LECTURE 12: RESEARCH REACTORS REQUIREMENTS

MODULE OBJECTIVES:

At the end of this module, you will be able to list a :

1. **Example of uses of research reactors**
2. **The methods used to calculate physics parameters in research reactors**
3. **Examples of code and data validation approaches used**

- The only common characteristic requirement of the many types of research reactors is the need for flexibility of the reactor Physics codes used to analyze their fueling and safety: A research reactor is basically a large neutron source for a variety of experimental or isotope production irradiations. Table A gives a few examples of their potential uses.

Table A: Examples of Research Reactor Uses

USE	FACILITY
Code Validation	Zero Power Lattice with Varible configuration (ZED-2)
Fuel Testing	High Flux Core, with Loops Containing Variable Fuel/Coolant (NRU)
Isotope Production, Neutron Activation	High Flux Core, with Penetrations for Irradiation of Targets (MAPLE)

- The variability of configurations and targets to be introduced in the core, combined with the need for accurate knowledge of the flux and power effects of these variable loads, requires a variety of calculational approaches.

- The majority of the codes used are similar to those used in power reactors: the fueling engineer needs the same information for research and power reactors. However, there is more frequent need for more complex approaches than few group diffusion models, such as multigroup transport calculations.

- Fortunately there are available many international codes supported by national laboratories with the capability to validate them.

METHODS AND CODES USED

COMPUTER CODES AND METHODS FOR STATIC NEUTRON PHYSICS CALCULATIONS

Each organization have their computer codes for performing:

- **Cell calculations: These codes are used to perform spectral calculations in the cells and to produce condensed few-group constants, macroscopic absorption and fission cross sections, and macroscopic reaction rates for use in the core calculations. Two calculational methods are generally used, discrete ordinates transport theory and the collision probability form of the transport equation.**

- **Core calculations: Three calculational methods are generally used, diffusion theory, discrete ordinates transport theory and Monte Carlo theory, to solve the Boltzmann transport equation. The CIAE also use the nodal method to calculate criticality, flux and power distributions, and reactivity coefficients.**

- **As shown in Table 2, the key core performance parameters are generally calculated using different methods to provide independent verification of the results. The only exception is in the case of fuel depletion calculations where only diffusion theory is generally used to estimate the core burnup.**

- **Most of the computer codes listed in Tables 1 and 2 have a long history of applications in many projects. Nevertheless, the SQA programs for many recent research reactor projects (e.g., ANS and IRF) require that the computer codes used for design calculations and safety analyses be verified and validated for the specific applications,**

- **Verification of the computer codes are addressed as follows:**

  - **AECL relies on benchmark problems, inter-code comparisons and verification reports from code maintainers; the software includes in-house development (e.g., WIMS-AECL) and international sources (e.g., 3DDT, MCNP and DANTSYS),**

Table 1: Summary of Computer Codes and Methods

METHOD	AECL	CIAE	JAERI	ORNL	Siemens
Cell Calculation - Collision probability - transport	WIMS-AECL	WIMS-D4 PASC-1	 ANISN TWOTRAN [22]	AMPX/ SCALE	RSYST, MONSTRA
Core calculation - diffusion	3DDT [15]	CITATION [19] EXTERMINATOR - 2[20]	CITATION	VENTURE [23]	DIF1D, DIF2D DIXY
- Monte Carlo	MCNP [16] KENO [13}	MCNP	MCNP	MCNP KENO	MORSE-K MOCA
- transport	DANTSYS [17,18]	ANISN [21] DOT3.5	ANISN TWOTRAN	DORT [24]	IANISN, SN1D DOT
- nodal method		PSUI - LEOPARD/ NGMARC			

Table 2: Summary of the Codes Used to Calculate Key Physics Parameters

Parameter	AECL	CIAE	JAERI	ORNL	Siemens
K-effective	3DDT MCNP	MCNP ANISN DOT3.5 PSUI- LEOPARD/N GMARC	CITATION TUD TWOTRAN MCNP	MCNP KENO DORT	MORSE-K MOCA DOT
reactivity worth	3DDT MCNP DANTSYS	MCNP	ANISN TWOTRAN MCNP	MCNP	MORSE-K MOCA DOT
reactivity coefficients	3DDT	CITATION ANISN DOT3.5	CITATION TUD ANISN TWOTRAN	VENTURE MCNP DORT	DIF2D
flux and power distribution	3DDT MCNP	CITATION MCNP ANISN DOT3.5	CITATION TUD ANISN TWOTRAN	VENTURE MCNP DORT	DIF2D MORSE-K DOT
fuel depletion	3DDT/ FULMGR	CITATION 2DFGD EXTREMINATOR-2	CITATION TUD	VENTURE/ BURNER	RSYST MARS

- CIAE relies on software obtained from international sources (e.g., RSIC, NESC, NEA),
- JAERI has verffied SRAC system using international benchmark problems,
- ORNL relies on verification reports from code developers (e.g., ORNL, LANL) for the ANS Project, and
- Siemens relies on inter-code comparisons between the RSYST and MARS systems for FRM-II.

- Validation of the computer codes are addressed as follows:
  - AECL: Code validation relies on comparisons against benchmark problems, inter-code comparisons and comparisons against critical experiments.
  - CIAE relies on IAEA benchmarks.
  - JAERI relies on IAEA benchmark critical experiments and commissioning data.
  - ORNL has validated their computer codes against:
    - Los Alamos critical mass data for enriched uranium in bare H_20- and D_20reflected critical experiments, ORNL H_20-solution critical experiments, and D_20-moderated, natural uranium ZEEP critically-buckled lattices,
    - FOEHN *critical experiments* [29] *to validate predictions* from MCNP, VENTURE/BURNER, DORT and KENO,
    - ANS critical experiments to supplement validation from the FOEHN experiments, and
    - HFIR and ILL operating data to validate the fuel depletion calculations.
- Siemens has commissioning data from RSG-GAS-30 (Indonesia) for validation.

NUCLEAR DATA LIBRARIES

Nuclear data librarie listed in Table 3 are being used for physics calculations. Verification and validation of the nuclear data libraries are combined with the verification and validation of the codes.

Table 3: Summary of Nuclear Data Libralies and Computer Codes

LIBRARY	AECL	CIAE	JAERI	ORNL	Siemens
CRSL-IV		PASC-1			
ENDF/B-IV	KENO	MCNP	ANISN, PIJ TWOTRAN		MONSTRA CGM
ENDF/B-V	WIMS-AECL MCNP		MCNP	AMPX/ SCALEDO RT VENTURE/ BURNER MCNP, KENO	CGM
ENDF/B-VI	WIMS-AECL				
GAM/ THERMOS		PASC-1			
GAM/ THERMOS					
JEF1					CGM
JENDL-2			PIJ, TWOTRAN		
VITAMIN-C		PASC-1			
WINFRITH	WIMS-AECL	WIMS-D4			