AFENX: A REACTOR CORE ANALYSIS CODE BASED ON ANALYTIC FUNCTION EXPANSION NODAL METHOD

Nam Zin Cho and Do Sam Kim

Korea Advanced Institute of Science and Technology
Department of Nuclear and Quantum Engineering
373-1 Kusong-dong, Yusong-gu
Taejon, Korea 305-701
Email : nzcho@mail.kaist.ac.kr ; kds@nurapt.kaist.ac.kr

1. Program Name and Title: AFENX: A Reactor Core Analysis Code Based on Analytic Function Expansion Nodal Method.

2. Computers and Operating Systems for Which Program is Available: UNIX-based workstations and Windows-based personal computers

3. Problems Solved: The AFENX code solves static and time-dependent multigroup neutron diffusion equations based on the analytic function expansion nodal (AFEN) method[1-6]. It can treat the multigroup problems with albedo boundary conditions in three-dimensional rectangular, hexagonal and cylindrical R-Z geometries. It can be used for full scope reactor core design and analysis of commercial light water reactors (LWRs) of rectangular geometry and VVER reactors, high conversion reactors, and fast neutron reactors of hexagonal geometry. Also, the pebble bed modular reactor (PBMR) in R-Z geometry can be analyzed with triangular nodes in the lower periphery of the reactor core. AFENX has a core depletion module which requires the steady state calculation with thermal hydraulic feedback for fuel management studies. Adjoint solutions[7] are also provided for steady state problems.

A. Static Calculation

Static calculations are used to study reactor behavior in normal operating conditions and carry out fuel management studies. The standard k_{eff} calculation[1-3] in AFENX is based on the standard source iteration. In order to establish the initial state, AFENX performs the critical boron concentration search or critical control rod search calculation.

B. Transient (Kinetics) Calculation

One of the primary functions of AFENX is the solution of the time-dependent neutron diffusion equation involving both delayed and prompt neutrons. The transient module treats the intranodal cross section distribution (e.g., due to temperature feedback, burnup, or control rod motion) explicitly without rehomogenization. The precursor fractions and decay constants of the groups can be specified by user at each node. This makes it easy to analyze the partially MOX loaded core.

C. Adjoint Calculation

The adjoint flux is widely used for estimating the effects of changes in reactor systems for a range of reactor characteristics such as reaction ratios and fuel burnup. In AFENX, the mathematical adjoint solution is found by solving the transposed matrix of the nodal equation.

D. Depletion

AFENX performs core depletion through steady state calculation with thermal hydraulic feedback to carry out fuel management studies. The data for the depletion calculation is based on the table set generated from the lattice physics codes such as HELIOS depending on the burnup steps and operating conditions. The moderator and fuel temperatures are obtained from steady state thermal-hydraulic equations or empirical formula. The number densities of Xe/Sm are updated by solving the respective balance equations using fluxes resulting from the neutronics calculation.

E. Pin Power Calculation

In order to obtain local pin power distributions from the nodal solution, it is necessary to reconstruct pin powers. This is performed by multiplying the hegerogeneous power form functions with the homogeneous intranodal flux distribution which is directly obtained from the steady state or transient AFENX calculation.

4. Method of Solution: The AFEN method represents intranodal flux distribution in terms of multi-dimensional analytic basis functions. Since the AFEN method does not use the transverse integration, it does not suffer from the drawbacks of the usual nodal methods

caused by the transverse integration. Tight constraints and the use of analytic basis functions result in high accuracy. Especially, it is known to be very effective for problems with strongly varying flux distributions. Accuracy can be further increased by addition of the transverse gradient basis functions[3]. Also, the method is robust to problems in rectangular, hexagonal and cylindrical geometries[1,2]. The multi-group extension[4] is made by using a matrix transformation method for complex eigenmodes. The coarse group rebalance (CGR) scheme[5] is applied for acceleration of the code with partial current formulation. In kinetics calculations, the solution decomposition method[6] is used. It can also treat the space-dependent feedback induced intranodal cross-section distributions explicitly without rehomogenization. The numerical singularity that occurs in any analytic nodal methods when the core contains nearly no-net-leakage nodes is removed by applying the continued factoring method described in Ref. 3.

5. Restrictions on the Complexity of the Problem: The maximum number of energy groups and mesh points are limited only by the physical memory of the platform machine.

6. Typica1 Running Time: The following tables give typical CPU times for AFENX static and transient computations.

A. Case 1 – Static Calculation

IAEA-3D problem, three-dimensional, two energy groups, quarter core, 1 nodes per fuel assembly, 1539 nodes (81 radial nodes, 19 axial nodes), 10^{-5} convergence criterion on flux, coarse group rebalance (CGR) acceleration, HP C180 workstation.

	Time (seconds)
Formulation 1[a]	1.77
Formulation 2[b]	1.42

[a] Nodal unknowns in each group: 1 node average flux, 6 interface average fluxes and 12 edge fluxes

[b] Nodal unknowns in each group: 1 node average flux, 6 interface average fluxes and 12 interface flux moments

B. Case 2 - Transient Calculation

NEACRP A1 and A2 Benchmark[13], three-dimensional, two energy groups, 1280 nodes (64 radial nodes, 20 axial nodes), 10^{-4} convergence criterion on flux, coarse group rebalance (CGR) acceleration, time interval : 0.0 ~ 5.0 sec, Alpha workstation (Zion-1000).

Case	Δt (seconds)	Number of time steps	Time (seconds)
A1	0.004368	1140	709.8
	Variable	465	615
A2	0.01	500	125.9
	Variable	32	33.8

7. Unusual Features of the Program: All functional modules of steady-state, depletion, transient, and T/H are seamlessly integrated in the AFENX code so that no additional effort is required for interfacing or transferring data between them. It also has versatile applicability to various types of reactor core design.

8. Hardware Requirements: The amount of memory required is problem dependent. In practice, 200 Mbyte RAM on PC is enough for the two-group, three-dimension, and full core analysis of typical PWRs.

9. Programming Language(s): ANSI C.

10. Other Programming or Operating Information or Restrictions: None.

12. References:

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