IEEE Trans. Automat. Contr., in press.
2. Lewins, J. and Babb, A.L., Optimum nuclear reactor control theory, in: Advances in Nuclear Science and Technology, Vol. 4, Academic Press, New York, 1968, p. 251.

A GENERALIZED SOURCE-MULTIPLICATION METHOD FOR DETERMINING REACTIVITY AND A SOURCE-MULTIPLICATION REACTIVITY

E. Greenspan

A source-multiplication method, capable of providing subcritical reactivities without the need for intercalibration against another independent reactivity measurement method (or against calculations) has recently been proposed(1). Such calibration is required by other source-multiplication (SM) methods. Instead, this method, designated the SMR method(1), calls for the measurement of the rate of linear power increase of the source-driven reactor when it is at a reference critical state. As previously derived(1) the SMR method can be applied only to slightly subcritical reactors; that is, to the determination of small reactivities. The reason for this restriction is that in the derivation we have expanded the subcritical reactor flux into normal modes and assumed that (a) the fundamental mode has the dominant contribution to the detector reading and (b) the fundamental mode of the subcritical reactor is similar to that of the critical one.

In a recent work(2) the SMR method is rederived in a general form [to be referred to as the generalized source-multiplication (GSM) method] that makes it applicable for the determination of absolute reactivities for any degree of reactor subcriticality. This is accomplished by relating the reactivity to the actual flux distribution in the source-driven subcritical reactor rather than to the fundamental mode. The resulting reactivity is referred to as the "source-multiplication (SM) reactivity." The relationship between the SM and other definitions of reactivity in common use was also established(3).

Several aspects of the practical implementation of the GSM method were discussed(2). The GSM method was compared(2) with the modified source multiplication (MSM) and other methods for reactivity measurement. The fundamental difference between the GSM and the MSM methods is in the calibration of the reactivity scale; the MSM method requires intercalibration against an independent reactivity-determination method. The two methods may also differ in the definition of the correction factors needed. These depend on the definition of the calibrating reactivity (in the MSM method) and of the reactivity to be determined[3].

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1. Greenspan, E., J. Nucl. Energy 27, 129 (1973); also, Trans. Amer. Nucl. Soc. 15, 456 (1972)
2. Greenspan, E., Nucl. Sci. Eng. 56, 100 (1975)
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ENERGY-DEPENDENT FINE-STRUCTURE EFFECTS ON MATERIAL AND DOPPLER REACTIVITY WORTH

E. Greenspan and Y. Karni

The effects of perturbations in the fine-structure of the neutron spectrum on the reactivity worth of resolved resonances were investigated, using a simple, space independent model, amenable to analytic solution. It was found[1] that these spectral fine-structure effects can contribute significantly to the reactivity worth of resonances. The multigroup perturbation theory methods commonly used do not adequately take these effects into account. Consequently, the reactivity worth associated even with an infinitesimal change in the amplitude of a resonance (associated with a change in material density) is overestimated in multigroup perturbation theory calculations by up to a factor of 2 [1]. Similarly, the reactivity worth associated with a change in the temperature of the resonance can be either overestimated or underestimated by as much as a factor of 2 even for an infinitesimal temperature change[2,3]. A clear correlation was found between the spectral fine-structure effects and the discrepancy between the calculated and experimentally determined material and Doppler worth observed for over a decade in fast assemblies.

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2. Karni, Y. and Greenspan, E., Trans. Amer. Nucl. Soc. 21, 494 (1975)
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DEVELOPMENTS IN GENERALIZED PERTURBATION THEORY

E. Greenspan

The generalized perturbation theory (GPT) formulations were extended to multiple ratios of linear functionals[1] as well as to composite functionals[2].

A composite functional has the general form

$$C_{IJK} = \prod_{i=1}^{I} \langle S_i^+,\phi\rangle^{P_i} \prod_{j=1}^{J} \langle \phi^+,S_j\rangle^{P_j} \prod_{k=1}^{K} \langle \phi^+,M_k\phi\rangle^{P_k}$$

where P_i, P_j and P_k can take the value of either +1 or -1. In another extension, GPT formulations were derived for functionals of eigensolutions of the time-absorption (α) eigenvalue equations[3].

A unified formulation and terminology for GPT was proposed[4]. It was shown that Stacey's[5] and Usachev-Gandini's versions of GPT[6,7] can be derived using conventional perturbation techniques. It was also shown that their formulations of GPT are but two of many possible versions; these versions are distinguished by the criticality reset mechanism to which they correspond. Each version has its own range of applicability. Finally it was shown that the perturbations in the distribution functions can be taken into account in two different forms, in terms of generalized functions (the conventional form) or in terms of perturbations in distribution functions. The more efficient form for a given application is problem-dependent.

Other developments in GPT include the derivation of GPT for (a) reactivity in the integral transport theory formulation[8], and (b) calculating the effect of changes in input parameters on a given integral property in an altered system[9]. All the developments mentioned above are summarized in Ref. (10).

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7. Gandini, A., J. Nucl. Energy 21, 755 (1967)
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9. ibid, p. 62.
10. Greenspan, E., Developments in perturbation theory in: Advances in Nuclear Science and Technology, in press.

PERTURBATION THEORY AND IMPORTANCE FUNCTIONS IN INTEGRAL TRANSPORT FORMULATIONS[1]

E. Greenspan

Perturbation theory expressions for the static reactivity derived from the flux, collision density, birth-rate density and fission-neutron density formulations of integral transport theory, and from the integro-differential formulation, were intercompared. The physical meaning and the relation of the adjoint functions corresponding to each of the five formulations were established. It was found that the first-order approximation of the perturbation expressions depends on the transport theory formulation and on the adjoint function used. The approximations of the integro-differential formulation corresponding to different first-order approximations of the integral transport theory formulations were identified. It was found that the accuracy of all first-order approximations of the integral transport formulations examined is superior to the accuracy of first-order integro-differential perturbation theory.

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1. Greenspan, E., Nucl. Sci. Eng., in press.

EXACT SOLUTION OF P_n SPACE-TIME DEPENDENT EQUATIONS WITH TIME-DEPENDENT CROSS SECTIONS FOR SLAB GEOMETRY[1]

M. Lemanska

Monoenergetic, space-time dependent P_n equations with time-dependent cross sections were solved using generalized Lie series. The variables were separated and the solution obtained. This solution was given in a closed form for the case of time-independent cross sections. Numerical results calculated for the P_3 approximation case were in good agreement with those obtained by the time-dependent S_4 code.

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1. Lemanska, M., J. Appl. Math. Phys. 26, 701 (1975)

NUMERICAL SOLUTION OF THE TRANSPORT EQUATION BY COLLOCATION WITH BIVARIATE SPLINES[1]

L. Finkelstein and A. Krumbein

A class of partial differential equations, directly connected with the transport equation was considered. It was shown that if the initial-boundary conditions are specified on a given net as univariate quadratic splines, then there exists a bivariate quadratic spline unique on the net, which satisfies exactly the initial-boundary conditions and satisfies the differential equation at the nodes of the net. The spline is then constructed by an exact finite-difference scheme. As a first application we provided a new algorithm for a spherically symmetric problem in neutron transport theory. This was further illustrated by numerical examples. Preliminary results were reported in Refs. (2) and (3).

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LATTICE STUDIES FOR NATURAL URANIUM FUELLED HEAVY-WATER MODERATED FUSION-FISSION HYBRID REACTORS

E. Greenspan and A. Schneider

Preliminary assessment of the performance of natural uranium fuelled, heavy water or graphite moderated blankets for fusion - fission hybrid reactors indicated[1] that the performance of such blanket concepts might be significantly superior to that of the corresponding fission reactor cores. To check the validity of our assessment[1] and to quantify it, we have initiated an investigation of the physical characteristics attainable from the lattices under consideration. The first series of calculations were performed with the lattice code HAMMER[2] and version II of the ENDF/B cross section files. The uranium oxide fuel (10.45 g cm^{-3}) is clad with aluminum (substituting Zircaloy owing to availability of cross sections) 0.04 cm thick. The heavy water contains 0.3% H_2O.

Figure 1 summarizes the relationship between the initial conversion ratio (CR) and the effective multiplication constant (k_{eff}) calculated for clean and cold (20°C) lattices. This k_{eff} is calculated for a buckling of 0.00007 m^{-2}. Also shown in Fig. 1 is the volumetric power density attainable with the lattices considered (as a function of their k_{eff}) relative to that of Pickering[3]. It is taken to be the smaller of the relative changes in either the total length or the surface area of the fuel rods per unit blanket volume.

The results of these calculations confirm our preliminary assessment that natural uranium heavy water blankets can be designed to be good breeders, and at the same time, have high power densities.

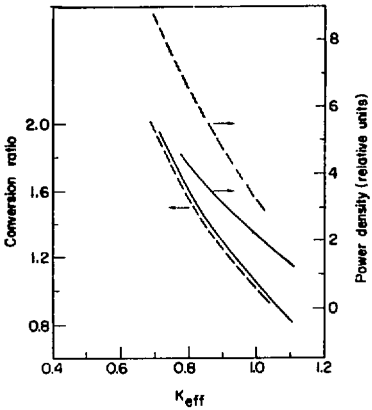

Fig. 1

Initial conversion ratio and volumetric power density
for single rod natural uranium heavy water lattices
_ _ _ _ 0.40 cm radius; _____ 0.72 cm radius

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FISSION-FUSION SYMBIOSIS IN HIGHLY COMPRESSED MICROSPHERES[1]

A.D. Krumbein

In previous work[2], the feasibility of producing a power source or a copious source of neutrons by the compression of microspheres of fissionable material was investigated by means of numerical calculations. The results for pure fission sources with and without a moderating blanket led to the conclusion that only a coupling between fission and fusion, producing a species of chain reaction, could make the concept feasible.

As a first approximation to such a system, an external source of neutrons from either the D-D or D-T reactions was introduced into the calculations. This fusion neutron source was computed as a function of temperature using the expressions in Lovberg[3] above 1 keV and power series in temperature below

1 keV to match the values of $<\sigma v>$ quoted by Thompson[4]. The temperature change with time was made to depend on the energy released in the fission process. The initial ion temperature distribution used in the calculations was taken from recent computational results reported by Brueckner[5] as occurring in an imploded sphere just before the fusion process begins.

The calculations were performed with five different mixtures of ^{239}Pu and deuterium, with volume fractions of plutonium ranging between 0.25 and 0.95. Spheres ranging in size from 0.005 to 0.40 cm in radius were included in the calculations. The three-region configuration of varying density[2], deduced from the calculations of Brueckner and Nuckolls, was used exclusively in these calculations. The neutron multiplication was computed as a function of time using the SNT[6] time-dependent neutron transport code to determine if a "runaway" neutron multiplication could be achieved. In none of the calculations, however, was an exponential increase in neutron multiplication, characteristic of such a runaway process, observed.

The above model, however, takes no account of thermal conduction in the plasma and is, therefore, in a sense conservative. A more detailed approach is now being implemented in which consideration is given to thermal conduction by electrons and ions in the plasma as well as the transfer of energy from electrons to ions. In this way, the energy produced by fission in compressed fissionable material can be transferred, for example, to the D-T portion of a layered configuration.

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STUDY OF U-238 BLANKETS FOR LASER FUSION REACTORS*[1]

A.D. Krumbein, M. Lemanska, Y. Gur and S. Yiftah

The economic feasibility of any laser-fusion process may depend critically on the inclusion of a blanket which breeds fissile fuel[2,3]. Such a blanket would contain ^{238}U or ^{232}Th and capture fusion neutrons to produce ^{239}Pu or ^{233}U.

*This work is partially supported by the U.S.-Israel Binational Science Foundation.

We have computed some of the properties of such a blanket using, as a model, a 14 MeV neutron source in vacuum surrounded by a spherical blanket containing ^{238}U. The SNG[4] spherical transport code was suitably modified for use in the calculations.

The results obtained to date for the number of fission, absorption and neutron multiplication events per 14 MeV source neutron give good agreement with Weale's[5] experimental results. The calculations are now being extended to study the effects of anisotropic scattering as well as that of the energy distribution of the neutrons produced in the (n,2n) and (n,3n) reactions. In addition, the sensitivity of the results to changes in neutron cross sections is under investigation. Work has likewise been undertaken to calculate blankets containing various coolants and structural materials.

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A MULTIGROUP MODEL FOR THE SLOWING-DOWN OF ENERGETIC IONS IN PLASMAS[1]

E. Greenspan and D. Shvarts

The energy dependent equation for the slowing-down of energetic ions in fully ionized plasma was cast into a multigroup form. The resulting multigroup equation is a linear equation for the flux of the energetic ions and is easy to solve. It can take into account large energy transfer reactions preserving their discrete nature. The sensitivity of the multigroup approximation to the group width and weighting function used for generating group constants was investigated. It was found that few groups with simple, problem independent weighting functions can yield results of reasonable accuracy. The relation between the multigroup and other methods for the solution of the slowing-down equation was considered.

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EFFECTS OF LARGE ENERGY TRANSFER REACTIONS ON SLOWING-DOWN PROPERTIES[1,2]

D. Shvarts and E. Greenspan

Large energy transfer (LET) reactions affect the slowing-down process via two mechanisms: a) discrete energy losses of the energetic ions leading to a faster slowing down process, and b) the production of energetic recoil ions from the bulk plasma that might also interact with the bulk plasma during slowing down.

The effects of LET reactions on the following plasma properties were investigated: a) the probability that high energy ions will cause fusion while slowing-down, b) the transfer of energy from the energetic ion to the bulk plasma ions and electrons, and c) the slowing-down time. Plasma parameters calculated with the MUGLET code, which takes into account LET interactions were compared with a continuous slowing down model (CSD). It was concluded that:

a) LET reactions have a non-negligible effect on the slowing down process for plasma electron temperatures above 10 keV and initial ion energies in the MeV range.
b) For high electron temperatures ($T_e \gtrsim 50$ keV) and high initial energies the fusion probability of a deuteron in a DT plasma is about two times greater when accounting for LET reactions, compared with the CSD model. This enhancement is mainly due to the contribution of the recoils.
c) With LET reactions taken into account, the bulk plasma ions get a larger fraction of the beam ion energy than predicted by the CSD model. The difference may reach a factor of two.
d) The slowing down time with LET reactions is shorter as compared with the CSD results. The inclusion of the recoils in the calculation may reduce the slowing down time even further.

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SPECTRUM HARDENING IN TWO-COMPONENT-TOKAMAKS

E. Greenspan and D. Shvarts

In the calculation of the slowing-down process in two-component-tokamaks (TCT) it is customary to assume that the beam ions slow down against bulk plasma ions and electrons having a maxwellian distribution. When the number of the slowing-down (or suprathermal) ions is not negligible compared with the number of thermal ions, however, the contribution of interactions between the suprathermal ions (i.e. beam-beam interaction) to the plasma properties may not be negligible. This contribution can be taken into account by describing the distribution of plasma ions, for the calculation of the slowing-down of the beam ions, as a superposition of a maxwellian and a slowing-down distribution (i.e., by considering a hardened spectrum).

Self-consistent calculations of the slowing-down process for plasma conditions typical to TCT devices were performed. The solution of the slowing-down equation was reached iteratively, using a modified version of MUGLET[1]. Starting with a maxwellian distribution for the bulk plasma ions, the distribution of the suprathermal ions was calculated. By superposing the maxwellian and the slowing-down spectrum thus obtained, the solution of the slowing-down equation was repeated (now against a hardened maxwellian) and the bulk plasma ion distribution accordingly adjusted.

For example, in the case of a 440 keV deuteron beam injected into a D-T plasma of $T_i = T_e = 20$ keV and $n = 10^{14}$ cm^{-3}, it was found that whereas the probability that a beam ion will undergo fusion while slowing down in a maxwellian plasma is about 0.004, the hardening of the spectrum, obtained when the beam ions constitute 25% of the total plasma ions, enhances the fusion probability by up to 0.006.

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NUCLEAR SCATTERING CROSS SECTION FOR FUSION CHAIN REACTION CALCULATION - THE (D, ^{3}He) CYCLE

Y. Karni, E. Greenspan and D. Shvarts

It has been shown that nuclear scattering may lead to a fusion chain reaction[1]. This is due to the fact that there are large energy transfer collisions which may generate a shower of fast recoils that undergo fusion reactions while slowing down. Such a chain has been shown to be self-sustaining under some conditions for the DT cycle[1]. To study the feasibility of a chain reaction for the D^{3}He cycle, we first set out to find the differential cross sections for nuclear scattering among the various species (D, ^{3}He, P, α) of this cycle.

To obtain an evaluated set of nuclear scattering cross sections a least-squares calculation was performed on a large number of published experimental data in the energy range up to 15 MeV with scattering angles of 20^{o}-160^{o}.

A similar approach for the DT cycle used phase shift analysis [2] or some arbitrary polynomial function [3]. We found the latter method to be easier to apply and reasonably accurate. For example, for p-D scattering we used the function:

$$\sigma(E,\theta) = \frac{\eta}{E^{2}\cdot(1-\cos\theta)^{2}} + \sum_{j=0}^{4}\sum_{m=0}^{3} a_{j,m}\, E^{m}\, (\cos\theta)^{j}$$

Taking 1300 data points from 19 references, a fit with an average relative deviation of 16% was obtained. Similar functions were used for the other scattering reactions.

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TDMG - A TIME-DEPENDENT, MULTIGROUP, ZERO-DIMENSIONAL CODE FOR PELLET FUSION STUDIES IN THE PRESENCE OF CHAIN REACTIONS

D. Shvarts

Recent stationary calculations [1] have shown that under suitable conditions fusion burn can proceed via fusion chain reactions. This is due to nuclear scatterings of the fusion burn particles that produce a shower of fast ions some of which undergo further fusion reactions during their slowing down. The purpose of the code developed is to study the dynamics of fusion burn via usual thermal reactions and chain reactions. Since fusion chain reaction proceeds via suprathermal ions, it is desirable, for a first study of the effects, to represent the energy dependence accurately, whereas the space dependence may be represented in an approximate manner.

The main physical assumptions and models of the present code are:

1. Point reactor model. We assume an homogeneous medium having the same temperature and density everywhere. Leakage phenomena due to the finite dimensions are considered by using approximate diffusion coefficients.

2. A multi-group, multi-species description of the suprathermal ion population. This is based on the multigroup formalism recently developed[2], accounting for coulomb friction processes as well as for large energy transfer reactions (due to nuclear scatterings) and nonthermal fusion reactions.
3. Two-temperature description for electrons and ions. The energy balance equations account for the following processes: energy transfer from fast ions to the bulk plasma ions and electrons; Bremsstrahlung radiation losses of the electrons; energy exchange between the bulk ions and electrons; expansion cooling.
4. Because hydrodynamic expansion is a dominant mechanism in the burn process, we have included its effect in an approximate manner, using the similarity assumption.

The code was checked against some published results of calculations of thermal burning of bare DT spheres with sophisticated codes[3], and the time dependent energy yield was found to be reasonably accurate.

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COMPUTATIONAL EXPERIMENTS WITH A ^{252}Cf SOURCE FOR POSSIBLE USE IN NEUTRON RADIOGRAPHY[1]

D. Kedem and M. Lemanska

The influence of the moderators Fe, Al, C, D_2O, Be and CH_2, and the reflectors, H_2O and C, on the thermal neutron flux was examined. Spherical assemblies with a ^{252}Cf source located at the center and at various distances from the center were considered. A configuration having a small thermalization factor and a flat thermal neutron flux, adaptable to neutron radiography, was obtained.

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NUCLEAR PHOTOEXCITATION(1)

R. Moreh

The various methods of nuclear photoexcitation using neutron capture γ-rays were considered. In particular, the photoexcitation of isolated nuclear levels by the method of random overlap between one of the incident lines of neutron-capture γ-rays and a nuclear level of the scatterer were illustrated. The extraction of nuclear spectroscopic data such as spin, parity and the width of the resonance levels were explained. The variation of the energy of the photons using both nuclear resonance scattering and Compton scattering were explained and the use of both methods for measuring the ground-state radiative widths of nuclear levels were illustrated. In addition, the photoexcitation of the continuum region using higher energy photons in the 9 MeV - 11.4 MeV region was considered. It was thus possible to study the low-energy branch of the giant dipole resonance and the nuclear Raman scattering and to test the theoretical predictions in this field. The effect of Delruch scattering of these photons was mentioned.

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THE 6.324 MeV HOLE STATE IN ^{15}N(1)

R. Moreh and O. Shahal

The 6.324 MeV level of ^{15}N was photoexcited by a chance energy overlap with a γ line obtained from the Cr(n,γ) reaction using thermal neutrons. By measuring the angular distribution of the scattered radiation and its polarization, the spin and parity of the level was determined to be $J^{\pi}=3/2^{-}$. The E2/M1 mixing ratio of the $3/2^{-} \rightarrow 1/2^{-}$ ground state transition was unambiguously determined to be X(E2/M1) = +0.137±0.005. An upper limit to the branching ratio of the decay of the level to lower lying levels in ^{15}N was obtained. The radiative width of the level was also measured and found to be Γ = 3.1±0.3 eV. The result was compared with theoretical predictions.

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ATTENUATION COEFFICIENTS OF γ-RAYS AT 9.00 AND 11.39 MeV

T. Bar-Noy and R. Moreh

Precise measurements of attenuation coefficients at 9.00 and 11.39 MeV in 10 elements between Be and U were carried out. The γ beam was obtained from the (n,γ) reaction on disks of metallic nickel. The results are given in Table 1, and are found to be much higher than the calculated values of Refs. 1 and 2. The systematic differences are very probably due to the fact that the calculated cross section for pair-production is underestimated.

TABLE 1

Measured total attenuation coefficients (mb/atom) compared with calculated values for photon energies 9.00 and 11.39 MeV

	E_γ=9.00 MeV			E_γ=11.39 MeV		
Element	Ref. 1	Ref. 2	Present work	Ref. 1	Ref. 2	Present work
Be	256.0	254.6	257.3±2.2	228.1	226.8	238.5±7.6
C	406.9	405.1	413.5±19.2	369.9	367.7	395.0±0.7
V	2310	2280	2322±4	2300	2277	2368±26
Fe	2751	2719	2729±13	2768	2743	2787±22
Ni	3067	3030	3068±18	3101	3074	3175±10
Zn	3402	3354	3434±19	3459	3420	3537±9
Ag	6724	6694	6865±16	7050	7007	7334±28
W	13793	13716	14170±13	14697	14631	14950±60
Pb	16265	16109	16520±33	17307	17255	18000±64
U	19513	19489	20040±109	20808	20801	22200±31

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ABSORPTION OF 6.42 MeV PHOTONS[1]

R. Moreh and Y. Wand

Precise measurements of the total attenuation coefficients of 6.418 MeV photons in 23 elements between Be and U were carried out with uncertainties which were generally less than 0.3%. The method utilizes resonance scattering of monochromatic photons obtained from thermal neutron capture in titanium. The resonant scatterer is ^{139}La; it serves as an analyzer of the γ beam energies passing through the absorber. The effective energy definition in this method is of the order of 20 eV. The measured coefficients were generally in close agreement ($\sim$0.6%) with calculated values. A significant deviation was observed only in the case of Ta.

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A NEW HIGH RESOLUTION GAMMA RAY MONOCHROMATOR[1]

R. Moreh, I. Jacob and R. Mourad

A high-resolution gamma monochromator having a resolution of $\sim 10^{-6}$ was developed. The variable energy γ source is obtained by nuclear resonance scattering of neutron capture γ-rays through various scattering angles. Several possible examples of combinations of γ sources and resonance scatterers were considered. In particular, a lead target was employed to scatter the 7.28 MeV γ line of neutron capture γ-rays of iron. Variation of the angle of the resonantly scattered 7.28 MeV photons between 60°-150° permits an energy scan of 400 eV in any absorber. Thus, nuclear energy levels in some absorbers were photoexcited and the corresponding ground-state widths were extracted from the measured absorption spectrum. The results for the case of a Ce absorber were considered in detail.

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A NEW INTEGRATED TARGET-ION SOURCE FOR ISOTOPE SEPARATION WITH THE SOLIS

S. Amiel, G. Engler, E. Ne'eman, Y. Nir-El and M. Shmid

As part of the research program on mass and charge distribution in fission, the SOLIS on-line isotope separator is used to separate isotopes for measurements of half-lives, fission yields, nuclear decay properties and delayed neutron emission probabilities of short-lived isotopes. A new surface ionization integrated target-ion source was developed. The present version is an improvement over a previously described system[1]. The modifications were aimed at achieving higher temperatures and greater operational reliability. The target, which contains 1 g of uranium coated on specially arranged graphite foils, presents a large area for efficient and rapid release of products and a large cross section to match a broad neutron beam. The target is situated in an oven, heated by electron bombardment. The species released from the target are conducted through a heated tube and filament where surface ionization takes place.

The ion source operates in the temperature range of $\sim$1800°C which results in very short diffusion half-times i.e. about 1 sec for Rb isotopes and about 5 sec for Cs isotopes. This permits measurements of half-lives of isotopes down to about 0.1 sec.

Using this new source isotopes of Rb, Sr, Cs and Ba were separated by positive surface ionization and their half-lives measured using beta activity detected by a silicon surface barrier detector. The results are given in Table 2. Two isotopes ^{147}Ba and ^{148}Ba were identified for the first time and their half-lives measured. Other results verify in some cases previous determinations or improve values which were obtained with less reliable statistics.

TABLE 2

Half-lives of isotopes studied, sec

Mass	Element	Half-life	Mass	Element	Half-life
94	Rb	2.81 ±0.01	143	Cs	1.78±0.01
	Sr	76.6 ±1.0		Ba	15.17±0.38
95	Rb	0.402±0.008	144	Cs	1.02±0.03
96	Rb	0.225±0.012		Ba	11.85±0.57
	Sr	1.103±0.022	145	Cs	0.65±0.04
	Y	6.32 ±0.18		Ba	3.79±0.19
97	Rb	0.181±0.010	146	Ba	2.14±0.37
	Sr,Y	0.865±0.028		La	8.5 ±1.9
98	Rb	0.098±0.018	147	Ba	0.72±0.07
	Sr	1.04 ±0.11		La	4.43±0.54
			148	Ba	0.47±0.20
				La	2.62±0.61

It is planned to extend the research to study nuclear level schemes and to measure fission yields. The target-ion source is to be adapted for negative surface ionization which will make it possible to extend the research to isotopes of Br and I.

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DISTRIBUTION OF NUCLIDES IN FAST FISSION OF ^{232}Th

T. Izak-Biran and S. Amiel

An analysis of the independent fission yields of thermal fission of ^{235}U (1) revealed an enhancement of products with an even number of protons and a decrease in the number of products with an odd number of protons. This odd-even effect is now one of the important contributions to the understanding of the fission process, especially from saddle to scission. The odd-even effect was not analyzed in other fission nuclides because there are no measurements of independent fission yields available in the literature.

It was decided to measure independent yields in fast fission of ^{232}Th, as Th , like U , has an even number of protons and it is divided into 2 even fragments or 2 odd fragments. Only the ^{134}I and ^{135}I (2) fission yields of ^{232}Th have been reported In this work the independent yields of ^{90}Kr, ^{91}Kr, ^{139}Xe, ^{140}Xe, ^{131}Sn, ^{132}Sn and ^{132}Sb were measured directly and the yields of ^{91}Rb, ^{140}Cs, ^{131}Sb, ^{132}Sb, ^{134}Te and ^{135}Te were determined indirectly. The yields were obtained from the gamma spectra of the isotopes that were measured at different times, from 20 sec after the end of irradiation up to a few hours.

Kr and Xe were extracted by a stream of He and their gamma spectra were measured. From these gamma activities and from the known gamma intensities and efficiency of the detector, the independent yields of these isotopes were calculated. The yields of Rb and Cs were obtained by subtracting the measured yields of Kr and Xe from the chain yields which are known from the literature. The method was checked for accuracy and reliability by taking similar measurements in thermal fission of ^{235}U. Excellent agreement between the measured yields and the yields cited in the literature was obtained.

Tin was separated chemically as a hydride and its gamma rays were measured. By comparing the activities measured in ^{232}Th with those that were measured under the same conditions in thermal fission of ^{235}U, and by knowing the tin yields in ^{235}U from the literature, tin yields in ^{232}Th were obtained. Antimony yields were obtained in the same way as those for Rb and Cs. In addition, ^{132}Sb separated as a hydride was measured directly from the growth and decay of its gamma spectra.

The odd-even effects of these isotopes were obtained by comparing the measured yields with the calculated "normal" yields. The average odd-even effect in ^{232}Th was 31%±13%. For the elements Kr, Xe, Sb and Te an effect of 31%±5% was revealed and an excellent fit was obtained for the complementary elements Kr and Xe. For ^{132}Sn, the effect is much higher than the mean value of ^{232}Th because ^{132}Sn has a closed shell of neutrons, N=82. The high odd-even effect that was obtained in ^{232}Th is in accordance with the mass distribution of this nuclide that reveals a fine structure and is in accordance with theoretical works of calculated fission barriers that also reveal a fine structure in ^{232}Th.

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RATIOS OF INDEPENDENT YIELDS OF KRYPTON AND XENON ISOTOPES TO THE CUMULATIVE YIELDS OF THEIR PRECURSORS IN THE THERMAL NEUTRON FISSION OF ^{235}U

H. Feldstein and S. Amiel

Krypton and xenon isotopes formed in the thermal neutron fission of U-235 were released from an uranium oxide - barium stearate target and mass separated by SOLIS (Soreq on-line isotope separator). The contamination of the separated isotopes by their precursors (iodine and bromine) was about 2%, and cross contamination by adjacent mass was less than 5%. The separated isotopes were beta counted during their accumulation in the collector and the yield ratio of the separated isotope to the cumulative yield of its precursor was calculated from the beta activity curve measured as a function of irradiation time (growth curve). The analysis of the growth curve was performed with the aid of the least squares computer program ON LINE assuming known half-lives of the separated isotope and its precursor, with correction for transfer time using an experimental transfer time function. Alternatively, an analysis of the exponential components of the growth curve was performed for the cases in which the transfer time could be neglected (CLSQ program). In the above calculations, a steady ion current is assumed, but otherwise the method does not impose any restrictions and is independent of counter efficiencies since only one isotope is counted. The basic equation for the calculation is (disregarding the transfer time function):

$$A = K[1-\exp(-\lambda t)] + K_p[1 - \frac{\lambda}{\lambda-\lambda_p} \exp(-\lambda_p t) + \frac{\lambda_p}{\lambda-\lambda_p} \exp(-\lambda t)]$$

where A is the measured activity, K is the constant rate of formation of the separated isotope, K_p is the rate of formation of the precursor, λ is the decay constant of the separated isotope, λ_p is the decay constant of the precursor. The results of the calculations are summarized in Table 3 and are in good agreement with other experimental results.

TABLE 3

Krypton and xenon yields in thermal neutron fission of ^{235}U

Isotope	Chain yield[1] %	Yield ratio*	Independent yield of noble gas, %	Cumulative yield of halogen, %	Other experimental values Noble gas	Halogen
Kr-87	2.55 ± 0.07	$3.87^{+0.1}_{-0.5}$	$0.507^{+0.06}_{-0.01}$	$1.96^{+0.01}_{-0.06}$	0.36 ± 0.03[2] 0.51 ± 0.06[3]	2.0 ± 0.22[3]
Kr-88	3.62 ± 0.07	2.07 ± 0.5	$1.16^{+0.22}_{-0.16}$	$2.405^{+0.16}_{-0.23}$	1.32 ± 0.1[2] 1.66 ± 0.11[3]	1.86 ± 0.44[3]
Kr-89	4.80 ± 0.10	-	3.40 ± 0.5	1.17 ± 0.5	3.26 ± 0.19[2] 3.41 ± 0.15[3]	2.32 ± 0.4[1] 1.07 ± 0.11[3]
Xe-137	6.26 ± 0.16	$2.7^{+0.2}_{-0.5}$	$1.726^{+0.245}_{-0.10}$	$4.66^{+0.10}_{-0.245}$	$2.97^{+0.4}_{-0.32}$ [2]	1.81 ± 0.4[4]
Xe-138	6.80 ± 0.17	0.196 ± 0.05	$5.39^{+0.28}_{-0.16}$	$1.06^{+0.16}_{-0.28}$	$5.03^{+0.62}_{-0.50}$ [2]	0.84 ± 0.17[4]
Xe-139	6.5 ± 0.12	0.094 ± 0.02	$4.67^{+0.08}_{-0.08}$	$0.44^{+0.08}_{-0.08}$	$5.16^{+0.5}_{-0.4}$ [2]	0.47 ± 0.25[4]

*Ratio of the cumulative yield of the precursor to independent yield of the separated isotope

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SYSTEMATICS OF DELAYED NEUTRON EMISSION PROBABILITIES IN MEDIUM MASS NUCLIDES (FISSION PRODUCTS)

Y. Nir-El and S. Amiel

According to the delayed neutron emission probability formula the various types of delayed neutron precursors exhibit a systematic behavior governed by the level density, the excitation energy and the pairing of the precursor nucleus.

The updated population of precursors has recently been increased and now includes 45 precursors with relatively well-determined emission probability values. This permits revealing of group features with good precision. The systematic behavior was found to be determined by the nuclear pairin and the mass region of the precursor. The derivation of the systematics is based on a simplification of the general formula of the emission probability. The comparison made with the available experimental data leads to a semi-empirical formula for delayed neutron probabilities. This formula was used for the prediction of unknown values of emission probabilities for unidentified precursors.

EXCITATION ENERGIES AND PROMPT NEUTRON YIELDS IN LOW-ENERGY FISSION

Y. Nir-El and S. Amiel

Excitation energies and prompt neutron yields of primary fragments were calculated for the fission of ^{236}U induced by thermal neutrons. Excitation energies were calculated using the relation between excitation energy and nuclear temperature, assuming a thermal equilibrium at the moment of scission. A semi-empirical calculation of prompt neutron yields was based on prompt neutron and gamma ray emission data. The Jackson evaporation model was used to calculate theoretical neutron yields and the results were compared with experimental data. Further improvements of these calculations are in progress.

PROMPT NEUTRONS IN FISSION

Y. Nir-El and S. Amiel

A linear relation between the prompt neutron yields from fission and fragment masses was obtained using gaussian mass distributions of the primary (pre-) and secondary (post-neutron emission) fission fragments. This relation discloses a link between the trend of variation of prompt neutron output as a function of fragment mass and the widths of the primary and secondary gaussians. If the secondary gaussian is wider (narrower) than the primary one, then the number of prompt neutrons decreases (increases) with the mass of the emitting fragment. The comparison was made between the primary gaussian and the secondary "clean" one by excluding the dispersion of prompt neutrons which broadens the final distribution of the secondary fragments.

By using experimental values for the standard deviations of the two gaussians in the case of cesium fission fragments produced in $^{235}U(n_{th},f)$, it was found that the secondary gaussian is narrower by 8% than the primary one.

IMPLANTATION OF RADIOACTIVE ATOMS BY α-RECOIL

S. Abrashkin and N.H. Shafrir*

The aim of this work was to explore the possibility of implanting radioactive atoms in metals, to depths of up to 500 Å, in order to study processes characterized by a low rate of material removal (wear in hydrodynamic lubrication, corrosion, etc.). High-activity beams of radioactive atoms having well-defined energies and geometry can be implanted with special accelerators, but few such accelerators are available. Alternatively, alpha-radioactive sources can be used as sources of radioactive atoms having kinetic energies of about 100 keV and atomic masses in the vicinity of 220. Offsetting this narrow range in mass and energy are the availability and low cost of suitable radioactive sources and the flexibility in implantation conditions made possible by their use. The implantation takes place in the low energy region (L.S.S.'s ε = 0.03 to 0.08)(1).

Practically all previous work on alpha recoil stopping made use of gaseous, mostly light targets. When targets are of higher Z crystalline materials, the penetration profile is broadened by channeling effects(2), a large range straggling and higher ratios of projected-to-true ranges. While less amenable to theoretical interpretation, these broadened profiles are more suitable for measurement of surface wear.

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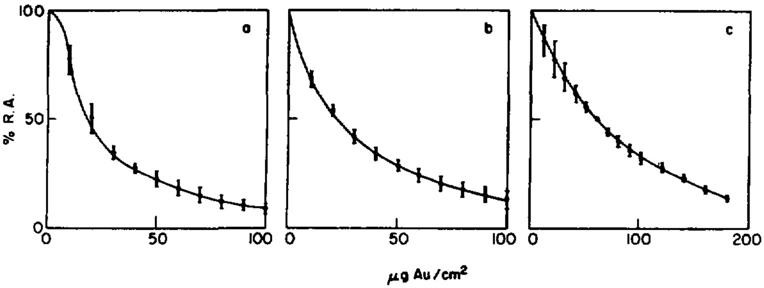

Fig. 1

Percent remaining activity (% R.A.) as a function of total thickness removed (µg Au/cm^2). Mean and standard error of 7,5, and 4 experiments for (a), (b) and (c) respectively

(a) % R.A. of ^{224}Ra, parallel beam implantation

(b) % R.A. of ^{224}Ra, isotropic implantation

(c) % R.A. of ^{212}Pb, implanted through 3 successive recoils from a planar ^{224}Ra source

A high atomic weight, microcrystalline material (metallic Au) was chosen as a target. Profiles of ^{224}Ra (recoil from alpha decay of ^{228}Th) were studied for two different implantation geometries (parallel and isotropic beam).The profile of ^{212}Pb, implanted from a ^{224}Ra source through three successive recoils, was also studied.

Very thin uniform sources of ^{228}Th, prepared by the ion-optical method of Leon and Shafrir[3], were used as energy standards, but were too weak to be used as implantation sources. These were prepared by evaporation of a ^{228}Th(NO$_3$)$_4$ solution on polished Au foils. The targets were 99.99% pure gold foils, mechanically polished through 0.05 µ Al_2O_3 to a mirror finish. The implantation profile was determined by peeling successive layers of target and measuring the residual activity by alpha spectrometry. The corrosion-film peeling technique of Andersen and Sorensen[4,5] was tried without success. An electrochemical procedure based on Whitton and Davies[6] was used. Since the

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S U M M A R I E S

I. Experimental Nuclear Structure Physics

MEASUREMENT OF THE QUADRUPOLE MOMENT OF THE FIRST EXCITED 2^+ STATE OF ^{18}O

A.M. Kleinfeld*, K.P. Lieb*, D. Werdecker* and U. Smilansky

We measured the static electric quadrupole moment (Q_{2^+}) and $B(E2, 0_1^+ \rightarrow 2_1^+)$ of the first excited state of ^{18}O at 1.98 MeV. The values obtained were $(-0.19 \pm 0.02b)e$ (for the positive sign of the interference term involving the 2_2^+ state) and $(0.0048 \pm 0.0002b^2)e^2$ for Q_{2^+} and $B(E2, 0_1^+ \rightarrow 2_1^+)$, respectively. For these values the ratio $|Q_{2^+}|/[B(E2, 0_1^+ \rightarrow 2_1^+)]^{1/2}$ is about 3, whereas for no other nucleus does it exceed unity by more than 30%.

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NUCLEAR REACTIONS BETWEEN OXYGEN ISOTOPES

D. Kalinsky, D. Melnik, U. Smilansky, N. Trautner, B.A. Watson*, Y. Horowitz**, S. Mordechai**, G. Baur*** and D. Pelte***

Elastic and inelastic cross sections have been measured for the system $^{16}O+^{17}O$ at c.m. energies from 12.5 to 15.5 MeV, and for $^{16}O+^{18}O$ at c.m. energies from 12 to 20 MeV, at angles between 60° and 125°. Position-sensitive detectors were employed, using the kinematic coincidence technique. The data have been analyzed with particular attention to the contributions of multiple-exchange processes.

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DECOUPLING MEASUREMENTS WITH LOW-CHARGE OXYGEN IONS RECOILING INTO GAS

M.B. Goldberg, M. Hass and Z. Shkedi

A decoupling measurement with low-charge oxygen ions recoiling into gas was interpreted using a phenomenological approach. Values thus deduced for the g-factor of the 3^- level of ^{16}O and for the ion-atom collision cross-section were found to be in good agreement with results of other experiments.

THE MEAN-LIFE AND MAGNETIC MOMENT OF THE FIRST EXCITED STATE OF ^{20}O

Z. Berant, C. Broude, G. Engler*, M. Hass, R. Levy and B. Richter

The 2^+ first excited state of ^{20}O was excited by the reaction ^{3}H (^{18}O,p)^{20}O. Using the recoil distance method, a decay curve with hyperfine interaction effects has been observed. The energy of the state has been measured as 1675.0±1.0 keV, the mean-life is 14.2±0.8 ps, and the value of $|g|$ is 0.39±0.04. Comparison is made with shell model calculations.

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A PARTICLE-GAMMA ANGULAR-CORRELATION MEASUREMENT SYSTEM FOR HIGH COUNTING RATES

E. Abramson, M. Birk and Z. Vager

A system employing four semiconductor detectors and four NaI(Tl) scintillation counters for particle–gamma angular-correlation measurements at high counting rates is described. Scintillation counter pulses are shaped into rectangular pulses of 200 ns duration in order to reduce pile-up effects, and a compensation circuit for gain variation with rate is used. A fast decision-making logic unit employing MECL 10000 I.C.'s controls data transfer into an on-line PDP-9 computer. Dead-time losses are measured using a pulse-pair generator simulating particle–gamma coincident pulses.
In an experiment with 5–8 MeV protons and 0.5–2.5 MeV gamma rays at a counting rate of approximately 100 000 cps at each of the scintillation counters, the gamma-energy resolution (fwhm) was 6–9%, depending on gamma energy, and the time resolution (fwhm) was 1.8 ns. Plots of typical results are shown.

OFFSET ELECTROSTATIC QUADRUPOLE TRIPLET CHARGE SELECTOR

G. Goldring, Z. Segalov and E. Skurnik

An electrostatic displaced triplet for charge selection in the terminal of a 14 UD Pelletron accelerator is described. The device was checked experimentally with purely geometric displacements and with combined geometric and electric offsets. The rejection rate of charges adjacent to a selected charge is better than 1/20 for a divergent beam of 7 mrad half angle and charges as high as 14+. For a collimated beam of 1 mrad half angle complete separation is achieved.

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Ranges of ^{252}Cf Fission Fragments by Gamma Spectrometry

Y. Laichter and N.H. Shafrir

Experimental data concerning fission fragment ranges in matter are of interest in determining the mechanism of the interaction of energetic heavy ions with matter. Such data concerning the spontaneous fission of ^{252}Cf are rare[(1,2)]. This study is aimed to measure ranges of individual fission fragments in various solid and gasseous media. Fission fragment ranges are measured using high resolution, high efficiency Ge(Li) gamma sprectrometry. The fragments are identified by their gamma radiation energy and their half life.

Using a thin ^{252}Cf source having an activity of approximately 10^8 fissions/min, ranges of more than twenty fission fragments in various solid and gasseous matter are under investigation.

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Fission Yields of ^{252}Cf as Measured by Gamma Spectrometry

N. Golczar, Y. Laichter and N.H. Shafrir

Cumulative yields of 18 products in the spontaneous fission of ^{252}Cf have been determined. Measurements were carried out by gamma spectrometry using a high resolution Ge(Li) detector. Yields calculations were made by an absolute method based upon the exact calibration of the measuring system.

The following parameters were determined:

- The photopeak area was determined by the gamma spectrum of the fission products.
- Source activity was determined by gamma spectrometry and found to be 8.2×10^4 fissions/min.
- Collection efficiency of products was determined as the geometrical factor and was equal to 0.44 (from 4π).
- The absolute full energy peak efficiency was calibrated by point source; fission products which are planar sources were adapted to the calibration geometry.
- Functional primary yields were determined by solving the Gaussian equation for the nuclear charge distribution.

Till now calculations of fission yields of ^{252}Cf were made by comparison methods based on the normalization as compared with one yield which was chosen as the reference.

The absolute method enables the independent determination of fission yields and eliminates systematic error introduced by the normalization process.

^{252}Cf Fission Fragment Mass and Energy Spectra

Y. Laichter and N.H. Shafrir

In connection with a study of the energy deposition efficiency of energetic heavy ions in matter, a computer code is developed to synthesize ^{252}Cf fission fragment mass and energy spectra in any media.

The spectra obtained are compared with experimental results measured with a Si-solid state detector. This comparison serves to test the validity of various theoretical and semi-empirical approaches for the energy deposition of heavy ions in matter,[1-4] as well as the accuracy of the fundamental data.

Spectra measured with no moderation are used to check the accuracy of the kinetic energy and r.m.s. width data for the various ^{252}Cf fission fragments. Spectra obtained after moderation in solid and gaseous media are used to check the validity of the stopping formulae and the most probable charge of the fission fragment isobaric chains moving in matter.

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Determination of Energy Spectra of Fission Fragments by Means of Dielectric Nuclear Track Detectors

H. Hershfeld and N.H. Shafrir

The spectrometric response of a number of solid state nuclear track detector (SSNTD) materials, have been previously studied.(1,2) Glasses with high phosphate content which were shown to have the most desirable properties for fission fragment energy spectrometry, were chosen for spectra determinations of ^{252}Cf fission fragments.

All the experiments were carried out using perpendicular irradiation, i.e. the fission fragments entered the detector at right angles. The energy of the fission fragments was controlled by the nitrogen pressure in the irradiation setup. Series of experiments at various energies and different etching times were carried out. The measured parameter of the track was the diameter and for each experiment a distribution of track diameters was obtained, showing similarity to Si-detector spectra.

Using a dynamic model of the track evolution during the etching, an attempt is made to determine the range of ^{252}Cf fission fragments in phosphate glass.

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Identification and Quantitative Assay by X-Ray Fluorescence Technique

D. Segal, Y. Segal and A.Notea

The intensity of a specific x-ray energy emitted from a sample following its excitation by external beam is not a linear function of the element content in it. The deviations from linearity result partially from the response function of the specific spectrometer and majorly due to the transport of the photons in the sample (e.g. the matrix effect, particles effect). Due to the complexity of the processes involved, most of the techniques employed at present are based on empirical (or semi-empirical) relations which are suitable for a very limited range.

In the present project the transport of the impinging (primary interrogation beam) and fluorescence photons is analyzed by numerical approaches developed for reactor physics calculations. The suitability of the Monte Carlo method for the analysis of the problem is tested. The obtained results provide a deep insight to the emission procedure and a better understanding of the significance of the parameters involved in the design of an assay system.

Evaluation of Radiation Transmission Techniques for Detection of Inhomogeneity in Medium

A. Ginsburg, A. Gutman, A. Fishman, A. Notea, Y. Segal and B. Shapiro

Defectoscopy depends on the characteristics of the radiation employed, the beam-sample-detector geometry, the examined medium and the detection system (e.g. the camera). The analysis of the result depends on the interpretational model derived. The project is aimed at the study of the relative importance of the various parameters and the definition of a reduced set of major characteristics. The approach is based on defining probability functions to the parameters and analyzing the distribution obtained for the measured parameter. Emphasis is given to the prediction of the measurement's quality that is expected under given working conditions. This will enable the optimization of the design-parameters of the systems.

The work is carried out for photons and neutron beams. The aspects under consideration involves transmission gamma gauges, gamma radiography and neutron radiography.

Benchmark Analysis of Plutonium Fueled Reactor Lattices

W. Rothenstein and E. Taviv

Analysis of Uranium fueled reactor lattice benchmarks, which have recently led to a renewed study of the adequacy of the ^{238}U basic nuclear data,[1] are being extended to Pu fueled thermal reactor cores in the data testing program.

The benchmark specifications of the Cross Section Evaluation Working Group include a number of unreflected critical spheres containing plutonium nitrate solutions.[2] These are used mainly to test the data in the thermal energy region although in some of the spheres containing ^{240}Pu the 1.0 eV resonance must also be taken into account.

In addition reactor lattices containing Plutonium, or mixed oxides as fuel must clearly also be studied with the latest ENDF/B data and analysis procedures which can handle these data exactly in accordance with the specifications, and perform the lattice analysis with sufficient precision.

In the current project a modified version of the HAMMER code[3] is used. It represents the resonances by detailed tabulations of their profiles. A modified group structure will probably have to be used than the one customarily employed for Uranium lattices so that the low ^{240}Pu resonance will be taken into account in the thermal energy region.

The modified code has proved to be effective in accounting in the case of Uranium fuel for a number of effects which are normally ignored, specially in resonance reaction rate calculations. It will now be used for the study of the Plutonium fueled systems and in particular for the lattices described in [4,5,6]. If necessary comparisons with Monte Carlo calculations in the resonance region will be made.

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Improvement of Thermal Benchmark Analysis Procedures

W. Rothenstein and J. Barhen

The adequacy of a nuclear data base for reactor applications is determined on the basis of the analysis of "benchmark" experiments. Methods used in the analysis of such experiments must be accurate enough to insure that uncertainties due to approximations in the analytic model are negligible with respect to nuclear data uncertainties.

A commonly used analytical procedure is based on the lattice analysis code HAMMER.[1] Calculations done with this code must however be corrected with the use of more accurate representations in some areas and in partigular in the resonance region.

In the current project basic changes of the resonance reaction rate calculations are being made. These make full use of the latest ENDF/B data, and instead of evaluating resonance absorptions from the resonance parameters as was customary in lattice analysis codes in the past, the cross sections in the resonance region are represented by detailed tabulations[2] which can be accurately Doppler broadened.[3]

The algorithms for treating resonance shielding are based on the Nordheim Integral Treatment[4] and a modification of the RABBLE method[5] which can be applied to a numerical integration procedure on a fine energy mesh and utilises the detailed resonance cross section tabulations by linear interpolation. Resonance scattering is also taken into account in the modified calculations. In the unresolved resonance region the shielding of the p-wave as well as that of the s-wave resonances is allowed for.

The resonance treatment is performed for the heterogeneous unit cell of the reactor core, as though it were located in an infinite lattice of similar cells. Subsequent leakage calculations are performed on the basis of asymptotic reactor theory for a homogenised core. The interface between these two basic modules requires great care in the handling of the neutron balance so that heterogeneity and leakage are properly handled in the final evaluation of the resonance absorption rates.

The results based on the new code will be compared with previous analyses which relied on Monte Carlo estimates of the resonance absorption rates.

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Evaluation of Gamma Transmission Gauge for Assay of Water Content in Soil

A. Fishman and A. Notea

The attenuation of gamma radiation in soil is employed for the measurement of the water content in a layer of a given thickness. The characteristics of the gauge are analyzed as function of the soil type, thickness of the layer, field water capacity, gamma energy, detector geometry, efficiency, etc. An interpretational model is derived and is used for the resolving power estimation. The model is based on analytical approach and experimental data.

Parameter Analysis of an Electrostatic Particle Guide Using Radioactive Recoil Ions

A. Kenigsberg, J. Leon and N.H. Shafrir

A thorough parameter analysis of an electrostatic particle guide (EPG), constructed in the frame of a wider program for the study of nuclear stopping cross sections for slow heavy ions, has been performed. To obtain optimal performance for time-of-flight spectroscopy, the operational conditions of the system were studied in detail, using $\sim$100 keV alpha disintegration recoils. The parameters investigated were collector radial position, collector axial position, source radial position and wire voltage. Clear and sharp spectra with an energy resolving power of 1.5-3% FWHM, and transmission enhancement factors up to 300-400, were obtained.

Preparation and Testing of Multigroup Data for In-Core Power Reactor Physics Calculations

W. Rothenstein and L. Reznikov

Core physics calculations for power reactors require an accurate determination of the neutron spectrum over the entire energy range from the lowest energies of the thermal region to the high end of the fission spectrum. The neutron spectrum depends on the composition of the reactor core, which throughout its life changes with burnup. It must therefore be evaluated separately for each fuel element at numerous stages during its residence in the reactor. In addition, the spatial dependence of the neutron spectrum must be taken into consideration at energies where shielding is significant. This is the case in particular in the resonances of isotopes, like the fertile materials, which have pronounced resonance peaks and are present in the fuel rods at high atom densities. Shielding is also very important at thermal energies where cross sections of fissile isotopes become large, specially at the low end of the spectrum, and when the 1/v absorption of poison or control elements may also necessitate a space-energy spectrum calculation.

An efficient study of the spectral characteristics of the neutron population in a fuel element throughout its life in the reactor core must be based on rapid calculations which can be repeated as frequently as is deemed necessary. Yet accuracy must not be sacrificed for the sake of speed, nor should the calculations be purely mechanical. A detailed understanding of the procedures on which the calculations are based and the nuclear data which are employed is essential whenever special problems arise.

The LEOPARD program[1] may be regarded as one of the basic tools for accurate core physics calculations. This program has the advantage that it contains its own built-in neutron data libraries, but these libraries are relatively old and do not take most recent neutron microscopic cross section evaluations into account. In addition the number of isotopes in the library is not as complete as is desirable or necessary.

An updated ENDF/B-IV LEOPARD library is currently being prepared for all materials available at present in the code and other nuclides which may be considered to be desirable additions. Comparative studies of PWR fuel assemblies will be made throughout their life in the reactor for both the old and the new LEOPARD libraries. Special attention will be given to the resonance region where LEOPARD relies on correlations of resonance integrals with integral measurements[2]. It is of interest to determine to what extent these correlations are consistent with the latest ENDF/B-IV data evaluations, and this matter will be pursued to the extent to which this is feasible within the framework of the present proposal.

The comparative studies for fuel assembly calculations with the old and new libraries will lead to information of the relative changes which the recent data produce in the broad group fuel parameters and in particular k_∞ as a function of burnup.

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Shielding Effects in Fuel, Burnable Poison and Control Cells of Power Reactor Lattices

W. Rothenstein and I. Szabo

In core fuel management codes such as Penn State Fuel Management Package, frequently rely on data provided by the reactor manufacturer to calculate effective cross sections for burnable poison rods and control rods. On the other hand it is desirable to replace such empirical information by analytical procedures even in codes which perform the reactor lattice analysis rapidly and are designed to follow the fuel burnup.

Heterogeneity effects have considerable influence on the thermal neutron spectrum in power reactor fuel assemblies. The neutron flux in the rod of the reactor lattice unit cell is increasingly shielded when the absorption cross section Σ_a becomes large. This is the case for the fissionable isotopes which have high Σ_s's because of the close energy spacing of their resonances, and for control and burnable poison isotopes which are chosen for their large thermal absorption. In addition the 1/v dependence of Σ_a makes the shielding effect even more pronounced at the low energy end of the neutron spectrum.

Computer codes such as LEOPARD[1] which analyse the thermal, as well as the epithermal spectrum in fuel assemblies contain rapid techniques to determine the shielding factor and the neutron energy spectrum since the analysis must be made at frequent intervals during the life of the assembly in the reactor. LEOPARD relies for the heterogeneity calculation on the method of Amouyal, Benoist, and Horowitz[2] which it performs in each of the 172 thermal groups without allowing for the interaction with other groups. After homogenisation of the cell in the fine groups the Wigner Wilkins energy spectrum[3] is calculated in order to obtain the final one broad thermal group parameters for the fuel assembly by cross section averaging with flux weighting.

For burnable poison rods, the self-shielding effect must be calculated by a transport code which is not part of the LEOPARD code. Once the self-shielding is known, the rod can be replaced by an equivalent amount of soluble poison in the moderator to yield the same absorption rate. The control rod treatment is similarly approximate.

In the light of these considerations alternative methods are being studied to calculate the space dependent thermal spectrum in the fuel assemblies for the poison rods. These methods will also be applicable for fuel rod cells and will be compared with the current LEOPARD procedure. They will be based on the latest ENDF/B data including the scattering law for water and the THERMOS code.

On the basis of these investigations improved procedures for handling the shielding in cells containing burnable poison, and for handling control rod calculations, will be formulated and included in the LEOPARD code which is a basic program for in-core fuel management calculations.

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Modelling of Nuclear Fuel Waste Drum Assay by Passive Gamma Technique

A. Bar-Ilan, A. Knoll, A. Notea and Y. Segal

The spatial distribution of fuel contaminated solid wastes in drums is inhomogeneous and is unknown. The assay is based on the detection of the radiation leakage from the waste container. The count rate is not a single value function of the fuel content in the container. This study is based on estimating the possible deviation from an homogeneous distribution of the source, as well as of the waste materials. This deviation is treated as an additional source of error. The errors involved in the measurement and in the interpretation are analyzed and their relative contribution to the total error point at the pros and cons of the technique under given conditions.

Implantation of Radioactive Atoms by Alpha Recoil

S. Abrashkin and N.H. Shafrir

The penetration depths of ^{224}Ra recoils (from a ^{228}Th source)[1,2] in a polycrystalline target (Au) were measured, for parallel and isotropic implantation beams. Also measured were the penetration depths of ^{212}Pb (by ^{212}Bi - ^{212}Po measurements) from a ^{224}Ra source (i.e., atoms implanted by 3 successive alpha recoils). The peeling technique of Whitton and Davies was modified for the purpose of the work[3].

The results were examined from the viewpoint of a possible application in sensitive wear and corrosion measurements, as well as that of their agreement with L.S.S. theory. The results of the work do not agree with an often found explanation of the peculiar penetration profiles in polycristalline materials by means of a channeling contribution. An approximate calculation based on different assumptions (empirically derived from the results of this work for parallel beam implantation), gives reasonable agreement with the results for other implantation conditions.

Based on this approach, penetration parameters and profiles were calculated for alpha recoils in a range of metallic, polycrystalline targets.

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Effects of Physical and Operational Conditions on the Temperature Fields in Reactor Fuel Elements

E. Wacholder, E. Elias*, D. Hasan** and S. Kaizerman

Prediction of fuel element behaviour in a nuclear reactor following a postulated accident requires as its first step a knowledge of the temperature field during stead-state conditions which reflect the initial stored energy in the fuel. To accomplish this the dependence of the thermal properties (such as thermal conductivity of the fuel and cladding, and the thermal conductance of the gap between the fuel and cladding) on temperature and burnup must be taken into consideration.

The steady state problem, as stated above, has been solved by two methods:

(a) The numerical Successive Over Relaxation (S.O.R.) method (known also as the Extrapolated Liebman Method).(1)

(b) A computer oriented analytical method, based on Fourier Series and Kirchoff Transformation.(2)

These solutions have been employed for extensive parametric studies of the fuel elements' behaviour in the hot channels in a variety of modern reactors of practical interest.

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Light Water Reactor Accident Analysis

E. Wacholder, D. Hasan*, S. Kaizerman and S. Pryluk

The analysis of hypothetical accidents serves as an important tool in the determination of design-bases and in the evaluation and licensing activities related to nuclear reactors. Various computer codes are used by research institutions, official licensing agencies and various companies in order to predict the fluid - and thermodynamic behaviour in the light-water reactor systems during accidents

The work currently carried out is divided into three topics:

(1) The transient solution of the thermal-hydraulic conditions in the reactor-core hot channel.

(2) The prediction of the fluid - and thermodynamic behaviour in the primary coolant system of a P.W.R. during accidents.

(3) The prediction of containment response to accidents and the estimation of the radiological consequences.

In the first topic, a computer program (TRANS2[1,2]) for the solution of the thermal response of the fuel-rod and the coolant in a hot channel, to transients, has been developed. The program utilizes the A.D.I. (Alternating Direction Implicit) method for the solution of the general-heat-conduction-equation in the fuel rod, coupled with the solution of the one-dimensional-energy-equation in the coolant, using an implicit method. In addition to it, the THETA1 B and RELAP4 computer codes are operated. Comparative analyses of various accidents are performed using the three above mentioned computer programs.[2,3]

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In the second topic, the details of modelling a specified reactor system and its components using the RELAP4 computer code are studied. The results of the calculations also serve as boundary conditions to the core heat-up calculations (first topic, above) and to the containment calculations (third topic, henceforward).

In the third topic, the containment response to accidents is calculated to serve as a tool which will enable the derivation of bases for the containment design. The PREST computer program has been operated, and is currently improved to suit the above target. A P.W.R. dry-containment is modelled also, using the CONTEMPT computer code.

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Safeguarding Nuclear Fuel During Burnup

A. Notea and S.H. Levine*

A research program has been started to develop inspection techniques to safeguard used fuel from being diverted for unauthorized reprocessing. The research involves a coordinated analytical-experimental program wherein sophisticated reactor physics techniques are used to define limits of detection when used fuel has been replaced with equivalent new fuel.

The analytical program uses accurate in-core fuel management computer programs, and the experimental program involves fuel characterization studies. The program should result in methods to provide improved safeguards for reactor fuels.

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Minimization of Nuclear Fuel Costs

W. Rothenstein, M. Keren and S.H. Levine*

Optimization methods are to be studied for reloading nuclear reactors. The ultimate goal is to prepare procedures for automatically determining the reload configuration that minimizes fuel costs for the power plant. Computer programs are to be effected which enable the utility to make quick decisions on the fuel reload composition and configuration. Such computer programs are necessary when a power plant has been forced to make an unscheduled shutdown for an extended time period near the end-of-life of the core. The optimization programs will provide flexibility to vary the length of future reactor cycles so as to provide a global optimum operating condition in terms of minimum costs of the nuclear fuel.

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