General Introduction to Methods of Physics Analysis and Physics Computer Codes

Benoit Arsenault, AECL

March, 2000

Methods of Physics Analysis and Physics Computer Codes

Must be able to model:

- **the perturbing effect of reactivity devices**
- **the static (time-independent) picture of the finite-reactor core**
- **dynamic (time-dependent) phenomena**
- **the effects of refuelling**

To carry out these tasks, reactor-physics analysis is heavily dependent on computer programs.

Basic Reactor-Physics Methods

To understand a reactor in detail we must know simultaneously

- energy distribution of the neutrons
- spatial distribution of reactions arising from neutrons born at a certain point.

The basic methods for obtaining this information:

- diffusion theory
- transport theory
- Monte Carlo methods
- modal method

Diffusion Theory

Diffusion theory makes two major assumptions:

a. probability of scattering is much greater than that for capture or fission,

b. scattering is isotropic (i.e., the probability of the scattered neutron going off in any direction is constant)

Considering neutron balance for a small volume dV one-energy-group neutron diffusion equation is:

$$\nabla . D \nabla \phi - \Sigma_a \phi + S = \partial n / \partial t$$

where:

- $\partial n / \partial t$ = **rate of change of neutron density in dV**
- $\nabla . D \nabla \phi$ = **net neutron leakage rate into or out of dV**
- $\Sigma_a \phi$ = **neutron absorption rate in dV**
- S = **production rate of neutrons in dV**

Diffusion Theory (con't)

The solution of the diffusion equation relates the neutron flux at any position in a region to:

- the material properties of the region,
- the neutron sources in that region,
- the leakage rate into or out of the region.

The form of the solution is dependent on the geometry of the assembly and a set of boundary conditions must be specified.

D in the equation is the diffusion coefficient and is related to the transport cross-section by:

$$D = \frac{1}{3\Sigma_{tr}}$$

Transport Theory

Transport theory makes no assumptions and in principle gives exact results.

However, too complicated to solve the transport equation exactly, so approximations are made:

- Neutron energies are divided into groups (usually 30 or more) to approximate a continuous energy distribution. For heavy water natural uranium reactors, a two-group calculation usually gives sufficient accuracy. In two-group theory, the neutrons are divided into a thermal group and an epithermal group in which neutrons are slowing down. This gives two diffusion equations to solve simultaneously.

Transport Theory (con't)

- **Within any group, the neutrons are assumed to obey diffusion theory with transport corrections, or transport theory.**
- **Neutrons slowing down below the lower energy limit of any group are treated as being absorbed with respect to that group and as being a source in one of the lower energy groups.**

Two-energy-group diffusion is the most popular methodused to calculate the neutron flux distribution in a CANDU reactor.

Transport Theory (con't)

For time-dependent situations, a two-group time-dependent neutron diffusion equation is used. It includes terms in the time derivative, and terms involving delayed neutrons. The latter are very important to take into account in the analysis of fast transients, such as loss-of-coolant accidents.

Monte Carlo Method

An alternative to the multi-group transport-theory method is the Monte Carlo method:

Follows the life history of a typical neutron

Probabilities are assigned to:

- distance that a particular neutron travels between collisions
- the type of reaction which occurs
- the change in direction and energy upon scattering

Monte Carlo Method (con't)

Fast computers are required since, to obtain sufficiently high statistical accuracy, tens or hundreds of thousands of neutron histories are necessary.

Currently extremely expensive to run (CPU time) and normally used only for detailed localized calculations or to benchmark lattice codes.

Modal Method

- Alternative to the diffusion method - used to calculate the 3D flux distribution in the core.
- Based on a linear superposition of pre-calculated basis function (flux modes).
- Unknowns are amplitudes of the basis functions.
- Flux modes are pre-calculated harmonics of the static diffusion equation; 10-20 harmonics used.
- Small number of unknowns, therefore equations can be solved very quickly.
- Fast calculation - suitable for application to reactor simulations.
- Not very accurate.

Lattice Codes

The lattice (or cell) code provides the neutronics of the basic (bare) lattice.

- Provides the cell-averaged two-group nuclear cross-sections versus burnup to the finite-reactor code.
- Provides spatial flux distribution in cell, and perhaps expected ring power and burnup
- Sometimes used to calculate reactivity coefficients of the core in a quick and approximate way, without a full reactor calculation.

Lattice Codes (con't)

Examples:

- **WIMS-AECL, DRAGON - use fundamental methods of neutron transport theory.**

- **POWDERPUFS-V - semi-empirical code uses correlations and recipes collected from experiments on heavy-water-moderated lattices in Chalk River research reactors such as ZEEP and ZED-2. Specifically designed for CANDU modelling.**

Reactivity-Device Codes

This type of code determines how reactivity devices perturb the properties of the nuclear lattice in their vicinity.

Provides "incremental cross-sections" which are used to correct the bare-lattice cross-sections for the effect of a reactivity device.

These incremental cross-sections are fed, together with the bare-lattice properties, to the finite reactor codes.

Reactivity-Device Codes (con't)

Examples:

- **DRAGON - use fundamental methods of neutron transport theory.**
- **MULTICELL - solves diffusion equation with current-to-flux (CFR's) to represent fuel and reactivity device**

Finite-Reactor Codes

These codes treat the entire region of the reactor core.

Most are based on two-energy-group neutron-diffusion theory. However, flux-mapping and modal-method codes are also used.

Finite-reactor codes are required to make the following calculations:

- **Static**
- **"Quasi-Static"**
- **Kinetic**
- **Fuel-Management**

Static Calculations

They provide:

- **spatial distribution of neutron flux in the core,**
- **power distribution,**
- **the reactor multiplication constant keff**

Uses:

- **reactivity worth of various reactivity devices by calculating the values of system reactivity with the device first inserted in the core, and then withdrawn.**

"Quasi-Static" Calculations

"Quasi-static" calculations are used to predict the static core conditions at specific times in a slow transient, such as that associated with changes in Xe-135 concentration.

This type of calculation solves:

a. the two-group time-independent neutron diffusion equation and,

b. the time-dependent I-135/Xe-135 kinetics equations

"Quasi-Static" Calculations

Uses:

- **To follow the time variation of the spatial distributions of Xe-135 concentration following a power manoeuvre, device movement, or refuelling operation.**
- **To mimic the actions of the zone-control system (bulk and spatial control)**

Kinetics Calculations

Kinetics calculations analyze the time-dependent behaviour of the reactor flux distribution.

Main applications:

- large-loss-of-coolant accident
- pressure-tube rupture

Kinetics codes must take into account delayed-neutron effects.

Kinetics Calculations (con't)

Example of kinetics codes:

CERBERUS - solves the time-dependent neutron diffusion equation in two energy groups, and is coupled to thermalhydraulics codes such as FIREBIRD or CATHENA.

SMOKIN - modal method less accurate but fast and includes reactor-regulating-system (RRS) algorithms.

Fuel-Management Calculations

For core design from the fuel-management point of view, and to track reactor operating histories.

Core Design:

- **fuelling schemes**
- **burnup regions**
- **time-average flux and power distribution used as a target by the fuelling engineer**
- **expected fuel burnup**
- **refuelling rates in the various core regions**
- **channel dwell times**

Fuel-Management Calculations (con't)

Core tracking:

- **keeps track of each bundle's irradiation**
- **models core burnup steps**
- **takes into account channel refeullings**
- **takes into account device movements**
- **outputs the flux and power distributions**
- **allows fuel engineer to monitor the channel and bundle powers and CPPF (channel power peaking factor)**
- **assists the fuelling engineer in selecting channels for refuelling in subsequent few days of operation.**

Fuel-Management Calculations (con't)

Codes used:

AECL and Point Lepreau

RFSP (Reactor Fuelling Simulation Program).

Ontario Hydro

SORO for core tracking and OHRFSP (the Ontario Hydro version of RFSP) for design.

Hydro-Quebec (Gentilly-2)

HQSIMEX is used.

All these codes have the capability to solve the two group neutron diffusion equation. The flux-mapping methodology is also incorporated in HQSIMEX and RFSP.

Fuel Management Calculations

The RFSP Code:

Solves the two-group 3D diffusion equations by a finite difference iterative technique using cell-centre fluxes.

Capable of generating nominal power distributions and simulating reactor operations, including refuelling and burnup steps.

- **It allows variable mesh spacings.**
- **It can use, at any boundary, extrapolated boundary conditions or symmetric boundary conditions.**
- **It is of modular design.**

Reactivity devices and structural materials are represented by incremental cross-sections which are added to the fuel cross-section of the affected lattice cells.

Types of Simulations

Homogeneous Calculations:

- The fuel properties (cross-sections) are assumed constant along the channel and constant within given regions of the core
- simple, but does not take into account the detailed effects of the fuelling scheme

Time-Average Calculations:

- The fuel cross-sections are ~~averaged over the~~ residence (dwell) time of the fuel at each point in the core.
- Accounts for fuelling scheme

Types of Simulations (con't)

Instantaneous Calculations:

- **random age distribution (based on time average beginning and end of cycle)**

  $$\omega(i,j,k) = \omega_1(k) + f(i,j)\ (\omega_2(k) - \omega_1(k))$$

- **produces hot spots**
- **patterned age distribution**
- **starting point for equilibrium ~~fuelling study~~**

Types of Simulations (con't)

Core Tracking

time history of the flux and power distributions is calculated at discrete time steps with the irradiation distribution incremented from the previous step using the previous flux distribution.

During design stage:

- used to simulate the initial transient from startup to equilibrium

Types of Simulations (con't)

- used to investigate the effect of various fuelling rules
- to obtain accurate estimates of maximum powers, discharge burnups, etc.

During reactor operation:

- to obtain bundle power, channel power, and bundle irradiation histories use to:
- select channels for refuelling
- ensure that channel and bundle powers are kept within specified limits
- evaluate burnup

RFSP Modules (Listed in Alphabetical Order)

*CERBERUS	Solves the time-dependent neutron diffusion equation.
*CERBRRS	Models the action of the Reactor Regulating System for the CANDU 6 Reactor.
*CHANGENAM	Changes the name of specified records in the direct-access file.
*CLOSE	Closes the RFSP direct-access file and ends program execution.
*COMPARE	Compares sets of measured and simulated detector readings.
*DATA	Reads specified data blocks into the RFSP direct-access file for model setup.
*DELETE	Eliminates specified data from the RFSP direct-access file.
*DFECTPROB	Calculates the fuel defect probabilities according to a modified Penn-Lo-Wood correlation.
*DISPLAY	Displays data for the *CERBERUS module.
*DLCSENSIT	Calculates the relative sensitivities of detector lead cables to detectors.
*DND	Calculates delayed-neutron-precursor data.

RFSP Modules (con't)

*ENDPLATES	Encodes bundle end-plate type in serial number for use in NUCIRC
*EXTRACT	Maintains data files for eventual production of bundle-power-history files.
*FLUXMAP	Calculates the amplitudes of harmonic flux modes for a synthesized flux distribution producing a least-squares fit to specified in-core detector readings.
*FPDTOHIST	Creates a starting point for the simulationof the reactor history using the history-based local parameter methodology, at a specified value of FPD.
*FREQUENCY	Tabulates frequency distribution of number of bundles against power and burnup.
*INSTANTAN	Simulates instantaneous core snapshots.
*INTEGRALS	Computes reaction rate integrated over the core and over specified regions.
*INTREP	Calculates the thermal flux at specified in-core points (eg., in-core detectors) by interpolation in the thermal-flux distribution.
*K-CHANGE	Processes data output from a simulation run to calculate the core reactivity increase upon refulling.

RFSP Modules (con't)

*PRNT MASS	Prints the index of contents of the RFSP direct-access file.
*PRTPWR	Provides data useful for selecting channels to be refuelled, as well as various statistical information.
*READMORE	Reads data for a specified harmonic mode from a specified file and transfers it to the direct-access file.
*READ CARD	Reads arbitrary data from a specified file in specified format and writes it to the RFSP direct-access file.
*READ FLUX	Reads a flux distribution generated in a specific format.
*READ TAPE	Transfers a sequential file originally written by *RITE TAPE to the RFSP direct-access file.
*RIPPLE	Calculates a "rippled" fundamental harmonic mode.
*RITE CARD	Reads data from the direct-access file and writes it to a specified file in specified format.
*RITE FLUX	Writes flux distribution from direct-access file in a specific format onto a specified file.

RFSP Modules (con't)

*RITE TAPE	Transfers entire RFSP direct-access file to a sequential file.
*RMICACSII	Reads the direct-access file form an ASCII file.
*RNOSACCI I	Transfers a sequential ASCII file origianally written by the Cyber 990 version of RFSP to the RFSP direct-access file.
*RNSES	Reads an ASCII file created from an NSES database.
*SIMULATE	Calculates a snapshot, or a series of snapshots, in the reactor operating history, with intervening burnup steps and refuellings according to user-specified input.
*SORT	Lists channels in descending order of age, burnup or irradiation, for use in selecting channels for refuelling.
*START	Begins new case execution and sets up case title heading.
*STORE	Transfers data from one area of the direct-access file to another - usually used to move data from working storage areas to areas where it can be retained for later use.
*SUMMARY	Computes various averaged quantities using results from a time-average calculation.

RFSP Modules (con't)

*TAVEQUIV	Creates an instantaneous model neutronically equivalent to the time-average model.
*TIME-AVER	Performs a time-average calculation of the flux and power distributions.
*TRANSG	Calculates compliance limit statistics over time based on *MARGINS data.
*TRIP_TIME	Calculates trip time for *CERBERUS LOCA analysis.
*USE DAF	Reads direct-access file created by *MAKE DAF and transfers it to local RFSP direct-access file *STORE
*WMICASCII	Copies the direct-access file STORE to an ASCII format.
*WNSES	Writes the direct-access file STORE in NSES ASCII format.
*WROPFILE	Writes files for ROP (regional-overpower-protection) analysis with ROVER.