Recent Development Status of Neutron Transport Code STREAM

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1. Introduction

Neutron transport code STREAM has been developed at UNIST since 2013. Purposes of the developments are to resolve issues in computational reactor physics field and to develop a high-fidelity LWR core simulation tool. STREAM is used in reactor core analysis through the conventional two-step approach with downstream nodal diffusion code RAST-K [1], and direct whole core calculation which is more attractive approach in reactor physics field. There are a lot of features to be developed and implemented to support both of the two-step and the one-step approaches by STREAM. For high fidelity calculation, it is also needed to upgrade the conventional methodologies used in the old generation lattice physics codes. In this paper, overall methodologies and features of STREAM are summarized. BEAVRS whole core problem [2] has been analyzed by STREAM/RAST-K two-step approach and STREAM one-step approach to demonstrate whole core analysis capability of STREAM.

2. Methodologies and Features

2.1 Library of STREAM

STREAM needs several libraries to analyze light water reactor problems. First of all, STREAM uses 72 energy groups cross section (XS) library over 10^{-5} eV to 20 MeV. NJOY2016 cross section processing code [3] is used to generate multi-group cross section library for wide range of temperature and all nuclides existed in ENDF XS library. STREAM libraries have been generated for ENDF/B-VII.0, ENDF/B-VII.1, and JENDL-4.0. By default, ENDF/B-VII.1 library is used in simulation. In addition to multi-group XS library, resonance integral (RI) table is also generated by using NJOY2016 for wide range of background XSs. This RI table is used when STREAM uses equivalence theory for resonance self-shielding calculation. By default, STREAM uses pin-based pointwise energy slowing-down method (PSM) for fuel resonance self-shielding calculation [4]. For PSM, pointwise energy XS library is also generated by NJOY2016 and it is tabulated to an equi-lethargy energy grid. The intermediate resonance (IR) parameter and resonance up-scattering (RUP) correction factor are generated in-house slowing-down solver and Monte Carlo code MCS. Data used in energy release calculation and decay (*i.e.*, energy release per fission, half-life, branch ratio, and etc.) are extracted from ENDF library by in-house Python scripts.

2.2 Resonance Self-shielding Method and Multi-group Cross-section Generation

STREAM uses PSM to calculate effective multi-group XS of fuel. In PSM, pointwise energy slowing-down equation is solved for every single fuel pin existed in a model. A fuel pin is sub-divided radially and then, collision probability table is made and corrected by the Dancoff factor to consider shadowing effect or effects from neighboring materials. The slowing-down equation is solved and multi-group XS is condensed by using point-wise energy flux. PSM has been verified through a lot of benchmark problems [4]. PSM is the state-of-the-art methodology which can hold both of great accuracy and calculation efficiency. For the other structure material, the equivalence theory is used to get effective multi-group XS. The equivalence theory based on two-term based rational approximation is used to treat resonance of cladding and annular resonance material. For irregular geometries, such as baffle, pad, barrel, and resonance material smeared in coolant are treated by one-term based equivalence theory [6].

Anisotropic scattering effect is important for large leakage problem such as critical experiment and realistic whole core problem, and fuel/reflector color-set problem. There are two ways to consider the anisotropy scattering effect. The first one is the inflow transport correction [7], and the second is a higher order scattering model which is several times slower than the transport correction. STREAM uses the inflow transport corrected P_0 (TCP_0) XS to capture both of accuracy and calculation efficiency.

2.3 Neutron Transport Method

For two-dimensional (2D) calculation, which is used in fuel assembly and reflector group constants generation for a downstream nodal diffusion code, Method of Characteristics (MOC) method is used to solve neutron transport equation. For three-dimensional (3D), a new 3D neutron transport method has been developed and implemented in STREAM. The method is named 3D Method of Characteristics/Diamond-difference (3D MOD/DD). In the method, 3D flux and source are constructed with combination of 2D radial component and 1D axial component. Differently with the conventional 2D/1D method, the 3D MOC/DD

method does not use axial solver. In addition STREAM does not homogenize the axial source into a square pin mesh to represent more realistic axial neutron streaming effect. The axial neutron source occupies a part of source in the MOC solver. The axial source depends on every single source region and MOC angles. The 3D MOC/DD method requires more computational burden than that of the conventional 2D/1D method because a few centimeters axial source mesh is used. However, the 3D MOC/DD method can be used practically in whole core analysis, and the method does not have stability issues which are occurred with the conventional 2D/1D method for coolant void problem and a problem with a very thin plane.

2.4 Depletion Calculation Method and Energy Release Model

STREAM uses a depletion chain composed of about 1,600 nuclides. The chain among nuclides is connected by realistic neutron-nuclide reaction types and decay modes. The semi-predictor/corrector method is used to reduce error occurred by time discretization. For problem with gadolina fuel, quadratic depletion feature is turned on to capture dramatic change of reaction rate of gadolinia material [8]. Batman equation is solved by Chebyshev rational approximation method (CRAM) [9]. The linear system constructed by CRAM is solved by sparse Gauss-Seidel solver.

For direct energy release calculation, STREAM adopts the direct energy release model used in CASMO-5 [10]. The direct energy release model can eliminate a poor approximation for energy released by (n,γ) reaction. The information for energy released from coolant is used in multi-physics coupling calculation.

2.5 Few-group Constant Generation

STREAM has a capability to generate assembly and pin-by-pin few-group constants. For conventional two-step approach, two-group constants are generated by homogenizing 1/4 assembly. In case of asymmetric fuel assembly such as assemblies in the BEAVRS benchmark, full fuel assembly is solved, and then four different group constant files are generated. The critical spectrum calculation is performed to consider the neutron spectrum of assembly in whole core [11]. The assembly and reflector discontinuity factors are generated to prevent loss of accuracy due to diffusion approximation in downstream nodal code. The corner discontinuity factor is also generated to perform pin power reconstruction in the nodal diffusion code. The microscopic and macroscopic XSs are edited at a lot of branch and history to consider wide range of condition of fuel assembly in a reactor core [1]. The microscopic XS is used in nuclide-wise depletion in RAST-K. The pin-by-pin XS editing is also possible for pin-by-pin SP_3 solver which is adopted in RAST-K version 3.

2.6 Multi-physics Calculation Solver

For direct whole-core transport analysis, STREAM uses built-in simple multi-physics solver to calculate temperature and density instead coupling with external sub-channel code and fuel performance code. By using built-in multi-physics solver, it is possible to easily organize mesh structure, which is used in thermal-hydraulic calculation and fuel temperature calculation, based on neutron transport solver. Fuel conductivity is function of burnup, temperature and gadolinia content so that it is possible to consider realistic condition encountered in reactor core. Fuel-cladding gap conductance model is being under developing to consider fission gas in gap and complicated fuel behavior such as swelling, relocation, thermal expansion, and densification. The multi-physics solver considers ZrO_2 layer thickness to consider temperature drop across coolant and cladding outer surface.

2.7. Feedback Calculation and Thermal Expansion

The critical boron concentration (CBC) search and equilibrium Xenon search functions have been implemented for direct whole-core transport calculation. For CBC search, it is needed to calculate derivative of boron concentration to k_{eff}. Historically, this derivative is calculated during power iteration. However, the conventional method sometimes cause oscillation and slow convergence rate. To prevent this phenomenon, STREAM calculates the derivative by using neutron balance equation. When temperature is updated by the multi-physics solver, STREAM changes geometry and material density to consider the thermal expansion effect. When the temperature is changed, MOC ray is re-generated to consider the change of geometry. STREAM has a capability to model different pin-pitch among assemblies because the pin-pitch is adjusted depending on coolant temperature of each single assembly.

STREAM use an implicit fixed point iteration to couple the various feedback effects. When CBC and equilibrium Xenon search, and temperature feedback from multi-physics solver are used simultaneously, it is important to control each feedback effect because the feedback effect is calculated before neutronics solution is not converged yet. Therefore, an empirical relaxation method is used to prevent oscillation and unstable convergence behavior from various feedbacks.

2.8 Input / Fuel Shuffling for Whole Core Modeling

It is possible to easily model whole core problem by using STREAM. The number of lines to model a whole core problem is less than 700 lines depending on core configuration such as Fig. 1. STREAM has special input cards to easily model baffle, barrel, pad, and vessel. For multi-cycle simulation, material reloading and shuffling functions have been implemented.

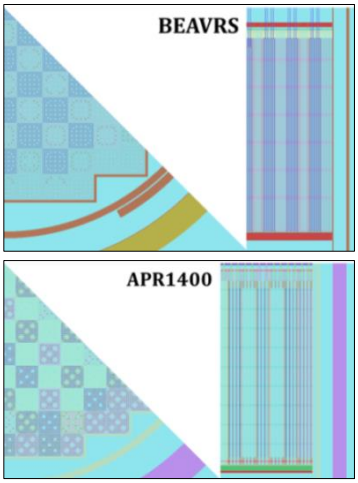

Fig. 1. BEAVRS [2] (Top) and APR1400 core [12] (Bottom) modeled by STREAM.

3. Numerical Results

BEAVRS core cycle 1 and cycle 2 problems [2] have been solved by STREAM and STREAM/RAST-K to demonstrate whole core multi-cycle simulation capabilities. The latest version 3.0.1 of BEAVRS benchmark [13] is used to model the core. For cycle 1 and 2 simulations, full power operation and all rods out condition are assumed because power level and rod position are changed frequently as time goes on. The cycle 1 and cycle 2 CBC results are shown in Fig. 2 and Fig. 3, respectively. The CBCs from calculations are interpolated to measurement points. At the first cycle, the STREAM and STREAM/RAST-K show good agreement with less than 20 ppm difference on average. Most of CBCs are within 50 ppm measurement uncertainty. CBCs are under-estimated. The unrealistic power history simulation may cause the bias.

Table I: BEAVRS core cycle 1 calculation parameters and run-time by STREAM one-step approach.

Parameter		Value
Ray spacing (cm)		0.05
# of azimuthal angles (2π)		48
# of polar angles (π)		6
Scattering source model		Inflow TCP_0
Core symmetry		Octant
# of source regions		88,820,139
# of axial source planes		181
# of axial material planes	All	64
	Active region	53
# of radial rings in fuel pellet		3
# of depletion steps		16
# of computing cores		168
Run-time (Days)		1.9
Run-time (Hours)		46.1
Run-time / step (Hours)		2.9
Run-time (Core-days)		322.7
Total memory (TB)		1.137

Table I shows calculation parameters and run-time by STREAM direct whole core simulation. Even though 181 planes are used in the simulation, STREAM shows a possibility to be applied to practical whole core simulation. It takes 322.7 core-days to simulate BEAVRS cycle 1 with octant core modeling.

For cycle 2, STREAM and STREAM/RAST-K shows good agreement with measurement data with less than 20 ppm error. The maximum difference between STREAM and STREAM and STREAM/RAST-K is 8 ppm. Compared to cycle 1 results, STREAM and STREAM/RAST-K show better agreement. For cycle 2, about 800 core-days is required to complete cycle depletion because of a rotational quarter core modeling.

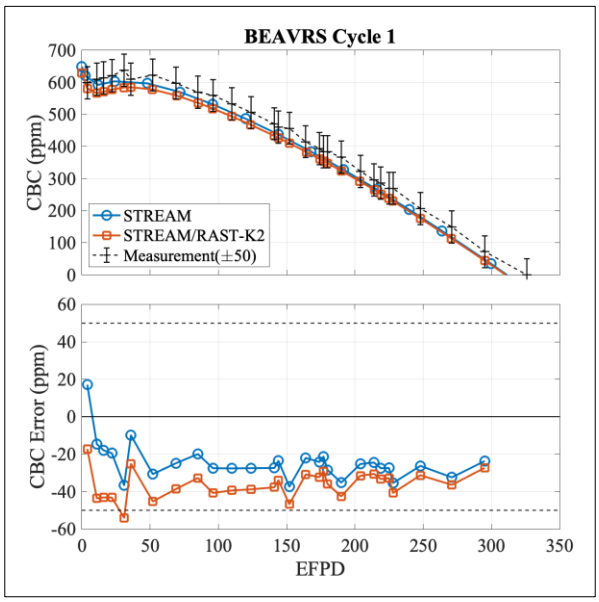

Fig. 2. BEAVRS cycle 1 CBC results by STREAM and STREAM/RAST-K.

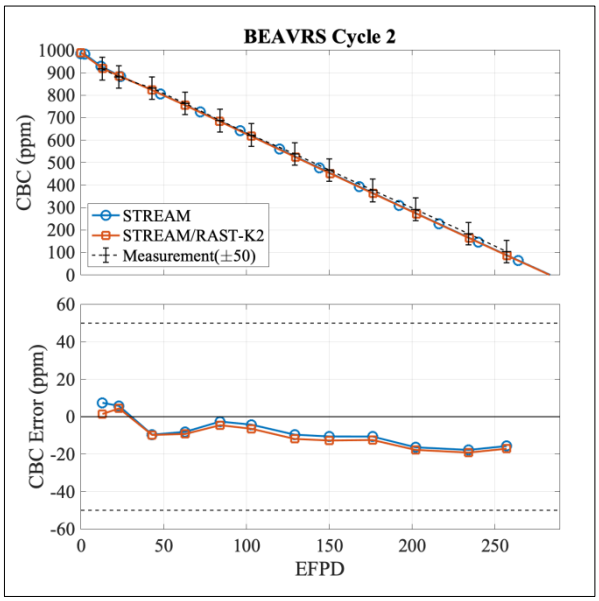

Fig. 3. BEAVRS cycle 2 CBC results by STREAM and STREAM/RAST-K.

3. Conclusions

The recent status of development of STREAM has been summarized in this paper. A lot of advanced and state-of-the-art methodologies and features have been implemented for high fidelity calculation in both of one-step and two-step approaches. The BEAVRS benchmark cycle 1 and cycle 2 cores have been solved to demonstrate capability of whole core simulation. STREAM and STREAM/RAST-K show good agreement with measurement data. In the future, a lot of reactor cores will be analyzed to validate and improve STREAM and STREAM/RAST-K.

ACKNOWLEDGMENTS

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