EVOLUTION OF RFSP 3.5 FOR CANDU ANALYSIS

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Abstract

From the outset of the development of the CANDU®[1] reactor design, the reactor physics analysis of the core has relied on computer programs developed in Canada and international codes that have been modified and improved. RFSP is the main scientific code for full-core neutronics simulation and analysis of CANDU reactors. Computer codes have evolved to account for the unique characteristics of CANDU reactors. The specific new features, functions, and methodologies that have evolved in RFSP 3.5 series since the last major release of RFSP-IST version 3-04 in 2006 are reviewed in this paper. The new versions of RFSP 3.5 series offer unique capabilities for the design, operate and safety analysis of CANDU-type reactors.

Key Words: CANDU, RFSP, neutron diffusion

1. Introduction

The neutronic analysis of the CANDU reactor core has relied on computer programs developed or advanced in Canada. Deterministic CANDU analysis is a three-step process, and each step is conducted with a dedicated computer code that accounts for the characteristic features of CANDU reactors, such as the use of a cluster of fuel pins in rings surrounded by a relatively large volume of heavy-water moderator, the three-dimensional arrangement of reactivity devices perpendicular to the fuel channels, and the on-power-refuelling feature of CANDU in channels with bi-directional coolant flow and bi-directional refuelling. Significant time and efforts have been invested by the Canadian nuclear industry to develop and maintain the reactor-physics codes that are used for CANDU analysis [1]. As shown in Figure 1, the codes WIMS-AECL [2][3], DRAGON [4][5] and RFSP [6][7] are the three Industry Standard Toolset (IST) codes currently used in the CANDU industry. WIMS-AECL is a two-dimensional (2D) lattice code for neutron-transport calculation and cross-section homogenization and condensation. DRAGON is a three-dimensional (3D) supercell neutron-transport code used for calculation of reactivity devices. RFSP is a 3D finite-difference neutron-diffusion code for static and dynamic core analysis.

The specific physics toolset methodologies that have evolved in RFSP 3.5 series since the last major release of RFSP-IST version 3-04 in 2006 are reviewed in this paper.

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2. RFSP Capabilities

RFSP is a major computer code for the neutronics design and analysis of CANDU reactors. It models the entire reactor core and provides to the design analyst, safety analyst, or the operation physicist crucial system data, such as reactor criticality, reactivity change, 3D flux, power and burnup distributions, fuel exit burnup, reactivity worth of devices, device performance, refuelling ripples, etc. The main function of RFSP is to calculate both the static and kinetic neutron flux and power distributions in the core, using a two-energy-group, neutron-diffusion theory in three spatial dimensions, with homogenized cell-averaged parameters and reactivity-device incremental cross sections, as shown in Figure 2.

RFSP is a very large computer program, with more than 1500 subroutines. RFSP is modular in nature. Each module of the program, identified with an (*), has a specific function. There are for instance modules for model definition (*DATA), for time-average (*TIME-AVER) and instantaneous (*SIMULATE) calculations of the in-core neutron flux and power distributions, and for kinetics (*CERBERUS and *CERBRRS) calculations. The various modules execute independently of one another and share data through the RFSP direct-access file (DAF), which contains all current model information. The main capabilities of RFSP that are built equally into the RFSP modules for time-dependent or time-independent calculations are listed here:

- Snapshot (instantaneous) calculations of the flux and power distributions in the core for given configurations of instantaneous irradiation (or burnup) distributions.
- Core-follow calculations, taking into account specific sequences and timing of channel refueling.
- Snapshot calculations with randomized or patterned "age" (time from last refuelling) distributions of fuel channels.
- Time-average calculation of the CANDU core flux, power, burnup, and refuelling-frequency distributions; the time-average model is an extremely useful design tool.
- Fuel-shuffling and on-line-refuelling calculation.
- Static and kinetic neutron-diffusion calculation for the eigenvalue and fixed-source problems.
- Fast-kinetics calculations with distributed values of the delayed-neutron fractions, based on the distribution of fuel burnup.
- Modelling of the ^{135}I-^{135}Xe kinetics for calculation of the effects of both the steady-state and the time-dependent ^{135}Xe concentration in three dimensions in core, which is extremely important for CANDU reactors on account of their on-power-refuelling feature.
- Coupling with system thermalhydraulic codes for both static and transient analysis.
- Modelling of the actions of the Reactor Regulating System (RRS) in quasi-static and dynamic modes.
- Modelling of detectors.
- Flux-mapping calculations in three dimensions based on in-core-detector readings and pre-calculated flux modes.
- Calculation of the reactivity increase expected on refuelling of individual channels.

3. New Features in RFSP 3.5

RFSP is a legacy code with a long history of development and innovation. In 1974, AECL developed the Fuel-Management-Design Program, FMDP. This program was created to analyze the neutronics of CANDU reactors. Its intended applications were for both design and core-follow, including expected core behaviour for a variety of different fuelling schemes. FMDP was developed to have considerable flexibility, in order to allow the user to model the reactor in varying degrees of detail and to accommodate design changes as they occurred. Over the years, additional program modules were incorporated into FMDP. At the same time, new modules were developed to meet changing requirements. In 1980, a subset of FMDP, specifically designed for use at CANDU generating stations, was developed and became known as RFSP. In 1999 June, the first IST version, RFSP-IST version 3-00, was created by merging AECL's RFSP version 2-17 and Ontario Power Generation's OHRFSP version R1.06 [7].

Since the creation of the first IST version (RFSP-IST version 3-00) of RFSP in 1999, many developments and enhancements have been made in the past decade. Since the last major release of RFSP-IST version 3-04 in 2006, the following versions of RFSP 3.5 series have been created and released, culminating in the latest release of RFSP, version 3.5.3:

- RFSP 3.5 (released in 2008 April)
- RFSP 3.5.1 (released in 2009 April)
- RFSP 3.5.2 (released in 2010 October)
- RFSP 3.5.3 (released in 2011 December, to be an IST version)

The new features developed and implemented in the RFSP 3.5 series since the last major release of RFSP-IST version 3-04 in 2006 are described in this section.

3.1 Fixed neutron-source calculation methods

A phenomenon of particular importance is the power generated by a sub-critical core due to neutron sources not resulting from induced fissions. In CANDU reactors, there are two such neutron sources: one is spontaneous fissions, primarily of U-238 which constitutes most of the fuel; the other is decay of fission products, which give rise to delayed neutrons and photoneutrons.

Neutrons from spontaneous fissions contribute a very small yet constant fraction to the overall reactor power, ranging between 10^{-12} to 10^{-18} of full power (FP), and are thus only important at reactor startup, or after a prolonged shutdown, when no other neutrons are present.

Neutrons from the decay of fission products contribute a larger portion to the overall reactor power, typically 0.5% FP for delayed neutrons and 0.03% FP from photoneutrons. The delayed and photoneutron source strengths depend on the precusor concentrations, which in turn depend on how long the reactor has been operating. These sources of neutrons are important following a shutdown. It should be noted that the precusors for the delayed neutrons have relatively short-half lives and that the effect of the delayed neutrons is only significant immediately following a shutdown. However, the precursor half-lives for the photoneutrons are much longer and the fission products produced from power operation in the weeks before a shutdown will continue to generate photoneutrons for weeks after the shutdown.

To account for these small neutronics effects at deeply-subcritical, low-power conditions, new diffusion-theory reactor physics methods, identified as the fixed neutron-source calculation methods, were recently implemented in RFSP 3.5 series. The neutron-source calculation methods have the capability to model the distribution of neutrons arising from the spontaneous fission of U-238, as well as to model the distribution of delayed neutrons and photoneutrons arising from precursor materials.

The fixed neutron-source calculation methods developed in RSFP have been benchmarked against MCNP full-core results of device worth in the Guaranteed Shutdown State (GSS) for Pickering B and Darlington CANDU reactors. The fixed neutron-source calculation methods developed in RSFP have also been extensively validated against Point Lepreau generating station (PLGS) site measurement data at different deeply-subcritical states such as 30-day shutdown, initial fuel loading, and full-core defueling.

The development of the fixed neutron-source calculation methods in RFSP represents a significant innovation in the modelling of CANDU reactors. The application of the fixed neutron-source calculation methods in modelling subcritical states during GSS in CANDU reactors will assist in an important design improvement. The fixed neutron-source calculation methods could be applied to other sub-critical systems such as Accelerator-Driven Systems (ADS).

3.2 Evolution of the Cross Section Models Used in RFSP

For an accurate calculation of the power distribution in the CANDU reactor, it is important to generate few-group, homogenized cross sections for each fuel bundle in the core, accounting for local conditions (fuel temperature, coolant density, etc.) and for the burnup history of the fuel bundle, as well as for the effects of the environment if necessary. Three approaches were developed over the years for cross-section modeling: these are the uniform-parameter method, the grid-based local-parameter method [8], and the history-based local-parameter method [9]. The cross-section models that can be employed in the commonly-used modules of RFSP are summarized below:

- *TIME-AVER Module (Equilibrium)
    - PPV/WIMS Uniform-Parameter Method
    - PPV Grid-Based Method
    - Multi-cell Correction Method (type-2 only)

- *SIMULATE Module (Static calculation)
    - PPV/WIMS-AECL Uniform-Parameter Method
    - PPV History-Based Local-Parameter Method
    - SCM History-Based Local-Parameter Method
    - Micro-Depletion Method
    - Multi-cell Correction Method

- *CERBERUS and *CERBRRS Modules (Kinetics calculation)
    - PPV History-Based LP Method
    - WIMS-AECL Grid-Based Method
    - SCM History-Based Local-Parameter Method
    - Micro-Depletion Method
    - Multi-cell Correction Method

3.2.1 Simple-Cell Methodology

These three cross-section methods were originally developed for use with the POWDERPUFS-V (PPV) [10] lattice code. With the adoption of WIMS-AECL for lattice calculations, the functionality of the uniform-parameter and local-parameter modelling has been maintained directly. However, the much longer running times of WIMS-AECL put the history-based capability in jeopardy. In response, the simple-cell method (SCM) [11] was developed as a surrogate to reproduce WIMS-AECL results in a shorter computational time. Since SCM relies on specific knowledge of the lattice cell behaviour, so far it has been used for CANDU 6 and Ontario's CANDU reactors with the natural-uranium (NU) fuel only. Large errors may be expected with SCM for fuels that are significant different with the NU fuel.

Significant enhancement on SCM has been made recently in RFSP 3.5 series since the first release of SCM at the end of 1990s. The enhancement includes improved perturbation strategy on the combined perturbation on fuel temperature and coolant density, corrected one-dimensional (1D) annular simple-cell geometry, and consistent meshing treatment in WIMS Utilities and RFSP when solving the 1D multi-group neutron-diffusion equations.

3.2.2 Micro-Depletion Methodology

The micro-depletion (MD) method [12] was developed and implemented in RFSP in order to use the WIMS-AECL properties directly for CANDU-type reactors with any fuel types. The MD method is state-of-the-art and proven technology and it has been used widely in the LWR industry. The MD method implemented in RFSP can explicitly track not only the bundle-average nuclide concentrations but also the nuclide concentrations for different fuel regions in a bundle, which is very important for the modelling of xenon dynamics, even for the CANDU NU fuel, as shown in Table 1.

The MD method developed in RFSP has been successfully verified against WIMS calculations for almost all CANDU-type fuels in the world such as: 37-element NU fuel, Bruce-B LVRF fuel, various ACR (Advanced CANDU Reactor) fuels, Recovered Uranium (RU) fuel, thorium fuel, Canadian SCWR (Super-Critical Water-cooled Reactor) fuel, and Japanese FUGEN fuel, etc. The MD method developed in RFSP has also been validated against measurements from the PLGS for a xenon transient during a power-derating transient [13] and for a 1.5-year core-tracking simulation [14]. The MD method developed in RFSP has been used for the engineering design and safety analysis in many Projects such as the LVRF commercial Project, the ACR-1000 Project, and the EC6 (Enhanced CANDU 6) Project, and the AFCR (Advanced Fuel CANDU Reactor) Project, etc.

Significant enhancement on the MD method has been made recently in RFSP 3.5 series since the first development of the MD method in RFSP in 2004. The enhancement includes improved calculation accuracy for the LOCA (loss-of-coolant-accident) transient after the long shutdown.

3.2.3 Multi-Cell Correction Methodology

The homogenized lattice properties employed in RFSP are usually calculated for a single lattice cell (considered as a heterogeneous medium) with reflective or periodic boundary conditions without considering the effect of the environment due to nearby lattice cells. With the development of the multi-cell capabilities in WIMS-AECL version 3.1 [3], the modelling of lattice properties in RFSP has been enhanced by the addition of a multi-cell correction to the regular single-lattice-cell-based lattice properties. This is especially important in configurations where there are large heterogeneities in the reactor core. The original multi-cell correction method, called as the type-1 multi-cell correction method, was developed in RFSP first to account for the checkerboard-voiding effects in ACR-1000 [15]. The extended multi-cell correction method, called as the type-2 multi-cell correction method, was developed later in RFSP to account for effects at the core-reflector interface in ACR-1000. Compared with other methods developed to account for the core heterogeneity effect, the multi-cell correction method is simple and effective, and has fewer approximations. It allows an efficient treatment of heterogeneity, while maintaining the basic structure of the single-cell-based reactor-physics methodology used for CANDU reactors for many years.

The multi-cell correction method developed in RFSP has been successfully benchmarked against more rigorous continuous-energy-based Monte-Carlo results for a 3D ACR LOCA transient analysis [15], a 3D ACR full-core simulation [16], a 2D CANDU test problem [17], and a 2D SCWR test problem [18]. The multi-cell correction method developed in RFSP has also been validated against measurements from the PLGS for a 1.5-year core-tracking simulation [14]. It has been shown that the multi-cell correction method is essential for the ACR-1000 core [16] and the SCWR core [18] with strong heterogeneities. It has also been shown that use of the multi-cell correction method will also marginally improve the channel-power prediction for the traditional CANDU reactor [14].

3.3 Modelling of multiple reflectors

Compared with the LWR core, the CANDU reactor fuel channels are simply surrounded by heavy-water (moderator) reflector which is usually about 2-lattice-pitch thick. Traditionally the reflector cross sections are approximately calculated using a super-cell model, shown in Figure 3, in which an extra cylindrical region of heavy-water moderator is added outside the regular fuel lattice cell, and a reflective boundary condition is applied at the external surface of the cell. Before enhancement, the reflector cross sections calculated in this way are considered spatial independent; and only one type of reflector can be modelled in RFSP for simplicity.

With the development of the multi-cell capabilities in WIMS-AECL version 3.1, it is possible to generate spatial dependent reflector cross sections more rigorously, from the multi-cell transport calculations. A typical multi-cell model is comprised of an array of five lattice cells with two reflector (R1 and R2) and three fuel lattice cells (F1, F2, and F3), as shown in Figure 4. To accommodate spatial dependent reflector cross sections in RFSP, enhancement has been made in RFSP to model multiple reflector types for the following kinds of calculations:

- Uniform-parameter calculations in the *INSTANTAN, *SIMULATE and *TIME-AVER modules of RFSP; and
- History-based local-parameter calculations in the *SIMULATE, *CERBERUS and *CERBRRS modules of RFSP with both the SCM and the MD method

3.4 Modelling of space- and energy-dependent extrapolation distances

Before enhancement, one difficulty in the validation of RFSP using ZED-2 experiments is that RFSP accepts only one value of extrapolation distance in each radial and axial direction, and in each energy group, respectively. For CANDU reactor, this is acceptable as the reactor is bounded by the same calandria in every direction. However this leads to difficulties for the reactor core with different reflectors in different directions, such as the ZED-2 reactor which has different side reflector and bottom reflector. To accommodate this, enhancement has been made in RFSP to model extrapolation distances that are spatially dependent (both radially and axially) and energy dependent (2-energy groups).

3.5 Enhanced fuel-temperature option

For an accurate calculation of the power distribution and power coefficient in the reactor core, it is important to accurately calculate the fuel temperature. Before enhancement, RFSP calculates the bundle-average fuel temperature by means of the fuel-temperature correlation curves that are based on the CANDU NU fuel. Enhanced fuel-temperature option has been made in the *SIMULATE, *CERBERUS, and *CERBRRS modules of RFSP 3.5 series to improve the accuracy of the power distribution and the power coefficient calculation in the reactor core by accounting for the effect of the fuel temperature for fuel types other than 37-element NU and CANFLEX-NU fuels using a built-in correlation that is a function of bundle power, coolant temperature and burnup. The correlation can also be user-defined with the calculated fuel temperature values saved in the DAF and viewable using the *PRINT module.

3.6 Modelling of aged fuel channels

Before enhancement, two of the impacts of reactor aging cannot be effectively modeled in RFSP: one is the channel sag and the other is the axial bundle displacement. As the degree of channel sag is, in general, assumed to be proportional to the neutron fluence, the amount of displacement will vary throughout the core. Axial elongation of the pressure tubes can result in the position of the fuel bundles within the fuel channel being different for each channel.

Significant effort has been spent to enhance RFSP's capabilities to address these two aging issues. After the enhancement, sagged and axially displaced fuel channels can now be modelled in the *SIMULALE, *TIME-AVER, and *CERBERUS modules of RFSP 3.5 series with both the SCM and the MD method. The effectiveness of the RFSP modelling of aged fuel channels has been benchmarked against MCNP full-core results for a typical CANDU reactor.

3.7 Enhancement in the record structure

RFSP simulations should be as accurate and precise as possible when modeling a real reactor. To achieve higher accuracy and precision may require finer mesh lines in all directions, a larger number of fuel types in the reactor core, and a greater number of transient cases in the kinetics simulations. More data requires more records and indices in the direct access file used by RFSP. RFSP currently allows 40 records or sub-indices to be written under each index of the direct-access file. To allow RFSP to model finer and more detailed reactor models, the maximum number of the records/sub-indices that may be written under each index of the direct-access file needs to be increased. The maximum number of records/sub-indices has been recently increased in RFSP 3.5 series from 40 to 700 on the LINUX and PC Windows platform. Increase in the number of records and sub-indices in the direct-access file gives RFSP capabilities to improve modelling accuracy and precision by allowing:

- a larger number of mesh cells in the z direction;
- a larger number of fuel types in the reactor core;
- a larger number of transient cases to be modelled in kinetics simulations; and
- a larger number of refuelling cases to be modelled in core-tracking simulations

3.8 Rod-based modelling of bulk and spatial control

Rod-based modelling of bulk and spatial control has been developed in the *SIMULATE module of RFSP 3.5 series to model rod insertions (used for ACR-1000) in addition to volume fills (used for CANDU). Fractional insertions of the rods are calculated using existing CANDU 6 equations while additional rod movements can be modelled by means of user-defined rules. The rod-based modelling accounts for rod overlap and incorporates a new radial overpower tilt parameter (ROTP) as defined below:

$$ROTP = \frac{\dfrac{1}{n_{inner}} \sum_{i}^{n_{inner}} overpower_{bundle\ i}}{\dfrac{1}{n_{outer}} \sum_{j}^{n_{outer}} overpower_{bundle\ j}}$$

3.9 Additional and new development works in the RFSP 3.5 series

Besides the code development and enhancement described above, many other new capabilities have also been developed in RFSP 3.5 series. Because of limit of the paper content, only few of these development works are listed here:

- Ability of the *CERBERUS and *CERBRRS modules to conduct depletion calculations for all isotopes including xenon during transient analyses when the SCM or the MD method is selected.
- Ability to conduct SCM and MD calculations at the lattice level by means of a new module called *SINGLECEL.
- Calculation of the fission rates for fissile and fissionable nuclides from SCM and MD fuel tables, the results of which are displayed using the *INTEGRALS module.

RFSP continues to evolve and further development work is being performed such as the development of modelling of a new *GENRRS module to model the Reactor Regulating System (RRS) action for all CANDU reactors (including OPG and Bruce reactors, ACR-1000, and EC6). The *GENRRS module is programmed in a modular fashion using FORTRAN 90/95 so as to be easily maintained and upgraded as required. The development of the *GENRRS module based on RFSP 3.5.3, co-funded by AECL and Candu Energy, is almost complete. The qualification of the *GENRRS module, funded by Candu Energy solo, is still in progress.

4. Conclusion

RFSP has played a central role in the excellent performance of CANDU reactors both in Canada and abroad for over 30 years. Significant time and effort has been invested by the Canadian nuclear industry to develop, enhance, qualification, and maintain RFSP. Physics computer codes have evolved to account for the unique characteristics of CANDU reactors, and design development of new reactor and fuel designs. The specific new features, functions, and methodologies that have evolved in RFSP 3.5 series since the last major release of RFSP-IST version 3-04 in 2006 are reviewed in this paper. The new versions of RFSP 3.5 series offer unique capabilities for the design, operate and safety analysis of CANDU-type reactors.

5. Acknowledgement

A large number of people in many organizations over a considerable period of time have contributed to RFSP and its predecessor FMDP. While unable to thank all of them, the authors would like to acknowledge the major contributors to the creation of the latest RFSP 3.5 series since the last major release of RFSP-IST version 3-04 in 2006 made by: B. Phelps, K. Ho, E. Varin, J. Mao, I. Martchouk, D. Altiparmakov, T. Liang, and D.A. Jenkins. Apologies if there is any omission.

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Table 1 Discrepancies in MD- and SCM-Predicted Lattice Reactivity (mk) for CANDU 6

Discrepancy	Depletion Calculation at Nominal Condition		Max. Diff.	Xenon-Transient Calculation	
	SCM	MD		SCM	MD
Fresh	-0.4	0.0		2.3	0.8
Mid-Bu	-0.8	-0.3			
Exit Bu	-1.1	-0.3			

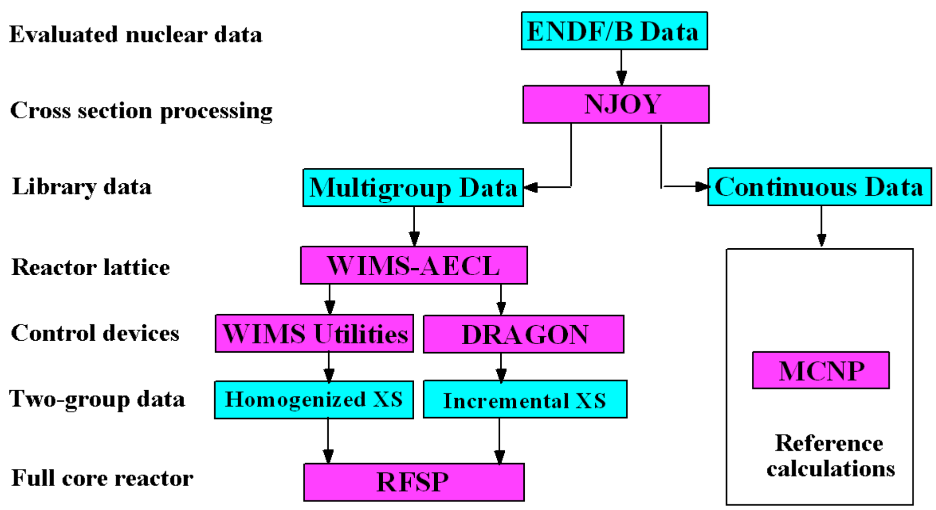

Figure 1 The 3-stage scheme of deterministic neutronics calculations as a part of AECL's reactor physics computational scheme

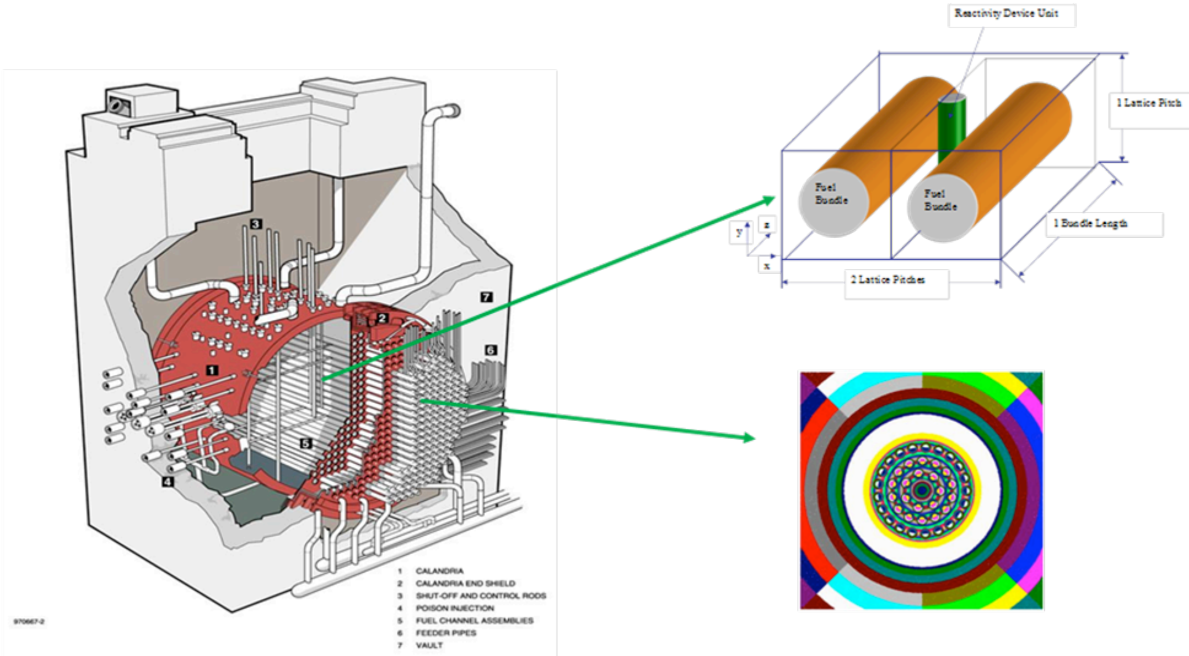

Figure 2 Illustration of RFSP Modelling of a CANDU Core

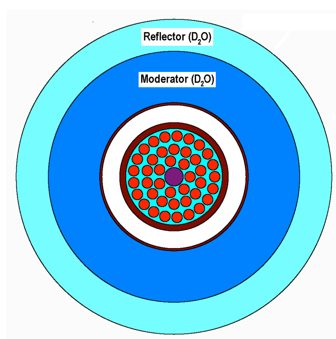

Figure 3 Super-cell model used for the calculation of reflector properties

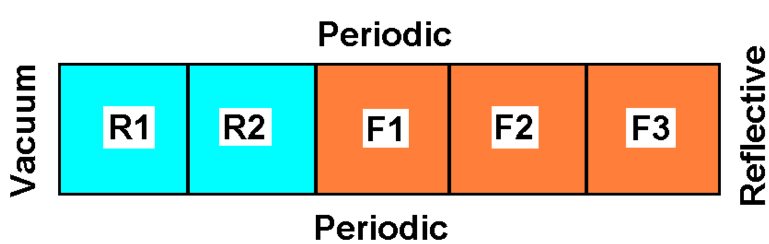

Figure 4 Schematic representation of the planar-type core-reflector interface