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MASTER

NUCLEAR ENERGY AGENCY COMMITTEE
ON REACTOR PHYSICS

REACTOR PHYSICS ACTIVITIES IN OECD COUNTRIES

JUNE 1973 - MAY 1974

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

NUCLEAR ENERGY AGENCY

38, boulevard Suchet, 75016 PARIS

The Organisation for Economic Co-operation and Development (OECD), which was set up under a Convention signed in Paris on 14th December, 1960, provides that the OECD shall promote policies designed:

- *to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;*
- *to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development;*
- *to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.*

The Members of OECD are: Australia, Austria, Belgium, Canada, Denmark, Finland, France, the Federal Republic of Germany, Greece, Iceland, Ireland, Italy, Japan, Luxembourg, the Netherlands, New Zealand, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States.

The OECD Nuclear Energy Agency (NEA) was established on 20th April 1972, replacing OECD's European Nuclear Energy Agency (ENEA) on the adhesion of Japan as a full Member. NEA now groups eighteen European Member countries of OECD, Japan and Australia, with Canada and the United States as Associated countries. The Commission of the European Communities takes part in the work of the Agency.

The objectives of NEA remain substantially those of ENEA, namely the orderly development of the uses of nuclear energy for peaceful purposes. This is achieved by:

- *assessing the future role of nuclear energy as a contributor to economic progress, and encouraging co-operation between governments towards its optimum development;*
- *encouraging harmonisation of governments' regulatory policies and practices in the nuclear field, with particular reference to health and safety, radioactive waste management and nuclear third party liability and insurance;*
- *forecasts of uranium resources, production and demand;*
- *operation of common services and encouragement of co-operation in the field of nuclear energy information;*
- *sponsorship of research and development undertakings jointly organised and operated by OECD countries.*

In these tasks NEA works in close collaboration with the International Atomic Energy Agency, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

LEGAL NOTICE

REACTOR PHYSICS ACTIVITIES IN OECD COUNTRIES

This document is a compilation of the reports on reactor physics activities during the period June 1973 - May 1974, presented at the Seventeenth Meeting of the Committee (held at CEN Cadarache, France in June 1974).

(Next Austrian progress report will cover the period June 1973 to May 1975. No progress reports were received from Belgium, Iceland, Ireland, The Netherlands and Turkey.)

Reactor Physics Activities in Spain

R. Ortiz Fornaguera

Experimental fast Reactor CORAL-I.

In the field of the fast neutron spectroscopy techniques, two proton recoil proportional counters filled with hydrogen and methane have been used to measure the scape neutron spectrum of the reactor. The results agreed with theoretical calculation performed using multigroup methods.

Using the solid state track recorder method, the absolute efficiency, ε, of the counters used in the count-to-count interval distribution technique has been measured. As we already knew the values of ε/β_{eff} through the count-to-count method, the value of β_{eff} has been evaluated. The result, $\beta_{eff} = (6.63+0.17)x10^{-3}$ agees with the calculated one using perturbation techniques, within the corresponding errors.

JEN-1 reactor.

A new logic unit for reactor safety has been developed and will be put in operation in the next months. With this unit the previous reactor instrumentation has been up to date modified.

JEN-2 reactor.

A new method has been developed to adjust the experimental probabilities $p_k(\tau)$ of obtaining k counts in a detector located in the center of the reactor. The experimental values have been adjusted by means of a generating probability function. These experimental probabilities $p_k(\tau)$ have been fitted for each one of the τ intervals of time. By this method a negative reactivity measurement can be completed in few minutes. The MINUS and MIGRAD codes have been used.

Reactor design and fuel management activities during 1973.

A fuel management study of the reload cycles for the 160 MWe Zorita PWR reactor, was conducted in assistance to the utility. Neutronic calculations using different codes and methods were performed for assesment of the design of a swimming-pool research reactor for Chile. Criticality calculations of different transportation configurations of fresh LWR's fuel were performed for safety purposes. Effects on neutronics characteristics of discharge burnup increase in the Vandellos GCR reactor were analyzed.

A number of computer codes for neutronics, thermalhydraulics and transient analysis, adquired through the CPL of NEA-OCDE at Ispra and from other sources, were implemented and tested on the UNIVAC-1106 computer of the JEN. Different codes and utility routines were developed and programmed for providing linkage between different codes and improving calculational gaps in different areas. Studies of different reactor types were performed, providing benchamark of the different codes and methods and experience in its use.

Reactor Physics Activities in Sweden, July 1973 - May 1974

By E Hellstrand

Theory

Thermal reactors

The MICBURN-BUXY code system for calculations on BA-assemblies has been verified against measurements on a fuel assembly which was irradiated in the Oskarshamn BWR. The burnup of individual fuel rods was determined by gamma-scanning and by use of the Nd148 method, and the isotopic composition of gadolinium in the BA-rods was measured. The predicted burnup distribution and gadolinium depletion agree with the measurements within the experimental uncertainties.

MICBURN has also been used for calculation of the microscopic burnup within fuel rods containing unpoisoned UO_2 or UO_2-PuO_2. This investigation shows that in ordinary fuel rods the average number densities of heavy nuclides and fission products is well predicted by use of a homogeneous treatment of the depletion.

Some new options have been added to the fuel assembly burnup code BUXY. A modification of the method for burnup calculations has made it possible to extend the burnup steps between spectrum calculations considerably, especially for fuel assemblies containing burnable absorbers. Typical step lengths are now 0.25-1 MWd/kg U for BA-assemblies.

A two-dimensional cell code, COXY, for calculation of neutron fluxes in xy-geometry has been developed. It is based upon the so-called transmission probability method. At the boundaries of the meshes, the fluxes are expressed by the double P_1-approximation. The neutron currents obtained from them are used as sources for calculating the transmission of neutron through each mesh. The angular dependence of the currents is given by three components for each of the outgoing and the incoming direction. The contribution to the outgoing current from the inner source, which is allowed to vary linearly within the mesh, is given by escape probabilities. The code has been tested against programmes based on collision probability and S_4 approximations. The calculations have been made on BWR boxes with and without a control rod, and also with rods containing uniformly distributed burnable absorbers. The size of the mesh grid can be at least a factor two larger in COXY than in the other two methods for the same accuracy. The time for computing transmission and escape probabilities is less than a tenth of the total computing time. The code is therefore an order of magnitude faster than the collision probability code used for comparisons.

A 69-group cross section library for thermal reactor calculations has been produced from ENDF/B-III. The SPENG library for fast reactor calculations has been used for obtaining fast and epithermal data. An improved version of the FLANGE-2 code has been used for obtaining thermal data. In the original FLANGE-2, the scattering matrix is calculated from point-wise data. In the Studsvik version, integration over primary and secondary groups is made using an appropriate weighting spectrum. The present library contains 57 nuclides.

Large-mesh (nodal) methods are generally used to calculate reactor core power distributions. Work has been started to create a solid theoretical foundation for large-mesh methods and concurrently to increase the accuracy of the methods. It has been possible to formulate and interprete a theory that largely preserves the relatively small amount of calculational work typical for earlier nodal schemes. The new methods will be well suited to experimental comparisons with results from small zero-power reactors as well as large reactors. The work is continuing.

Fast reactors

Integral data calculations have been made on nine uranium-loaded critical systems and on ten plutonium-loaded systems. The neutron data used have been processed from ENDF/B-III and are given in the SPENG library. During the processing, some discrepancies in group cross sections have been found when comparing with data obtained from MC^2. Some of these discrepancies have been attributed to an error in MC^2, but important discrepancies still remain unexplained. In the integral data, large deviations of k_{eff} from unity have been obtained for ZPR-III/32, ZPR-III/55 and ZEBRA/8C.

Breeding performance and fuel costs have been studied for gas-cooled and sodium-cooled breeders on a consistent basis using the fuel-cycle code EQUICYCLE. Fuel cost parameters established through this code is further used as input to long range strategy calculations regarding the role of fast breeders in the future power system in Sweden.

Experiments

Critical experiments

The analysis of the 1972-73 high temperature critical experiments in the KRITZ facility on lattices with mixed oxide fuel containing 1.5 % Pu have been completed. There are no large discrepancies between theory and experiments though the calculated values of reactivity and temperature coefficient seem to be systematically more negative than the experimental ones.

High temperature critical measurements on fuel with two different enrichments (about 2 and 3 %) fissile Pu were performed during the summar 1973 for reactor vendors outside Sweden. Analysis of these experiments have not yet been completed.

Negotiations about further KRITZ experiments on Pu-bearing fuel are going on. The measurements are planned for the autumn of this year.

Fission product activities

Residual power from fission products

The time behaviour of the residual power from fission product in reactor fuel shortly after a power shut-down is not well known. We are therefore planning to perform calorimetric measurements on small samples. The intention is to study the time period below 100 seconds after shut-down.

Calculations

The fission product activity program BEGAFIP utilizes two methods to calculate the total beta and gamma powers. At short decay times tabelled experimental values of beta and gamma production at different times after fission, integrated over the operating time, gives the desired values at a certain decay time. The second method calculates the number of atoms of each fission product by means of the Bateman equation whereupon the total beta and gamma powers are given by a simple summation. A combination of the two methods is necessary as the gamma spectra of the nuclides with short half-lives are not well known. The data library for the second method contains gamma radiation energies and intensities for each nuclide which means that in addition to the total gamma power also the fission product gamma radiation spectrum may be calculated.

BEGAFIP also yields the concentrations of various transuranium isotopes. Comparisons between results from BEGAFIP calculations and those based on the BUXY program are under way.