MCNP 4A COMPUTER CODE FOR CRITICALITY, NEUTRON FLUX AND SPECTRUM CALCULATIONS ON THE REACTOR VR-1

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MCNP is a general purpose Monte Carlo code for calculating the time-dependent continuous-energy transport of neutrons, photons, and/or electrons in three-dimensional geometries. Both fixed source and k_{eff} criticality problems can be solved and a number of output tally options are available. Data representations either can be fully or partially continuous or multigroup. The code is rich in variance reduction techniques that improve the efficiency of difficult calculations.

MCNP is used for many applications: nuclear criticality safety, radiation shielding, nuclear safeguards, detector design and analysis, nuclear well logging, personnel dosimetry, health physics, accelerator target design, medical physics and radiotherapy, including BNCT, PET and neutron and photon oncology, aerospace applications, radiography, waste disposal, decontamination and decommissioning, and both fission and fusion reactor design.

MCNP is used with MCNP Standard Neutron Cross Section Data Library Based in Part on ENDF/B-V (or ENDF/B-VI) general purpose files including some multigroup cross sections for MCNP.

MCNP was released in 1990 at Los Alamos National Laboratory in New Mexico, USA and from 1991 is distributed internationally from Radiation Shielding and Information Centre at Oak Ridge National Laboratory.

Department of Nuclear Reactors decided to buy this program at the end of 1995 and in the first half of 1996 year MCNP computer code version 4A with ENDF-V Cross Section Library was delivered from Oak Ridge National Labs. In the second half of the year the program was adopted in the Department of Nuclear Reactors and the first test and benchmark tasks were provided in the field of VR-1 both criticality and neutron flux calculations. Current VR-1 reactor fuel assemblies (IRT-2M) were modelled by several methods and criticality in a few core configurations was calculated, particularly the cores which were in the VR-1 reactor history of operation. The results were compared with both the measurements in the reactor and the previous calculation procedures (WIMS and DIFER codes).

Both the neutron flux and the spectrum were measured by the activation folios in the VR-1 reactor, the thermal neutron flux mainly by the thin gold folios in the field of well-known standard procedures. The folios were irradiated in various core positions, mainly in the dry vertical channels, and the activity was measured by HPGe detector. New measurement procedures were developed for some location within the core (gap between fuel assemblies, etc.). Both the fast neutron flux and the spectrum measurement were prepared in the preliminary experiments and will be extended for standard measurements in the near future.

All the results of criticality, neutron flux and spectra calculations and measurements are the first step on the way of standardization and validation calculation of the VR-1 reactor operation and will be continued in the next few years.

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