IJS Delovno Poročilo
IJS−DP−9701
Izdaja 2, maj 2008

CORD-2 Core Design System for PWR Type Reactors

M.Ravnik, A.Trkov, M.Kromar,
S.Slavič, I.Mele, B.Žefran, N.Železnik

	Ime in priimek	Datum	Podpis
Izdelal			
Pregledal			
Odobril			

Institut "Jožef Stefan", Ljubljana, Slovenija

Izvajalec: Institut "Jožef Stefan"
1001 Ljubljana
Jamova 39
Odsek za reaktorsko fiziko

Naslov poročila: CORD-2 Core Design System for PWR Type Reactors

Avtorji poročila: M.Ravnik
A.Trkov
M.Kromar
S.Slavič
I.Mele
B.Žefran
N.Železnik

Št. delov. poročila: IJS−DP−9701

Kopije: IJS knjižnica (1x)
QA arhiv odseka (1x + original)
avtorji

Komisija za pregled: dr. Igor Lengar
univ. dipl. fiz. Luka Snoj

Ljubljana, maj 2008

NASLOV POROČILA:

IJS−DP−9701
Maj 2008

CORD-2 Core Design System for PWR Type Reactors

M.Ravnik, A.Trkov, M.Kromar,
S.Slavič, I.Mele, B.Žefran, N.Železnik

KLJUČNE BESEDE:

Nuklearni projekt sredice, upravljanje z gorivom, tlačnovodni reaktor

Poročilo je podano v angleščini.

REPORT TITLE:

IJS REPORT 9701
May 2008

CORD-2 Core Design System for PWR Type Reactors

M.Ravnik, A.Trkov, M.Kromar,
S.Slavič, I.Mele, B.Žefran, N.Železnik

KEYWORDS:

Nuclear core design, Core Management, PWR

Contents

1 Purpose

CORD-2 system is intended for core design applications of pressurized water reactors. A number of code packages for such purposes exist, but they are commercial or proprietary. The main goal of the work presented herein was to assemble a core design system which could be used for simple calculations (such as frequently required for fuel management) as well as for accurate calculations (for example, core design after refuelling).

2 Description of the CORD-2 System

Parameters which appear in a nuclear core design can be calculated from a series of calculations for the effective multiplication factor k_{eff} and the power distribution for different core configurations and operating conditions. These can be obtained by solving the neutron diffusion equation over the domain of the reactor core. Using modern nodal codes, such a solution is not very computationally intensive, even in 3D geometry. Most of the time is spent in generating the coarse mesh homogenized few group cross sections. Since this is a crucial step in core design calculations, it is essential that simple and reliable procedures for producing the cross sections are available.

CORD-2 system consists of the following:

- a number of libraries with precisely defined format,
- a set of utility programs for library maintenance, data retrieval and other data handling tasks,
- calculational modules (for lattice transport, diffusion and thermohydraulics computations etc.).

In the CORD-2 package, two basic reactor physics codes are used, namely WIMS [1] and `GNOMER` [2]. WIMS is a well known and widely used lattice code. Version `WIMS-D/4` is available from the NEA Data Bank in Paris. `GNOMER` solves the neutron diffusion equation in three-dimensional cartesian geometry by the Green's function NOdal MEthod [3]. It also includes advanced features for cross section homogenization [5] and a simple thermohydraulics module from the `CTEMP` code [4], so that thermal feedbacks can be taken into account. The remaining parts of the CORD-2 software are utility codes which perform various functions, such as: library maintenance, data managements and input preparation for the reactor physics codes.

Standard FORTRAN-77 has been used as much as possible, with the minimum use of the job-control language for better portability. The package has been tested on VAX/VMS, OpenVMS and on PC.

The CORD-2 data libraries contain various classes of data, such as reactor core geometry, material composition etc. All codes of the package share the data from the same libraries. This reduces the user-input to the essential parameters which need to be varied in a particular calculational sequence and simplifies the quality assurance and quality control procedures. The list of libraries is the following:

DIMlib - dimensions and geometry description,

FASlib - fuel assembly ident, mass, enrichment and burnup history,

ISOlib - burnup-history dependent fuel isotopic inventory,

WIMlib - summary file for WIMS material assignment,

CORlib - core distributions data (e.g. power, temperature, burnup etc.),

RCFlib - reactivity coefficient library.

A block diagram representing the system is shown in Figure 2.1.

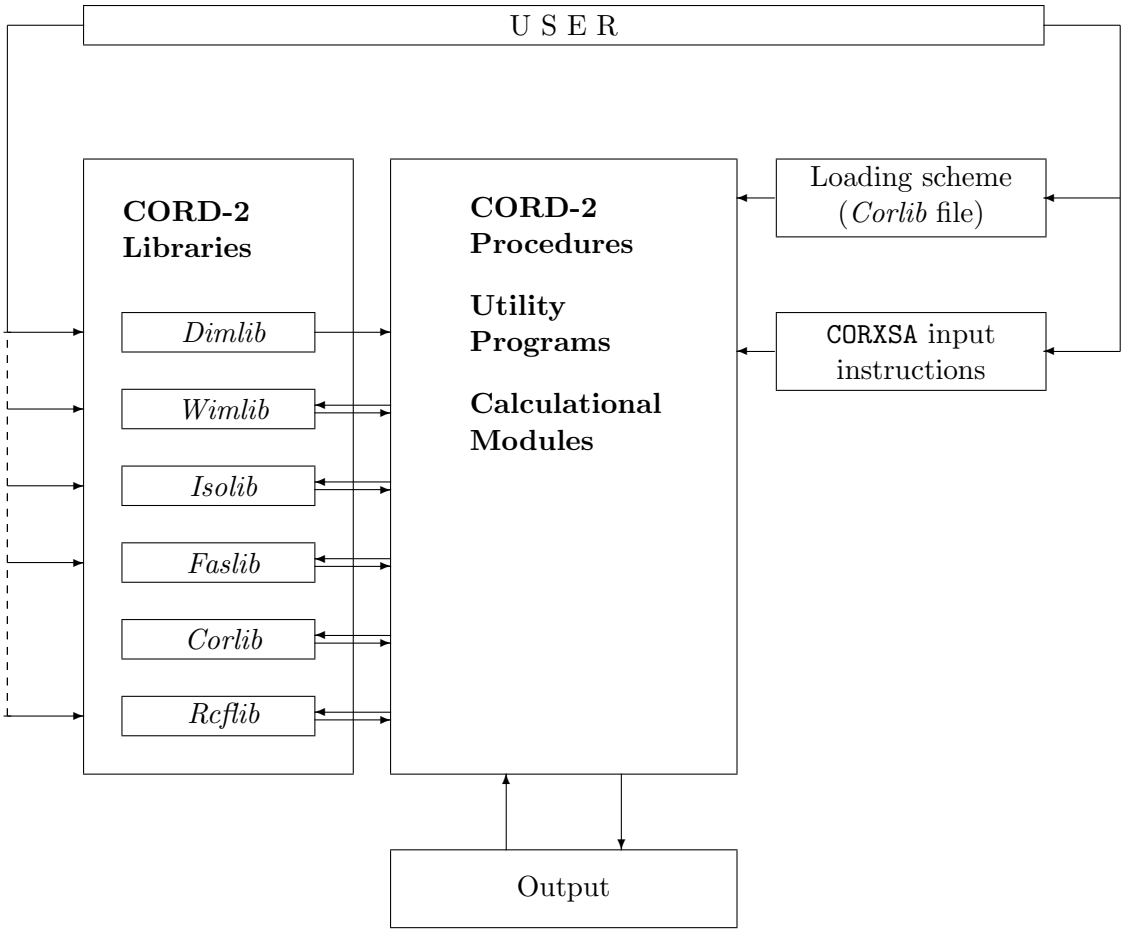

Figure 2.1: Block diagram representing the CORD-2 system

3 The methods of CORD-2

A solution for the whole core in 3D geometry is sought in several steps. On the reactor lattice level, transport methods are applied. From the solution, cell-homogenized multigroup cross sections are obtained. These are used in the calculations for a fuel assembly, where it is found that the solution in the multigroup diffusion approximation is adequate. Assembly-homogenized effective two-group cross sections can then be defined, which are suitable for solving the diffusion equation for the whole core by a coarse mesh nodal method.

Lattice cell homogenization: To homogenize a reactor lattice cell with WIMS, an array of 3×3 lattice cells is considered. The geometry is shown schematically in Figure 3.2 where the detailed internal structure of cells has been omitted for clarity. The outer cells f always contain fuel, while the central cell c may contain a fuel pin, a water hole, a burnable poison pin or another type of special cell. With a fuel cell in the centre, an extra layer of water is added on the outside of the cell array to preserve the total fuel assembly moderator to fuel ratio,

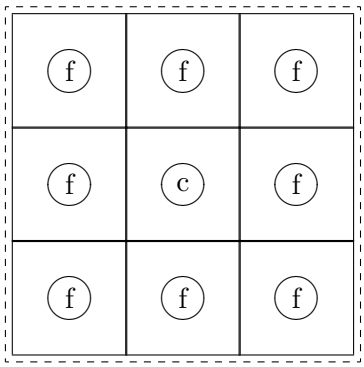

Figure 3.2: Lattice cell array configuration (schematic drawing).

A solution for the array of cells is obtained using WIMS. Since an accurate solution is required for the central cell only, the surrounding cells are smeared into concentric rings so that the problem is reduced to 1-D radial geometry. This is done automatically in WIMS. By default, transport calculations are performed in 32 energy groups.

From the WIMS results, the cell-homogenized few group cross sections are obtained for the central cell. The effective diffusion homogenization method (EDH) is used [5], which guarantees not only the reaction rate conservation but also the conservation of partial currents on the boundary of an equivalent homogeneous cell in the diffusion approximation.

The calculation for the array of cells is performed for each type of cell in an assembly, for each axial region (usually ten regions are considered) and for each assembly in the core. To reduce the computation time, core symmetry assumptions (i.e. core quadrant or octant) may be applied.

WIMS input instructions for all cell types in the core are generated automatically. WIMS outputs are processed by an auxilliary program to calculate the cell cross sections for fuel assembly calculations.

Fuel assembly calculations: An array of homogeneous cells which constitute a fuel assembly are considered. By default, ten energy groups are used. The solution is obtained in the diffusion approximation using `GNOMER`. Critical core conditions, for which k_{eff} is equal to one are usually of interest, therefore a *critical albedo* search is performed. Although such treatment is approximate and ignores the specific properties of the neighbouring fuel assemblies, it nevertheless represents the assembly surrounding correctly in an average sense. It is considered superior to the critical buckling search, which is commonly applied.

The result of the calculation are assembly-homogenized two-group cross sections. Like in the lattice cell case, the EDH method for homogenization is used. The calcu-

lation is repeated for each axial region of an assembly, for all assemblies in the core, considering core symmetries, if any.

The GNOMER input instructions for all assemblies are generated automatically.

Global core calculations: Calculation to determine the global core power distribution and the excess reactivity is the last step in the sequence. Solution is obtained with GNOMER in two energy groups and on a coarse mesh which usually corresponds to one node per assembly and ten regions in the core axial direction, with two additional regions for the axial reflector.

For a certain type of global core calculation the GNOMER input depends on the reactor core geometry and does not change. For this reason any automatic input generation is impractical and unnecessary.

Thermohydraulic feedbacks: In calculating the global core power distributions, the cross sections should be generated, taking into account the actual boron concentration and temperature distributions. However, these depend on the power distribution which we are trying to calculate. An iterative procedure is required. Due to the complex nature of the thermohydraulic feedback effects and because the flux distributions are sensitive to small perturbations, about ten iterations are necessary.

Fuel material burnup: The treatment of burnup in CORD-2 differs from the approach commonly adopted in other code systems, where burnup-dependent cross sections are parameterized. In CORD-2 the ISOlib library of isotopic composition vectors is constructed, parameterized as a function of fuel assembly operating conditions during burnup. An isotopic composition vector gives the masses (in *grams*) of all isotopes in the WIMS library per ton of heavy metal in fresh fuel. An axial section of a fuel assembly is assumed to have operated at constant, average operating conditions. An isotopic composition vector which corresponds to such conditions is retrieved from the ISOlib library by interpolation and is used to prepare the input for WIMS. The cell-homogenized cross sections are calculated from the WIMS output. When equilibrium xenon conditions are required, a burnup step of zero length is performed in WIMS. In this way, xenon poisoning is treated explicitly without further approximations.

At present, the parameters for tabulating the ISOlib library are the fuel assembly type, the enrichment, the moderator density, the fuel temperature and the average number of inserts during burnup.

4 Validation

CORD-2 system has been validated against the experimental data of the Krško NPP (Westinghouse 600 MWe) and the Almaraz benchmark [6]. Detailed results are presented in separate reports [7] and they indicate that CORD-2 system is sufficiently accurate in predicting the parameters which are required to insure safe operation of nuclear power plants.

5 References

[1] J.R.Askew, F.J.Fayers, P.B.Kemshell, *A General Description of the Code WIMS*, J.Br.Nucl.Energy Soc., **5**, p.564, (1966).

[2] A.Trkov, *GNOMER - Multigroup 3-Dimensional Neutron Diffusion Nodal Code with Thermohydraulic Feedbacks*, Institute Jožef Stefan, Ljubljana, Slovenia, IJS-DP-6688.

[3] A.Trkov, M.Najžer, L.Škerget, *Variant of Green's Function Nodal Method for Neutron Diffusion*, J.Nucl.Sci.Technol., **27**, 8, pp.766-777, (1990).

[4] M.Kromar, S.Slavič, A.Trkov: *CTEMP: A Code for Thermohydraulic Calculations*, Institute Jožef Stefan, Ljubljana, Slovenia, IJS-DP-6143.

[5] A.Trkov, M.Ravnik: Effective Diffusion Homogenization of Cross Sections for Pressurized Water Reactor Core Calculations, Nucl.Sci.Enging. **116**, 2, pp.86-95, (1994).

[6] M.J.Crijns (Scientific secretary): *In-Core Fuel Management Code Package Validation for PWR's*, Coordinated research project, International Atomic Energy Agency, Vienna, IAEA-TECDOC-815, Aug.1995.

[7] A.Trkov, M.Ravnik: *CORD-2 Package for PWR Nuclear Core Design Calculations*, Proceedings of the International Conference on Reactor Physics and Reactor Computations, Tel-Aviv, 23-26.Jan.1994, Ben-Gurion University of the Negev Press, (1994).