Review of Worldwide Reactor-Physics Codes and Their Applicability to CANDU® and ACR-1000® Analysis[1]

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Abstract

This paper reviews and summarizes the current stochastic (Monte Carlo) codes and deterministic codes used in reactor-physics analysis, with particular emphasis on those aspects relevant to CANDU® and ACR-1000® analysis. The review concentrates on the main features of each reactor-physics toolset with respect to state-of-the-art reactor-physics and numerical modelling aspects, as well as to new methodologies. Where relevant, the applicability of these codes to CANDU and ACR-1000 analysis is covered as well.

1. Introduction

Reactor-physics codes can be divided into two categories: stochastic (Monte Carlo) codes and deterministic codes. Although continuous-energy Monte Carlo codes like MCNP5 [1] have been available for a number of years, deterministic codes based on approximate methods are still used for "standard" reactor-physics calculations based on the fact that the stochastic codes are still not sufficient enough for the solution of the time-dependent problems such as burnup and transients, etc. The deterministic codes are based on separating the total neutronics problem into two or more levels of analysis, one level of which is classified as the lattice calculation and the second level of which comprises the whole-core analysis that applies homogenized properties based on the results of the lattice-cell and supercell calculations.

There have been significant time and effort invested by the Canadian nuclear industry to develop and maintain the reactor-physics codes that are used in CANDU[2] industry [2]. In particular, the development of the 2-D lattice code WIMS-AECL [3][4], the 2-D/3-D neutron-transport code DRAGON [5], and the 3-D fuel management and core-analysis code RFSP [6][7], as well as maintaining the reference data libraries, have received significant attention by the industry in the past decades. Though the current physics codes have been in widespread use for safety and/or operational analysis over many years in the CANDU industry, it is necessary to review the worldwide reactor-physics codes and to evaluate how these codes can be applied to current and future CANDU applications.

[2] CANDU is a registered trademark of Atomic Energy of Canada Limited (AECL).

2. Review of worldwide reactor-physics codes for CANDU and ACR-1000[3] application

This section provides a high-level summary of some reactor-physics codes, with particular emphasis on those aspects of relevant to CANDU and ACR-1000 analysis. The review concentrates on the main features of each toolset[4] with respect to state-of-the-art reactor-physics and numerical modelling aspects as well as the methodologies. Where relevant, the applicability of these codes to CANDU and ACR-1000 analysis is covered as well.

2.1 Requirement specifications of a reactor-physics toolset for CANDU and ACR-1000 applications

In order to perform reactor-physics analysis for CANDU and ACR-1000, a toolset should be able to meet the following requirements as minimum:

- Qualified libraries containing D_2O data and self-shielded data for burnable poisons such as Dy, Gd, and Hf;
- 2-D transport calculations for the fuel cluster geometry;
- 3-D transport calculations for the reactivity devices;
- Fuel shuffling and on-line refuelling calculation;
- History-based local-parameter method (with moderator and coolant be treated as separate parameters) in the core-analysis code for the calculation of lattice cross sections;
- Static and kinetics neutron-diffusion calculation for the eigenvalue and fixed-source problems;
- Coupling with a system thermal-hydraulic code for transient analysis;
- Modelling of the Reactor Regulation System (RRS) in quasi-static and dynamic modes;
- Modelling of detectors;
- Flux-mapping calculations; and
- Time-average calculation for the design of the equilibrium core.

2.2 Review of the current reactor-physics toolset used in CANDU industry

The WIMS-AECL, DRAGON and RFSP are the three Industry Standard Toolset (IST) codes currently used in the CANDU industry. WIMS-AECL is a 2-D lattice code used for transport calculations and cross-section condensation in CANDU lattices. DRAGON is a 2-D/3-D cell and supercell code used for 3-D transport calculations of the incremental cross sections of CANDU reactivity devices, which are perpendicular to the fuel channels. RFSP is a core-analysis code for CANDU full-core 3-D static and dynamic analysis.

Main features of the lattice code WIMS-AECL:

- Library: 89-group libraries based on ENDF/B versions V, VI and VII
- Geometry: 2-D, single-cell with fuel cluster, multiple cells with fuel cluster, sector, off-center sagged model

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[3] ACR-1000 is a registered trademark of Atomic Energy of Canada Limited (AECL).
[4] A toolset is defined as a production code(s) covering the lattice-cell, supercell, and full-core calculations. A toolset could be a suite of codes (like WIMS-AECL/DRAGON/RFSP) or a single code (like MCNP).

- Method of Solution: 2-D Collision Probability (CP) method in full energy-group structure
- Resonance Treatment: Advanced self-shielding capabilities (capability to represent distributed self-shielding effects)
- Programming Features: Modular design, Re-written with FORTRAN90/95
- Other Features:
- Critical buckling search
- Simple input cards for production run
- Application: Industry code[5] developed for CANDU and ACR-1000

Main features of the supercell code DRAGON:

- Library: multi-group libraries based on ENDF/B versions V and VI
- Geometry: Capabilities to define any complicated 2-D and 3-D cluster geometry, multiple cells with cluster
- Method of Solution: 2-D/3-D CP method and Method of Characteristics (MOC)
- Resonance Treatment: advanced self-shielding capabilities including subgroup method
- Programming Features: Modular design, FORTRAN77, GANLIB, 64-bit capability
- Other Features:
- Comprehensive SPH (Super Homogenization) technique
- Critical buckling search
- Visualization of 2-D geometries
- Flexibility (users can implement some computational scheme in CLE-2000, without having to recompile the code)
- Application: Academic code[6] developed for general application, qualified as an IST toolset for CANDU and ACR-1000

Main features of the core-calculation code RFSP:

- Problem Solved: Two-group static and kinetics neutron-diffusion equations for the eigenvalue and fixed-source problems, fluxmapping equations
- Geometry: 3-D, Cartesian geometry
- Method of Solution: Finite-difference method for the diffusion theory
- Cross-Section Model: micro-depletion method, simple-cell method (SCM), multicell methodology
- Other Features: Coupled to a thermal-hydraulics code, such as CATHENA [8]
- Application: Qualified industry code developed for CANDU and ACR-1000

In addition to the applications of IST codes, greater and greater use is being made of Monte Carlo analysis; MCNP5 is being used for special applications in both lattice and full-core calculations, and for benchmarking of other codes.

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[5] An industry code implies that the code was used in the industry with extensive qualification process.
[6] An academic code implies that the code was developed at the universities for research purpose with limited qualification process.

2.3 Review of worldwide reactor-physics codes

For a better understanding of the methods mentioned in the review, the computational methods commonly used in the reactor-physics codes and the evolution of most known worldwide lattice codes are shown in Figure 1 and Figure 2, respectively. Note that all reactor-physics codes summarized in this section are listed below in alphabetic order.

2.3.1 AEGIS/SCOPE2 (Academic Toolset)

The AEGIS/SCOPE2 System [9] is a next-generation code system developed by Nuclear Engineering Ltd (NEL), Nuclear Fuel Industries Ltd (NFI) in cooperation with Nagoya University in Japan.

Main features of the lattice code AEGIS:

- Library: Ultra-fine-group (32000) library
- Geometry: Flexible geometry handling in 2-D
- Method of Solution: MOC in 172-group structure
- Application: Academic code, maybe applicable to CANDU and ACR-1000

Main features of the core-calculation code SCOPE2:

- Problem Solved: Multi-group SP_3 equation
- Geometry: 3-D, Cartesian geometry
- Method of Solution: Nodal method
- Cross-Section Model: Macro- and micro-depletion models, pin-by-pin calculation
- Other Features: Fully parallelized using MPI (Message Passing Interface)
- Application: Academic code developed for BWR (Boiling-Water Reactor) application

2.3.2 APOLLO2/ARTEMIS/TRIPOLI4 (Industry Toolset)

The APOLLO2 lattice code [10] is developed at the Commissariat à l'Énergie Atomique (CEA), France, and is currently used for calculation of all types of thermal lattices available in France. The reactor-core simulator ARTEMIS [11] has been entirely developed by Areva NP in-house based on the three existing production reactor-core simulators CRONOS2, CASCADE, and NEMO [11][12]. It is a new "converged" next-generation reactor-core simulator. The TRIPOLI4 [13] Monte-Carlo code is a continuous-energy or multi-group Monte-Carlo code. It solves the transport equation for neutral particles in general 3-D geometric configurations.

Main features of the lattice code APOLLO2:

- Library: Multi-group libraries based on JEF

- Geometry: Complex 2-D geometry including cluster geometry
- Method of Solution: 1-D/2-D CP method and discrete ordinate (Sn) method, 2-D/3-D MOC
- Resonance Treatment: Advanced self-shielding capabilities (capability to represent distributed self-shielding effects)
- Programming Features: Modular design, FORTRAN
- Other Features:
- Critical buckling search
- SPH homogenization
- Flexibility (users can implement any computational scheme in GIBIANE, without having to recompile the code)
- Any complex 2-D geometry can be input with the graphical user interface SILENE
- Application: Industry code developed for general application, can be used for CANDU and ACR-1000

Main features of the core-calculation code ARTEMIS:

- Problem Solved: Multi-group static and kinetics neutron-diffusion equations
- Geometry: 3-D, Cartesian geometry
- Method of Solution: Classic NEM (Nodal Expansion Method) in both polynomial and semi-analytical variants for diffusion theory and SP_3
- Cross-Section Model: micro-depletion method (uses a generalized B-spline approach)
- Other Features:
- Depletion solver of all nuclide reactions including feedback reactions like α-decay or n-2n transitions
- Different solvers for the depletion equations: ORIGEN classical matrix exponential approach, Runge-Kutta approach, and Krylov-subspace-based matrix exponential approach
- Fuel rod model for fuel analysis
- Pin-power reconstruction
- Coupled to a thermal-hydraulics code, such as COBRA3-CP
- Application: Industry code developed for LWR (Light-Water Reactor) application

Main features of the Monte-Carlo code TRIPOLI4:

- Problem Solved: Neutron, photon, coupled neutron-photon transport equations with fixed sources
- Library: Continuous or multi-group energy libraries
- Geometry: General 3-D geometric configurations
- Method of Solution: Monte Carlo method
- Programming Features: Programmed in C++
- Application: Industry code for general application, applicable to CANDU and ACR-1000

2.3.3 CASMO-5/SIMULATE-4 (Industry Toolset)

The CASMO-5/SIMULATE-4[14][15] System is developed by Studsvik Scandpower Inc in USA.

Main features of the lattice code CASMO-5:

- Library: 586-group neutron and 18-group gamma libraries based on JEF2.2 and ENDF/B-VI
- Geometry: 2-D geometry (unable to model fuel cluster)
- Method of Solution: 2-D MOC calculation in a few-group structure
- Resonance Treatment: Advanced self-shielding capabilities (capability to represent distributed self-shielding effects)
- Programming Features: Re-written with FORTRAN90/95, 64-bit capability
- Other Features
- Simple input cards for production run
- Application: Industry code developed for LWR application

Main features of the core-calculation code SIMULATE-4:

- Problem Solved: Multi-group static and kinetics neutron-diffusion equations
- Geometry: 3-D, Cartesian geometry
- Method of Solution: analytical nodal method for diffusion theory and SP_3
- Cross-Section Model: micro-depletion method
- Other Features:
- Pin-power reconstruction
- Ad-hoc approximations reduced, but still exist
- Coupled to a thermal-hydraulics code
- Application: Industry code developed for LWR application

2.3.4 DRAGON/DONJON (Academic Toolset)

The DRAGON/DONJON toolset [5][16] is developed by Ecole Polytechnique de Montreal in Canada. The main features of the lattice code DRAGON has been described in Section 2.2.

Main features of the core-calculation code DONJON:

- Problem Solved: Multi-group static and kinetics neutron-diffusion equations for the eigenvalue and fixed-source problems
- Geometry: 3-D, Rectangular and hexagonal geometries
- Method of Solution: Finite-difference method or finite-element method for diffusion theory and SP_N
- Cross-Section Model: Macro-depletion method, History-based method such as direct use of DRAGON to perform lattice calculations inside DONJON

- Programming Features: Modular design, FORTRAN77, GANLIB
- Other Features:
- Flexibility (users can implement some computational scheme in CLE-2000, without having to recompile the code)
- Application: Academic code developed for general application, applicable to CANDU and ACR-1000

2.3.5 HELIOS (Industry Code)

HELIOS [17] is a multi-group two-dimensional transport theory code developed by Scandpower Inc in Norway. Its main characteristics are:

- Library: 190-group libraries based on ENDF/B version VI
- Geometry: General 2-D geometric capabilities (capabilities to define any complicated 2-D geometry including cluster)
- Method of Solution: Current Coupling Collision Probabilities (CCCP)
- Resonance Treatment: Subgroup method
- Other Features:
- Visualization of 2-D geometries
- Neutron and gamma transport
- Application: Industry code for general application, applicable to CANDU and ACR-1000

2.3.6 MCNP5 (Industry Toolset)

MCNP5 [1] is a general-purpose Monte-Carlo N-Particle code developed by LANL in the USA over 40 years. Its main characteristics are:

- Problem Solved: neutron, photon, electron, or coupled neutron/photon/electron transport equations
- Library: Continuous or multi-group energy libraries
- Geometry: Arbitrary 3-D configuration
- Method of Solution: Monte Carlo method
- Programming Features: Re-written with FORTRAN90, 64-bit capability
- Other Features:
- Visualization of 2-D geometries
- Application: Industry code for general application, applicable to CANDU and ACR-1000

2.3.7 NESTLE (Academic Code)

NESTLE (Nodal Eigenvalue, Steady-state, Tansient, Le core Evaluator) [18] is a core-calculation code developed at the North Carolina State University in 1990s. Its main characteristics are:

- Problem Solved: Two-group and four-group static and kinetics neutron-diffusion equations for the eigenvalue and fixed-source problems
- Geometry: 3-D, Rectangular and hexagonal geometries
- Method of Solution: Finite-difference method and nodal method
- Cross-Section Model: macro-depletion method, micro-depletion method (burnable isotopes limited to the principal fissile and fertile isotopes, and lumped fission products)
- Other Features: Embedded thermal-hydraulic feedback model
- Application: Academic code developed for LWR application, a CANDU version was developed for CANDU application

2.3.8 NGM (Academic Code)

NGM (next generation method) [19] is developed by Mitsubishi Heavy Industrial (MHI) in Japan. Its main characteristics are:

- Problem Solved: Multi-group transport theory
- Geometry: 3-D, Complex geometry such as assembly gap and irregular fuel rod arrangement
- Method of Solution: Coupling 2-D heterogeneous MOC calculation with 3-D homogeneous pin-by-pin nodal S_N calculation
- Other Features: SPH-corrected pin-cell homogeneous cross sections
- Application: Academic code developed for LWR application

2.3.9 PARCS (Academic Code)

PARCS (Purdue Advanced Reactor Core Simulator) [20] is the core-analysis code used in CAMP (Code Assessment and Maintenance Program). Its main characteristics are:

- Problem Solved: Two-group static and kinetics neutron-diffusion equations for the eigenvalue and fixed-source problems
- Geometry: 3-D, Rectangular, hexagonal and cylindrical geometries
- Method of Solution: Finite-difference or nodal method for diffusion theory and SP_3
- Cross-Section Model: Macro-depletion method (multidimensional cross-section table)
- Other Features:
- Pin-power reconstruction
- Coupled to thermal-hydraulic codes RELAP5 and TRACE
- Application: Academic code developed for LWR application, used for some specific ACR-1000 applications such as inter-code comparisons

2.3.10 WIMS9/PANTHER/MONK (Industry Toolset)

WIMS9 [21] is the latest generation of the lattice code WIMS. It combines all reactor physics methods developed by Serco Assurance in UK over 30 years. PANTHER is a modern full-core code developed over the past 10 years at British Energy and BNFL Magnox Generation to improve the performance of its own nuclear reactor plant. The MONK [22] Monte-Carlo code is a multi-group Monte-Carlo code. It solves the transport equation for neutral particles in general 3-D geometric configurations.

Main features of the lattice code WIMS9:

- Library: 172-group libraries based on JEF
- Geometry: Flexible geometry handling in 2-D
- Method of Solution: CP or MOC in 2-D; Diffusion theory (SNAP), Monte Carlo (MONK), MOC (CACTUS in 3-D), and a hybrid Monte Carlo method (MAX) in 3-D
- Resonance Treatment: Subgroup method
- Application: Industry code for general application, applicable to CANDU and ACR-1000

Main features of the core-calculation code PANTHER:

- Problem Solved: Multi-group static and kinetics neutron-diffusion equations
- Geometry: 3-D, Rectangular and hexagonal geometries
- Method of Solution: Finite-difference method or nodal method
- Cross-Section Model: macro-depletion method
- Other Features:
- Pin-power reconstruction
- Application: Industry code developed for LWR application, used for some specific ACR-1000 applications such as inter-code comparisons

Main features of the Monte-Carlo code MONK

- Library: Hyper-fine-group (13,193 fine groups together with a four-group sub-group pre-shielding treatment) libraries
- Geometry: General 3-D geometric configurations
- Method of Solution: Monte Carlo method
- Other Features:
- Treatment of the hole geometry
- “Super-history” technique used to reduce variance in big criticality cases
- Application: Developed for general application, applicable to CANDU and ACR-1000

2.3.11 UNIC (Academic Code)

The code UNIC (Ultimate Neutronic Investigation Code) [23] is currently under development at Argonne National Laboratory (ANL). The aim is to provide a neutronic solver with the same geometric flexibility as Monte Carlo codes and without approximations associated with the common multi-step approach currently used. Its main characteristics are:

- Problem Solved: Multi-group neutron-transport equations
- Library: Ultra-fine-group (upwards of 10,000) library
- Geometry: 3-D, Flexible geometry option as in the case of Monte Carlo codes
- Method of Solution: spherical harmonics method (P_NFE), MOC, and S_N method
- Other Features:
- Common multi-step procedures of group collapse and homogenization are avoided
- CUBIT package is used as the primary mesh generation tool for reactor analysis
- Application: Academic code under the development for general application.

3. Summary

Based on the review summarized in previous section, the following conclusions can be made:

First, deterministic codes based on approximate methods are still used for “standard” reactor-physics calculations based on the fact that the stochastic codes are still not sufficient enough for the solution of the time-dependent problems such as burnup and transients.

Second, none of the codes listed in Section 2.3 has all basic features currently available in WIMS-AECL/DRAGON/RFSP suit of codes for CANDU and ACR-1000 applications. The current IST codes (WIMS-AECL/DRAGON/RFSP) will continue to form the basis of the CANDU and ACR-1000 physics computational scheme. In the area of 2-D/3-D lattice-cell and supercell calculations, the new versions of WIMS-AECL and DRAGON are quite representative of the current state-of-the-art for production codes. In the area of 3-D core-analysis calculations, RFSP is commonly applied for CANDU and ACR-1000 problems. However, RFSP could be enhanced to have multi-group solvers for the diffusion or SP_N theory by using the modern homogenization technique. The fuel-pin power reconstruction capability is another enhancement area that is currently being considered in RFSP code development.

Third, the deterministic lattice codes such as AEGIS, APOLLO2, DRAGON, HELIOS, and WIMS9 are applicable for CANDU and ACR-1000 applications. However, qualification of these codes for CANDU and ACR-1000 applications has not yet been performed. Table 1 shows the summary of the worldwide lattice codes and their features and applicability to CANDU analysis.

Forth, the core-analysis codes such as ARTEMIS, DONJON, PARCS, NESTLE, PANTHER, SIMULATE-4 can be used to perform some snapshot analysis for CANDU and ACR-1000, for the design verification or inter-code comparison. However, qualification of these codes for CANDU and ACR-1000 applications has not yet been performed. Table 2 shows the summary of the worldwide reactor core codes and their features and applicability to CANDU analysis.

Finally, the stochastic codes such as MCNP5, MONK, and TRIPOLI4 can be used to perform some snapshot lattice and full-core calculations for CANDU and ACR-1000. However, qualification of these codes for CANDU and ACR-1000 applications has to be performed.

4. Acknowledgement

The authors wish to thank all reviewers, E. Varin and D. Altiparmakov in particular, for their careful and in-depth review of this paper.

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Table 1 Summary of the worldwide lattice codes and their features

Lattice Code	Library	Geometry	Solver	Self Shielding Method	Industry code	Applicable to CANDU
AEGIS	Ultra-fine-group (32000)	Flexible geometry handling in 2-D	MOC in 172-group structure	Not available	No	Yes
APOLLO2	Multi-group libraries based on JEF	Complex 2-D geometry including cluster geometry	1-D/2-D CP method and Sn method 2-D/3-D MOC	Advanced self-shielding capabilities: equivalence method and sub-group method	Yes	Yes
CASMO-5	586-group neutron and 18-group gamma libraries based on JEF2.2 and ENDF/B-VI	2-D geometry (unable to model fuel cluster)	2-D MOC	Advanced self-shielding capabilities	Yes	No
DRAGON	multi-group libraries based on ENDF/B versions V and VI	Complicated 2-D and 3-D cluster geometry, multiple cells with cluster	2-D/3-D CP method and MOC	advanced self-shielding capabilities including subgroup method	Yes	Yes
HELIOS	190-group libraries based on ENDF/B version VI	General 2-D geometric (including cluster)	CCCP	Subgroup method	Yes	Yes
NGM	Not available	3-D, Complex geometry	Coupling 2-D heterogeneous MOC calculation with 3-D homogeneous pin-by-pin nodal S_N calculation	Not available	No	No
WIMS9	172-group libraries based on JEF	Flexible geometry handling in 2-D	CP or MOC in 2-D; Diffusion theory (SNAP), Monte Carlo (MONK), MOC (CACTUS in 3-D), and a hybrid Monte Carlo method in 3-D	Subgroup method	Yes	Yes
UNIC	Ultra-fine-group (upwards of 10,000) library	3-D, Flexible geometry option as in the case of Monte Carlo codes	Spherical harmonics method (P_NFE), MOC, and S_N method	Not available	No	Yes
WIMS-AECL	89-group libraries based on ENDF/B versions V, VI and VII	2-D, with fuel cluster	2-D CP	Advanced self-shielding capabilities with distributed self-shielding effects	Yes	Yes

Table 2 Summary of the worldwide reactor-core codes and their features

Reactor Core Code	Geometry	Solver	Energy groups	Cross-Section Model	Industry code	Applicable to CANDU
SCOPE2	3-D, Cartesian geometry	Nodal method	Multi-group	Macro- and micro-depletion methods, pin-by-pin calculation	No	No
ARTEMIS	3-D, Cartesian geometry	NEM in both polynomial and semi-analytical variants for diffusion theory and SP_3	Multi-group	Micro-depletion method (uses a generalized B-spline approach)	Yes	No
SIMULATE-4	3-D, Cartesian geometry	ANM for diffusion theory and SP_3	Multi-group	Micro-depletion method	Yes	Only for some specific problems
DONJON	3-D, Rectangular and hexagonal geometries	FDM or FEM for diffusion theory and SP_N	Multi-group	Macro-depletion method, History-based method	No	Yes
NESTLE	3-D, Rectangular and hexagonal geometries	FDM and NEM	Two- and four-group	Macro-depletion method, micro-depletion method	No	Only for some specific problems
PARCS	3-D, Rectangular, hexagonal and cylindrical geometries	FDM and nodal method for diffusion theory and SP_3	Two-group	Macro-depletion method (multidimensional cross-section table)	No	Only for some specific problems
PANTHER	3-D, Rectangular and hexagonal geometries	FDM and nodal method	Limited Multi-group (up to 8 groups)	Macro-depletion method	Yes	Only for some specific problems
RFSP	3-D, Cartesian geometry	FDM for the diffusion theory	Two-group	Macro-depletion method, SCM, micro-depletion method, multicell methodology	Yes	Yes

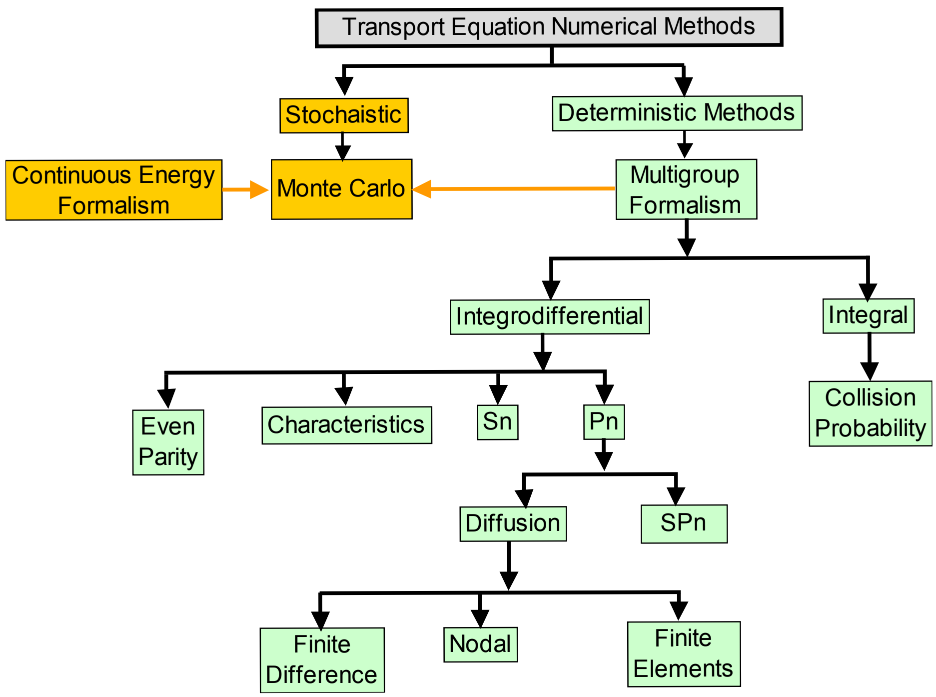

Figure 1 Transport computational methods

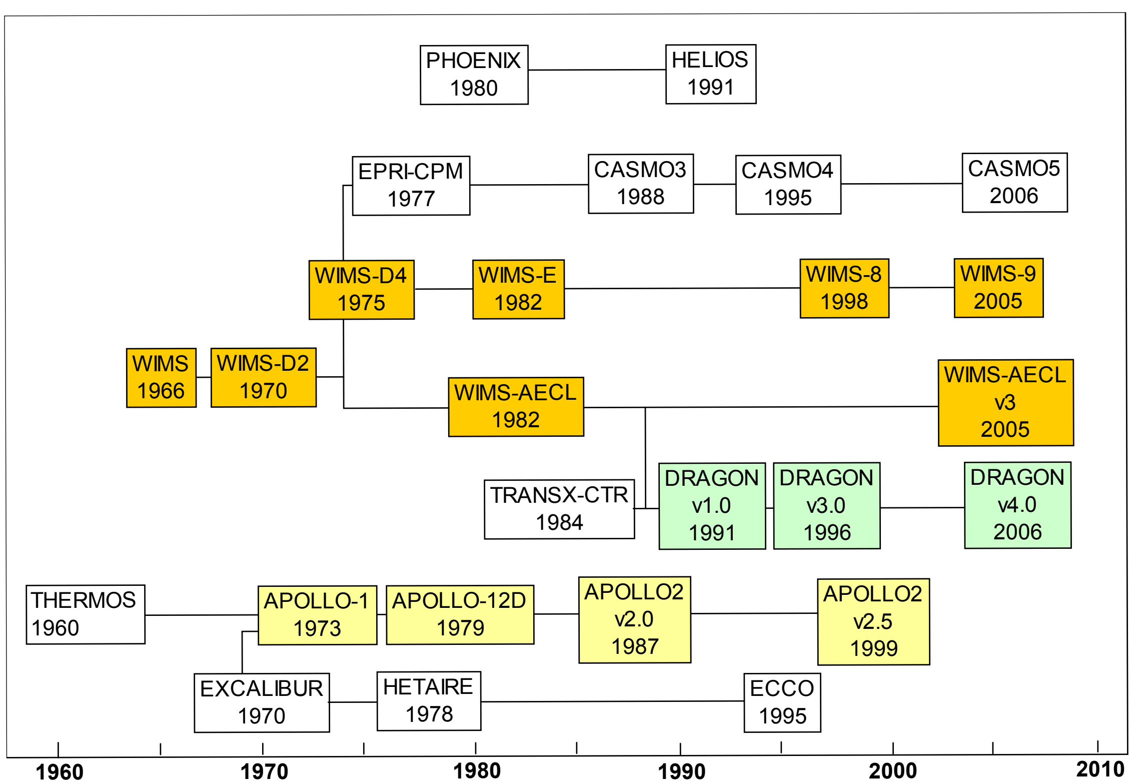

Figure 2 Historical overview of the lattice code development[7]

[7] This figure was taken from the reference [24] with slight updates