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THE ANGULAR DISTRIBUTIONS OF NEUTRON IN REACTIONS (d, xn) AND (α,xn). The angular disrtibutions of neutrons from reactions (d.xn) at E_d = 22 KeV and (α,xn) at
α = 27 MeV from targets ranging in mass from 27 to 181 are analyzed in framework of phenomenology approach. It is shown that the shapes of continuum angular distributions can be described in terms of Legendre polynomials with the coefficients being a simple function of the emitted particles energy.

SPECTRA OF NEUTRONS FROM (d, xn)-REACTIONS AT THE DEUTERON ENERGY OF 22.3 MeV. Angular and energy distributions of neutrons from (d, xn)-reaction on nuclei 27Al, 53Cr , 90Zr, 94Zr, 115In, 122Sn, 181Ta at the deuteron energy of 22,3 MeV are measured.
The measurements were made by time-of-flight spectrometer at the 150 cm cyclotron of PBI for laboratory angles at 30° to 150° . The level density parameters are determined. The fractions
of compound and non-compound cross-sections are estimated.

CR0SS-SECTIONS OF NEUTRONS WITH ENERGY 0,001-2,5 MeV ON EVEN-EVEN INFORMED NUCLEI IN STRONG CHANNEL COUPLING METHOD. Results of the generalized optical model calculations neutron cross-sections on even-even deformed nuclei of actinide region 230Th, 232Th, 234U, 238U, 242Pu, 246Cm, 252Cf are presented. Comparison
with other similar calculations is carried out and possible reasons of different cross-section dependence with energy are discussed. The reliability of the obtained results is confirmed by a good agreement found for 238U test calculations.

EMBEDDING OF INTERNATIONAL SYSTEM OF UNITS IN RADIATION SHIELDING. Peculiarities of transition on SI units in radiation shielding problems are investigated on base of Methodical manual RD 50-484-84 about embedding and usage SI units in the field of radiation. Some recommendations about employment of units for main physical values are given.

SOME IDEAS IN PREPARING OF CONSTANTS FOR GROUP CALCULATIONS OF NEUTRON FIELDS IN SHIELDING. The present status of constant system developing for reactor and shielding calculations is given. The data bank is consists from library of evaluated neutron data (FOND), system of 26-group constants (BRAB-85) and multi-group neutron library. The service functions for these libraries, a calculation of group Constants (codes GRUCON, NJOY), a preparing of constants (codes ARAMACO-S1, MULTIC) for calculations by codes ROZ-6, ANISN, DOT-III and other ones are organized.

THE PROGRAM COMPLEX OF PREPARATION OF 175-GROUP NEUTRON CONSTANT
SYSTEM WITH THE SUBGROUP RESONANCE RANGE. The program complex for preparation neutron constants for statistical model studies of neutron propagation was described. The neutron cross-sections from ENDL-78 and ENDF/B-IV were included in the system. The LINEAR and GROUPIE programs were realized on EC-computer.

THE GROUP CONSTANTS LIBRARY FOR PHOTON PRODUCTION CALCULATION IN NUCLEAR INSTALLATIONS. The group constants library for photon production in neutron - nucleus interactions are described. The library is based on the ENDL evaluated data files. The 49- group structure for neutrons and the 15-group structure for photons was taken. The library includes data for 41 nuclei to be important for nuclear reactors and shielding.

THE STATUS OF NUCLEAR DATA FOR NEUTRON PROPAGATION PROBLEM IN THE ATMOSPHERE AND THE ENVIRONMENTS OF MAIN ROCK ELEMENTS. The analytical approximation of macroscopic cross-section and neutron free paths in the atmosphere and soil were obtained in energy range 1 eV - 100 keV. Some of them are recommended for use
in neutron field calculation and evaluated data files. The status of standard data and recommended data are discussed.

OKC-3.5 - THE PACKAGE OF PROGRAMS FOR CONSTANT SYSTEM. The United system OKC-3,5 Includes the most well known group cross-sections systems such as ARAMAK0-2F, ARAHAKO-G, TERMAC and others. The group cross-sections, received with help of OKC, may be used for calculations of reactors and radiation shielding. Also OKC allows to carry out the reorganization of received files; the transfer from one format to another, the merging of two files together.

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THE SYSTEM OF CONSTANT PREPARATION FOR NONGROUP SHIELDING CALCULATIONS BY THE MONTE-CARLO METHOD. The system of constant preparation for non-group shielding calculations by the Monte-Carlo method based on the detail nuclear date information is described. The structure of data bank and service programs for its processing are briefly discussed.

TESTING OF EVALUATED NEUTRON DATA OF CHROMIUM AND NICKEL ON THE RESULTS OF MACRO EXPERIMENTS. On the base of macro experiments the test of the chromium and nickel evaluated neutron data developed in CJD (FEI) has been performed. The
calculations are made by means of the ANISN code using 100 group energy presentation of GAM-II obtained in the NJOY program.

THE DETERMINATION OF THE EFFECTIVE (n , n′) AND (n , 2n) REACTION CROSS-SECTIONS ON NIOBIUM. The problems connected with the study of the VVR-M reactor spectrum neutron induced activation of Nb-based structural materials that can be used in the creation of shield for nuclear-engineering units are considered. Data on resonance and threshold detectors and programs which were used for restoration of neutron spectra influencing the specimens studied is given. The estimates of the effective 93Nb(n , n'), 93mNb(E_{thr} – 0.1 MeV) and 93Nb(n , 2n)92mNb(E_{thr} - 10 MeV) reaction cross-sections which values were (126 ± 25) and (350 ± 80) mb, respectively, were made.

THE INFLUENCE OF SLOWING DOWN, RESONANCE SELF-SHIELDING AND ANISOTROPIC ELASTIC SCATTERING FOR RESULTS OF MULTI-GROUP CALCULATIONS IN FAST REACTOR SHIELDING. The influence of slowing down model, resonance structure of cross-sections and anisotropic elastic scattering in calculations of the fast reactor shielding are examined. The calculations were made by means of Monte-Carlo method. The calculations of fast neutron flux on a large distance from core depended on accuracy of slowing down model. It is shown that the calculation of fast reactor shielding in P_3 -approximation of anisotropic function is good.

PHYSICS OF FORMATION AND STATISTICAL ADJUSTMENT OF THE CALCULATIONS OF RADIATION FIELDS IN REACTOR AND SHIELDING. Comparisons
between different results of neutron field calculations for fast reactor radial shield configuration are made. Calculations are performed utilizing Russian and foreign cross-section libraries.
As an illustration of usage of sensitivity analysis and special channel theory for determination of certain detector response formation physics one shielding configuration is investigated. Statistical adjustment of results of transport calculation is made.

PHYSICAL PECULIARITIES OF NEUTRON TRANSPORT IN FAST REACTOR IN-VESSEL SHIELD. Possibility of usage of sensitivity analysis for reactor shielding project is analyzed. For bench-mark calculational configuration of spherical in-vessel fast reactor shield peculiarities of radiation transport are determined. Importance of different partial cross-sections and contributions of different isotopes are estimated.

UNCERTAINTY ANALYSIS APPLIED TO THE FAST REACTOR SHIELDING CALCULATIONS. Sources of uncertainties applied to the calculations of different detector responses for neutron fields in shielding are analyzed. Sensitivity analysis and uncertainty analysis of two bench-mark calculational configurations of in-vessel fast reactor shield were performed. Uncertainties of radiation heating and displacement rates calculations, secondary sodium activation calculations due to the total and partial cross-section errors are investigated.

EXPERIMENTAL DETERMINATION DELAYED NEUTRON GROUP YIELDS FROM THERMAL FISSION OF 233U AND 235U. A method for evaluation of fission product yield calculations by delayed neutron group yields measurements is presented. An apparatus for this methods realization, which is based on automatic pneumotransport system, is described.

A METHOD OF THE TAKING INTO ACCOUNT OF THE UNRESOLVED RESONANCE STRUCTURE IN THE SLOWING-DOWN EQUATION. The problem is investigated to obtain the detail slowing-down equation the probability resonance structure describe being used in modern neutron data libraries in the unresolved resonance region. There is proposed to consider the detail solution as a probability average, which can be obtained by averaging in described way
the parameters of the original equation.