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GOVERNMENT OF INDIA
ATOMIC ENERGY COMMISSION

THEORETICAL REACTOR PHYSICS STUDIES-ANNUAL PROGRESS REPORT (1965)

by

B. P. Rastogi
Reactor Engineering Division

ATOMIC ENERGY ESTABLISHMENT TROMBAY
BOMBAY, INDIA
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INTRODUCTION

The primary aim of the Theoretical Reactor Physics Section of Reactor Engineering Division is to develop necessary methods of calculations for various physics problems encountered in reactor design, operation, optimization, fuel-cycle studies and experimentation. The Section works on lattice physics, neutron-thermalisation, resonance absorption, transport theory, reactor control and dynamics and nuclear data. Various types of reactors are under consideration. However, particular emphasis is given to the liquid moderated thermal reactors and fast systems.

The work on these problems was started in the middle of 1961. However, due to the absence of proper computation facilities during 1961-64, much progress could not be made. With the installation of CDC-3600 computer complex at T.I.F.R. in the latter half of 1964, work was started on the development and adoption of computer codes, and the work done during 1965 is described in this report. This report gives the abstracts of the papers, reports and internal notes completed during 1965. The abstracts have been divided into eight categories.

The group of seven reports under the category General Studies discusses some of the calculations which were made for particular systems. The report on Prototype Power Reactor Project is a compilation of eleven notes which were written on the Project. The fuel cycle studies for CANDU and PHWR type systems were started during the year, though the methods of calculations were not very refined. Classical four factor formulation with Westcott cross-sections gave rather unsatisfactory results for enriched thorium lattices. It was also noted that four group formulation of CAROL did improve the results, but still had an uncertainty of about two percent in reactivity for the enriched lattices. Removal of this discrepency is one of the most important and challenging

problems confronting us. The problem of thorium utilization in thermal reactors and the possibility of breeding in thermal reactors are linked with this problem. The work on other problems such as neutron thermalisation and resonance absorption was done with the aim to resolve this discrepancy.

Five criticality and burn-up computer codes have been described, in the section on Reactivity Codes. These codes are based on one dimensional two group leakage model. Work on neutron thermalisation was done in a more systematic manner. A review of the theoretical work done on the subject was first made and subsequently, five codes were made for spectrum calculations in the lattice. There are five codes on resonance absorption and it is expected that EPITHET coupled with THERMNIL may be able to resolve the problem of reactivity discussed above. There are two abstracts on transport theory problems. The two codes on reactor control problems which are adequate for problems encountered in CIRUS, ZERLINA, and CANDU type of reactors, may not be adequate for PHWR type reactors. Work on nuclear data was started during the year and efforts are being made to make a computer code which could be used for compilation, storage and evaluation of nuclear data. The last abstract refers to a code made for loop operation in CIRUS. Work on adoption of a number of codes such as ELMOE, SNG, CANDID-2D, RESPECT, DICE, ABASUC-2, NEARREX and DOPINT is continuing. They have not been discussed in this report.

A review of this report reveals that a number of computer codes are available for studies pertaining to heavy water moderated reactors. However, many more codes are needed for analysis of start-up experiments and reactor operation. Reactivity calculations for light water moderated reactors could also be made with the help of CAROL code, but no codes are available for three dimensional burn-up and control studies. Codes for fast reactor studies are becoming available. It is also noted that most of the codes discussed are mainly based on the work done elsewhere, this was due to the fact that the computer became available only a year and half back and there was a need to have basic codes as early as possible. In the following years, it is expected that more sophisticated codes would be available, the work for which has already been started.

GENERAL STUDIES

1. Physics Studies of Prototype Power Reactor Project
(B.P. Rastogi, K.R. Srinivasan, A.N. Nakra, H.K. Bhatia, H.C. Huria K. Balakrishnan and H.D. Purandare, AEET-239).

This report discusses various physics aspects of Prototype Power Reactor Project. PPR is a heavy water moderated plutonium enriched reactor. There are 55 vertical channels in the core of which central 37 are cooled by D_2O. Outer 18 channels are available for development of various coolants and fuel elements. The fuel in the inner 37 channels is of CANDU type.

All calculations for this report were made on desk calculating machines. A number of problems were undertaken to obtain data for the feasibility and detailed design group working on the reactors.

2. *Some Physics Calculations of Thorium Fuelled Heavy Water Lattices (R. Gopalan, A.N. Nakra, K.R. Srinivasan, T.A. Subramanian and B.P.Rastogi, AEET-230).

This paper discusses the Deuterium Moderated Lattices Calculations program DUMLAC, Open Lattice Burnup Studies program OLBUS and Open Lattice Burnup Studies with Thorium and Plutonium program OLBUSTAP, for heavy water moderated lattices. The program DUMLAC is for clusters as well as for open type lattices whereas the programs OLBUS and OLBUSTAP are only for open type lattices. The results of the calculations obtained from these programmes have been compared with the experimental values obtained in the U.S.A. for U^{235} enriched thorium fuelled lattices.

Some preliminary studies of the fuel cycles using thorium plutonium fuel have been considered for two types of heavy water moderated reactors.

* Paper presented at the I.A.E.A. Panel Meeting on 'The Utilization of Thorium In Power Reactors', held in Vienna from June 14 to 18, 1965.

3. * Physics Aspects of Large Thorium Fuelled Fast Reactors
(R. Shankar Singh, AEET-229).

The feasibility of using U^{233}-Th as a fuel in a large fast breeder reactor has been studied from the point of view of the Doppler and sodium void coefficients. The composition and temperature dependent cross-sections for thorium have been evaluated using the latest resonance parameters available. The parametric studies over a range of compositions and sizes of core for metal and oxide fuels indicate that the reactivity coefficients due to sodium loss and Doppler effects are negative even in a core volume as large as 15,000 litres. Based on these encouraging results, a performance characteristics study of a 15,000-litre, thorium U^{233}-fuelled, sodium cooled fast reactor of the type considered for desalination of sea water was made. The parameters like critical mass, breeding ratio and reactivity coefficients have been evaluated for the 'clean' core as well as for the core with equilibrium composition using metallic and carbide fuels.

The effect of interference interaction between the resonances of Pu^{239} and Th^{232} on the Doppler effect calculations and the sodium void coefficients in Th-Pu^{239} fuelled fast reactors has been investigated. The temperature dependent effective cross-sections for Pu^{239} and Th^{232} have been evaluated taking into account both types of interference interactions, namely, the overlap of neighbouring resonances in one isotope and the interaction between the resonances of the two isotopes.

* Paper presented at the I.A.E.A. Panel Meeting on 'The Utilization of Thorium In Power Reactors', held in Vienna from June 14 to 18, 1965.

4. * Status Report On 'Thorium Utilization Programme - Physics Studies In India',
(B.P. Rastogi, R.S. Singh and K.R. Srinivasan).

This report discusses the aim and objects of thorium utilization programme in India. It also discusses the lines on which the physics studies are being carried out.

A number of computer programmes are being built for the study of thorium heavy water moderated lattices. A number of lattices experiments have been also planned. Heavy water moderated reactors of CANDU and PHWR types are being considered for thorium utilization.

Studies have been made for large fast reactors utilizing thorium. Work is continuing on evaluation of cross-section data and improved methods of calculation for reactivity coefficients.

* Paper presented at the I.A.E.A. Panel Meeting on 'The Utilization of Thorium In Power Reactors', held in Vienna from June 14 to 18, 1965.

5. ** Fuel Cycles in CANDU and PHWR Type Reactors,
(T.A. Subramanian, K.R. Srinivasan and B.P. Rastogi)

The heavy water moderated reactors, both of pressure tube design and pressure vessel design, are being studied in India. CANDU is of pressure tube design and has the advantage of cold moderator and bi-directional fuelling. The pressure vessel reactor of PHWR type developed in Sweden uses hot moderator, but no pressure tubes.

Plutonium recycle and gradual introduction of thorium is being studied for both these types of reactors. Necessary computer codes were developed and used extensively for the purpose. The studies suggested that use of plutonium enriched thorium as fuel, gives the same performance for CANDU and PHWR in the context of discharge irradiation, for the same type of fuelling, although CANDU seems to be slightly better. For recycling of plutonium with natural uranium in PHWR, close packed lattices are advantageous, and an early transition from natural uranium cores to plutonium enriched cores gives better fuel utilization.

** Paper presented at the Nuclear Physics and Solid State Physics Symposium, held in Bombay from February 21 to 25, 1966.

6. Use of Plutonium for Power Production - A Survey of the I.A.E.A. Panel Discussions
(R. Shanker Singh, internal report)

A meeting between the experts from different countries was convened from 7-11, December, 1964 in Vienna by the International Atomic Energy Agency to discuss the current status of plutonium utilization as a power reactor fuel and the research and development work taken up in this field by different countries. The various topics that were discussed during this meeting, the programmes of plutonium utilization presented by

the participating countries and the summaries of the different aspects are briefly outlined in this note.

7.** Analysis of Uniform-Lattice Critical Experiments With Thoria-Urania Fuel In Heavy Water and Light Water as Moderators, (H.K. Bhatia).

Calculations have been carried out to compare the uniform lattice experiments performed at Argonne National Laboratory and at Babcock and Wilcox Company, U.S.A., with ThO_2-$U^{235}O_2$ as fuel using heavy water or light water as moderators. The preliminary analysis of BNL-experiments with ThO_2-$U^{235}O_2$ rods in light water is also reported. METHUSELAH-I Code developed by A.E.E.W., and CAROL-Code developed by AB ATOMENERGI, SWEDEN, have been used for the computation of k_{eff} from the experimental bucklings. Both the codes use four neutron energy group formulation with some differences in the calculation of fine structure, Dancoff factor, etc. Theoretical and experimental bucklings are compared and the divergence of k_{eff} from unity is an indication of difference between the computed reactivity and the measurements.

** Paper presented at the Nuclear Physics and Solid State Physics Symposium, held in Bombay from February 21 to 25, 1966.

COMPUTER CODES

8** Computer Codes for Thermal Reactor Physics Problems (B.P. Rastogi, K.R. Srinivasan & S.V. Lawande)

This paper describes briefly some of the Computer Codes developed and adopted in the Theoretical Reactor Physics Section to study the various reactor physics problems encountered in the design of the heavy water moderated thermal reactors.

CAROL, OLBUSTAP and DUMLAC codes calculate the point reactivity values as a function of burn-up. THERMNIL and LATSP calculate the thermal neutron spectrum in the lattices. BAKER, KERN, FGKER and POLYKERN calculate the scattering kernels for moderators. Resonance absorption could be treated with the codes EPITHET, NINEGRI, JSIBETA and DANCOFF-I. WOCROD and HORVECO calculate

the reactivity worth of control rods. Diffusion theory codes CRITIC, TUMULT and CODBURN calculate the flux and power distribution. The parameters of a loop in CIRUS could be calculated by the code CIRLOOP.

** Paper presented at the Nuclear Physics and Solid State Physics Symposium, held in Bombay from February 21 to 25, 1966.

9. DUMLAC-A Program to Calculate Lattice Properties of Heavy Water Moderated Lattices
(A.N. Nakra and K.R. Srinivasan, to be published).

DUMLAC (Deuterium Moderated Lattice Calculations) is a CDC-3600 program to calculate the lattice properties of heavy water moderated lattices. This program was written with a view to study various heavy water moderated lattices as a function of fuel design, lattice pitch, temperature and burn-up. The methods of calculations are based on the classical four factor and two group leakage model. Most of the recipes used in the code are based on Canadian and Swedish prescriptions. The program is quite general in nature. Various types of fuel clusters or open type of lattices could be studied. Provision has been made to store the nuclear constants of different materials used for fuel, sheathing and coolant.

10. OLBUSTAP- A Burn-up Program for Heavy Water Moderated Uniform Lattices
(R.Gopalan and T.A. Subramanian, to be published)

This report describes the program OLBUSTAP (Open Lattices Burnup Studies of Thorium and Plutonium) developed for heavy water moderated, uniform lattices. It is in Fortran Language for the computer CDC-3600. A system of burn-up equations are solved numerically and the resulting isotopic compositions are used for calculating various lattice parameters by a conventional recipe.

11. TUMULT - A Program to Solve the One Dimensional Two Group Equations for Cylindrical Multizone Reactors
(K.R. Srinivasan, internal note).

TUMULT solves the one dimensional, two group diffusion equations in cylindrically symmetric multi-region reactor by taking the analytical solutions of the equations. The method is based on a procedure evolved by McKay[1] which reduces the multi-region problem

to that of a single annular region. This is accomplished by using the given boundary conditions at the outermost (or innermost) boundary of the reactor to eliminate the unknowns in the outer-most (or innermost) annular region and to calculate the new boundary conditions to be applied to the remaining system of annuli. Repetition of this procedure reduces the problem to that of a single region, which is solved to obtain the critical condition by the usual methods. The program also gives the flux distributions in the reactor.

(1) C.D.McKay - AECL - 1307.

12. CRITIC - A Code For Solving Two Group Diffusion Equations for a Cylindrical Multizone Reactors
(K. Balasubramanian, internal note)

This note describes a code for solving the problem of criticality in a multi-zone cylindrical reactor by using two group diffusion equations in one dimension. The code is based on the method developed by Hassitt[1] in which the two group diffusion equations are solved in the radial direction by finite difference method.

(1) Hassitt - AERE.T/R-1904.

13. CODBURN - A Cylindrical One Dimensional Burn-up Program
(K.R. Srinivasan, internal note)

Using point reactivity values, CODBURN calculates the reactivity and radial flux and power distribution of a multi-region reactor at various time steps. One dimensional two group diffusion theory is used. The burn-up for different types of fuelling schemes like fixed fuelling, in to out and out to in shuffling could be obtained.

NEUTRON THERMALISATION

14. Neutron Thermalisation
(S.V Lawande, internal report)

This report deals with steady state neutron thermalisation and is intended mainly to give a broad view of the subject. The matter is divided into three parts. Part-I is a brief introduction to thermalisation problems. Part-II discusses the theory of slow neutron

scattering and the formulisms leading to the evaluation of the scattering law for a given moderator. An account of neutron spectra in infinite homogeneous media and reactor lattices is given in Part-III. Although the report does not aim at giving a detailed account of the experimental work, a few comments regarding the relative positions of theory and experiments have been added in appropriate places.

15.**Neutron Thermalisation In Reactor Lattices-Program Thermnil
(V.K. Jain and S.V. Lawande)

A method is developed for computing thermal neutron spectra in reactor lattices as functions of energy and a single spatial co-ordinate. The Wigner-Seitz cell approximation is used. The program computes a number of parameters related to the spectrum such as neutron temperature, f, η, average cross-sections etc.

The method used is the multi-group multi-zone formulation of the Boltzmann equation. The scattering kernel is assumed to be isotropic. The program has a built-in Brown and St.John's kernel but option is provided to use any other kernel such as Nelkin H_2O or Honeck D_2O as input data. The energy dependence is solved exactly (within multi-group approximation), while the spatial dependence is computed by means of zone to zone first flight collision probabilities.

The epithermal source is computed using flat 1/E dependence. The energy mesh used is essentially the THERMOS mesh but an option is there to use any desired mesh subject to the restriction of 30 groups. The maximum number of homogenised zones is limited to 5. The number of spatial regions is limited to 15. Every zone can have a maximum of 10 isotopes. The program is very well suited for the study of neutron spectra in lattices with non-uniform temperatures.

** Paper presented at the Nuclear Physics and Solid State Physics Symposium held in Bombay from February 21 to 25, 1966.

16.** Thermal Neutron Spectra in Reactor Lattices
(S.V.Lawande)

A simplified method for computing thermal neutron spectra in reactor lattices is considered. Simplifications are in the nature

of treatment for space variable and in the assumption of Wigner-Seitz cell approximation. The energy dependent transport equation is solved exactly within numerical approximation for spatially averaged flux in fuel, cladding and moderator. The method is applicable for lattices at uniform temperature using natural or enriched uranium and any of the usual moderators like H_2O, D_2O, Be or graphite. A Fortran program called LATSP is written to carry out the computation on CDC-3600 computer. The program has options for the type of fuel (either U-metal or U-oxide), lattice pitch (square or hexagonal) and the scattering kernel. The kernel data could be supplied as input or it could be generated by either of the two sub-routines BAKER and FGKER as desired. Sub-routine BAKER computes the bound atom kernel according to the models of Nelkin (H_2O) and Honeck (D_2O). Sub-routine FGKER computes the free gas kernel according to the models of Wigner and Wilkins, and Brown and St.John.

Several results for enriched U-metal light water lattices and uranium oxide heavy water lattices have been obtained which show reasonably good agreement with the published results of Honeck.

** Paper presented at the Nuclear Physics and Solid State Physics Symposium, held in Bombay from February 21 to 25, 1966.

17. KERN and PAR Programs For Computing Scattering Kernels in H_2O, D_2O and Polyethylene
(S.V. Lawande, to be published)

This report describes a program for computation of kernels for thermal neutrons scattered inelastically from chemically bound target nuclei in moderators. The main program that achieves this is called KERN. The two auxiliary programs PAR-I and PAR-II compute the specific input parameters for KERN according to either Nelkin Honeck model for H_2O, D_2O or Goldman model for polyethylene. The KERN and PAR programs are essentially based on the KERNEL and PAM programs of KAPL[1]

(1) Federighi and Goldman, KAPL-2225, (1962).

18. INSPECT - A Program for The Computation of Thermal Neutron Spectrum

in an Infinite Homogeneous Moderator Using Gas Model
(S.V. Lawande and G.A. Desai; Internal note)

This note describes a program for the computation of thermal neutron spectrum in an infinite homogeneous moderator. The gas model of Wigner and Wilkins and 1/v absorption law are assumed and the program is written in FORTRAN for the CDC-3600 computer.

19. NSIHM - A Program for The Calculation of Neutron Spectrum in Infinite Homogeneous Moderators
(V.K. Jain and S.V. Lawande; Internal note)

This note describes the program NSIHM which calculates neutron flux distribution in infinite homogeneous moderator under the assumption of heavy gas model. The program also prints out the effective neutron temperature for any upper thermal cut-off energy. The calculations could be made only with a 1/v-absorber in the moderator.

RESONANCE ABSORPTION

20. EPITHET - A Multi-group Multi-region Code to Calculate Epithermal Flux and Absorptions in a Cylindrical Cell
(H.C. Huria, to be published)

The Code EPITHET uses a multigroup - multi-region technique to calculate epithermal flux and absorptions in a cylindrical cell. The method of calculation was formulated by Driggers[1]. The method consists in dividing the lattice cell into N-regions and the energy range of calculation into K-groups. The total collision density in a region for each group is calculated in terms of (a) number of neutrons slowing down to that group from higher energies and (b) currents entering the region through its outer and inner surfaces. The first term is obtained by calculating the contribution of the higher groups to the group under consideration through slowing down equations. The currents are obtained through neutron conservation relations for the region, using the escape probabilities, which, in turn are calculated by polynomials expressions given by Kennedy[2]. Once the collision density is obtained, calculation of the fluxes as a function of energy and space and the absorptions (resonance and non-resonance) follows and hence resonance integral and epithermal η and f. Some of the features of the program are:

(i) 100 energy groups, 20 spatial regions and 15 materials per region. Of the 15 materials there could be 5 with resonance absorption and/or 5 with fission-absorption.

(ii) Spacing of the regions and that of the groups is quite arbitrary.

(iii) Flux in the topmost group is either specified or could be assumed as flat.

(iv) Cross-section input is numerical so that the effects like Doppler broadening and interference between the resonances could be treated very easily.

(1) Driggers, AECL-1996: (2) Kennedy, see (1).

21. NINEGRI - A Program to Calculate the Resonance Escape Probability by Nine Resonance Energy Groups for D_2O Moderated Lattices (A.N. Nakra, to be published).

The program NINEGRI calculates the resonance escape probability of D_2O moderated lattice using nine resonance energy groups, based on the recipe given by Brooks and Soodak[1]. The calculated results were compared with a few experimental values. It was found that the calculated values overestimate the resonance absorptions in all the cases. However, the agreement with the experimental values could be obtained by using an empirical parameter.

(1) W.L. Brooks and H.Soodak - NDA 2131 - 19.

22. DANCOFF-1 - Calculation of Dancoff Correction for Two Isolated Infinite Circular Cylinders (K.Balasubramanian and G.V. Acharya; Internal note)

The note presents the calculations of Dancoff[1] correction C for the isolated infinite circular cylinders. Dancoff's formula used in this note is in the form obtained by Carlvik and Pershagen[2]. The correction factors have been calculated for the five different values of a/λ of Thie[3] where 'a' is the rod radius and λ the mean free path in the moderator. The values calculated by the program

compare well with those of Thie. As the third order Bickley function used in the calculation of C is computed by Simpson's rule, the computation time for calculation of C is large.

(1) Dancoff and Ginsburg, CP-2157
(2) Carlvik and Pershagen, AE-16
(3) Thie, Nucl.Sci.and Engg. 5, 75-77 (1959).

23. JSIBETA - A Program to Calculate Doppler Broadening Function J (ξ, β) (K. Balasubramanian and G.V. Acharya; Internal note)

The note presents the calculation of the function J (ξ, β) which calculates the temperature corrections to the resonance integral. The polynomial fit for J (ξ, β) used in the program is due to Doherty[1]. The fit has been successfully used to reproduce (a) the values of J (ξ, β) calculated by Dresner[2] and (b) the values of the resonance integrals of uranium metal at 300°K of Sumner[3], calculated under NR, NRIA and λ-approximations.

(1) Doherty - AAEC/TM - 96.
(2) Dresner - Resonance absorption in nuclear reactors, Pergamon Press
(3) Sumner - AEEW - M 304.

24. RESIN -1 The Resonance Integrals for Heterogeneous Mixtures (G.V. Acharya; Internal note).

The note presents the formulation for the calculation of resolved resonance integrals for the heterogeneous case of an absorber with any diluent. The formulation adopts λ and μ methods. The program RESIN-1 carries out the calculation. Sample calculations have been carried out for the case of uranium carbide and the calculated values of the resonance integrals are compared with Monte Carlo values of Levin[1].

(1) Levin, Nucl.Sci. and Engg.16, 271-279 (1963).

TRANSPORT THEORY

25 ** Multiple Collision Probabilities in Thin Slabs (D.C. Sahni)

The Boltzmann transport equation has been solved by using the method of Case to obtain the multiple collision probabilities in slab geometry when the neutrons are incident on one of its plane faces with various angular distributions. Calculations have been made for slabs of thickness 't' ranging from 0.1 to 2 m.f.p. and the average number of secondaries per collision 'c' varying from 0.6 to 0.95. The source angular distribution considered were uniform, $\cos\theta$, $\cos^2\theta$, and $\cos^3\theta$.

** Paper presented at the Nuclear Physics and Solid State Physics Symposium held in Bombay from February 21 to 26, 1966.

26. An Application of The Theory of Singular Integral Equations to Neutron Flux Problems in Cylindrical geometry (D.C. Sahni, Energia Nucleare, Nov. 1965).

The solution of the steady state integral Boltzmann equation is considered in a two region cylindrically symmetric system. The inner region is a scattering and absorbing medium, which may also contain isotropic sources. The outer region contains uniform isotropic sources, and is purely scattering with a cross-section equal to the total cross-section of the inner medium. The method consists in reducing the problem to a system of coupled singular equations. A numerical method for their solution is discussed. Flux distributions and linear extrapolation distances are determined for cylindrical cells of infinite radii.

REACTOR CONTROL

27 ** On The Reactivity Worth of Transverse Control Rods (H.D. Purandare)

The problem of finding the reactivity worth of control rods perpendicular to the axis of a cylindrical core cannot be solved by the Scalettar-Nordheim method in the usual form. However, an approximate solution for a cylindrical core can be obtained by dividing the core into a number of rectangular parallelepiped with the control rods perpendicular to a pair of faces. For each parallelepiped a criticality equation can be set up using theta functions. The solution for the whole cylinder can be finally obtained by variational technique.

A computer program HORVECO using this method for two energy groups was prepared for CDC-3600 and some calculations were done for the control system of the Rajasthan Atomic Power Project.

** Paper presented at the Nuclear Physics and Solid State Physics Symposium held in Bombay from February 21 to 25, 1966.

28. WOCROD - A Program To Calculate the Reactivity Worth of Rings of Control Rods.
(H.D. Purandare, G.A. Desai, internal note)

The program WOCROD, prepared for CDC-3600, finds out the reactivity worth of control rods in a bare core. The rods are parallel to the core-axis and may be arranged in the form of a number of concentric rings such that the number of rods in any ring is an integral multiple of the number of rods in one particular ring. WOCROD-1 uses one group theory and WOCROD-2 uses two group theory.

The flux along the axis is assumed to have a cosine distributions To find the critical radial buckling, a criticality equation is set up by applying the boundary conditions to the flux on the outer reactor boundary and on a rod of each ring wherein all rods are equivalent. For M rings of rods, the two group criticality equation is a determinent of order 2M. The input to be provided is the reactor parameters, rod radius, extrapolation distance and the inter-rod distances. The output is the reactivity worth.

NUCLEAR DATA

29 Review of Nuclear Data Related to Theorium Fuel Cycle
(R. Shankar Singh, A.E.E.T. - 234)

A knowledge of accurate and reliable nuclear data for the materials in any fuel cycle is essential in assessing its technical feasibility and economic potential. In this review, the nuclear parameters, in which a reactor physicist is usually interested, are discussed for the heavy elements associated with the thorium cycle, such as Th^{232}, U^{233}, Pa^{233} and U^{234} and the fission products. The existing data for these materials and the gaps that have to be filled up are described. Temperature dependent effective group capture cross-

sections for thorium, which were evaluated with the available resonance parameters, are also tabulated.

30 ** Temperature Dependent Effective Multi-group Capture Cross-sections Of Thorium
(R. Shankar Singh)

Effective multi-group capture cross-sections for thorium required in reactor physics calculations are evaluated from the resonance integral computations. The latest resonance parameters (resolved upto 4000 ev) for thorium have been used for this purpose and the p-wave contribution to the resonance integrals in the unresolved resonance region has also been taken into account. A code 'DOPINT' prepared for CDC-3600 has been used for the calculations and the multi-group capture cross-sections evaluated at four temperatures are tabulated.

** Paper presented at the Nuclear Physics and Solid State Physics Symposium held in Bombay from February 21 to 25, 1966.

SPECIAL STUDIES

31. CIR LOOP - A Program to Calculate The Reactivity Load and Relative Power Output of a Loop in CIR.
(K.R. Srinivasan and P.V. Suryanarayanan, AEET/234).

This report gives the details of the program CIR LOOP which has been written in Fortran language for CDC-3600. The program calculates the reactivity load and power output of a loop in CIR at the central position. The program is mainly based on the method given by McKay and Smith[1]. The CIR parameters have been taken from AECL-1443.

(1) McKay and Smith - AECL-456.