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Status of Reactor Analysis Methods and Codes in the U.S.A

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The reactor physics methodologies and codes that are used in the U.S.A have been reviewed and summarized in this paper. The status of representative neutronics analysis capabilities and ongoing development activities are presented. This review covers cross-section generation capabilities for thermal and fast systems, whole-core deterministic (diffusion and transport) and Monte Carlo calculation tools, and depletion calculation methods and codes.

The review indicates that the existing neutronic analysis tools are sufficiently accurate for the design of current power reactors and for early pre-conceptual design development and viability phase evaluations of advanced reactor designs. For refined analyses of the advanced systems, however, they require additional verification and validation tests. Additional improvements and capabilities might be needed in order to reduce computational uncertainties and improve the plant operational economics.

KEYWORDS: physics methods, cross section generation, lattice codes, diffusion theory, transport theory, Monte Carlo, depletion analysis, thermal reactor, fast reactor

1. Introduction

Continued efforts to improve reactor analysis methods are being made in industry, national laboratories, and universities. The industrial efforts have been motivated by the need to reduce the computational uncertainties and associated margins, and thereby improve the plant operational economics. The renewed interests in methods development at national laboratories and universities are boosted by the current national interests in advanced nuclear systems. In one such effort under the Generation IV program, six advanced nuclear energy systems (VHTR, GFR, LFR, SCWR, SFR, and MSR) have been selected for development. These systems are being investigated worldwide to achieve the goals of effective resource utilization and waste minimization (sustainability), improved safety, enhanced proliferation resistance, and reduced system cost. The design of these Generation IV systems will rely extensively on simulation capabilities to provide accurate predictions of system performance. Various activities are therefore ongoing to develop advanced methods and codes that can be applied to the design of these advanced systems.

The objective of this paper is to review the status of the neutronics computing capabilities and ongoing development activities in the U.S.A. This review is focused on the neutronics design methods and codes, and covers the cross section generation, whole-core calculation (deterministic and Monte Carlo), and depletion calculation capabilities being developed in the last decade or so. Other important topics such as nuclear data, perturbation and sensitivity analysis, and shielding are not covered.

2. Cross Section Generation Capabilities

2.1 Processing of Evaluated Nuclear Data

An initial step in the use of analytical models for reactor core calculations is the processing of evaluated nuclear data files for all pertinent nuclides into suitable forms for use in application codes (deterministic and Monte Carlo codes). The NJOY code is an internationally accepted code for processing nuclear data in the ENDF format into libraries for most application codes. [1] (The ENDF format has become the standard for representing nuclear data worldwide, being utilized in the ENDF/B, JEF, and JENDL libraries) NJOY was developed at LANL and is periodically upgraded. Recently added features include capabilities for high-energy libraries (up to 150 MeV), options for detailed treatment of charged particles, probability tables for unresolved range self-shielding treatment, and the capability to handle photonuclear reactions.

The accelerator-driven systems (ADSs) that have been proposed for the transmutation of waste utilize high-energy neutrons produced by the spallation process to drive a subcritical nuclear pile (blanket). While these high-energy neutrons can be accurately tracked and transported at energies greater than 150 MeV using intranuclear cascade models, accurate modeling at lower energies requires that nuclear structure effects are adequately treated using evaluated cross sections. To generate the cross section data in the 20-150 MeV, new methods have been implemented in the NJOY code to handle the wide range of options allowed by ENDF File 6, including using systematics to describe angle-energy distributions and other new representations.

In addition to neutrons, the representation of charged particles (protons) is also important in the modeling of ADSs because these particles are either used for the spallation process or produced by high-energy nuclear interactions. This has necessitated the development of evaluated data for incident charged particles and particle production in the 20-150 MeV range and algorithms for processing the data for use in application codes. NJOY methods have been developed to handle all pertinent aspects of charged-particle data. The code also contains capabilities for treating charged particle effects of importance to low-energy fusion calculations and astrophysics.

Probability tables are used in continuous-energy Monte Carlo codes for handling the effects of self-shielding in the unresolved resonance energy range. The effects of unresolved self-shielding are expected to be of importance in fast-reactor core modeling. The current version of the NJOY code (NJOY99) generates these tables using the PURR module. Other new features of NJOY include capabilities to produce ACE libraries containing delayed neutron data and to process photo-neutron data for use in application codes.

2.2 Lattice Codes for Thermal Reactor Analysis

The lattice calculation capabilities, which are employed for generating multigroup cross sections for whole-core calculations, are generally specific to design systems (thermal versus fast). For LWR analysis, there are a number of well-established 2-D transport lattice codes. The production codes widely used in the U.S. include CASMO-4, PHOENIX-P, TGBLA, HELIOS, WIMS8, and DRAGON. [2-7] Several of these lattice codes have recently been equipped with the method of characteristics for the transport solution, and been applied to 2-D whole-core transport calculations. There are also ongoing activities to develop new lattice codes such as PARAGON and NEWT. [6,8] Special interests have arisen in codes for treating the double

heterogeneity effects for coated particle fuel systems. The WIMS8 and DRAGON codes have models that can be adapted for this purpose, but additional improvements might be needed.

The CASMO-4 code is a multigroup transport theory capability used internationally for the static and depletion analysis of LWR lattices. [2] The code performs detailed heterogeneous calculations using the methods of characteristics for the solution of the two-dimensional (2-D) Boltzmann transport equation. Enriched uranium and mixed oxide fuels are modeled with the code. Burnable poisons such as Gd, B_4C, Pyrex, WABA, IFBA, and Erbia can also be treated. CASMO-4 can also be used for the analysis of fuel-storage-rack geometries. The extended CASMO-4E version of the code permits the application of JEF2.2 and ENDF/B-VI based libraries for lattice calculations. Over 300 nuclides are contained in the libraries, including 45 heavy nuclides and 200 fission products. Extended depletion chains for higher actinides and thorium are available in the code. Geometries for analyzing BWRs, PWRs, VVERs, advanced gas-cooled reactors, and other reactors are supported by the code. An azimuthal pin depletion model is also available in the code.

The DRAGON code has a collection of models for simulating the neutronic behavior of a unit cell or a fuel lattice in a nuclear reactor. [3] The typical functionalities found in most modern lattice codes are contained in DRAGON. These include interpolation of microscopic cross sections supplied by means of standard libraries; resonance self-shielding calculations in multidimensional geometries; multigroup and multidimensional neutron flux calculations which can take into account neutron leakage; transport-transport or transport-diffusion equivalence calculations; and modules for editing condensed and homogenized nuclear properties for reactor calculations. The code also performs isotopic depletion calculations. The code user must however supply cross sections in one of the following standard formats: DRAGON, MATXS (TRANSX-CTR), WIMSD4, WIMS-AECL, and APOLLO. Macroscopic cross sections can also be read by DRAGON via the input data stream. The current version of the code contains three algorithms for the solution of the integral transport equation, ranging from a simple collision probability method coupled with the interface current method to the full collision probability method. An attractive feature of the DRAGON code is the ability to treat particulate fuel in a matrix. This capability has been used for modeling the fuel elements of block-type, high-temperature gas-cooled thermal reactors and the pebble elements in alternative pebble-bed concepts.

The WIMS8 code provides an extensive software package for neutronics calculations. [4] The code employs an open structure that permits the linking of various methods to create a calculational scheme for a given thermal reactor design. These could range from simple homogeneous cells to complex whole-core calculations. Most generally, however, the lattice capabilities of the code are used for reactor analysis. Geometries are available for analyzing PWR, BWR, VVER, AGR, RBMK, CANDU, other reactor core designs, storage pools, and experiments. Methods for the neutron flux solution include collision probability (1-D or 2-D), method of characteristics, S_n method (1-D or 2-D), diffusion theory, and hybrid methods. The code also provides an integrated Monte Carlo method (MONK) for the purpose of internal validation. WIMS8 is supplied with 69- and 172-group libraries based on the validated JEF2.2 nuclear data. It is noted that the WIMS8 code has the WPROCOL module that provides a capability for calculating the collision probabilities of particulate fuel in an annular geometry that could be used in flux solvers to model the double heterogeneity effect of that fuel form.

HELIOS is a 2-D general geometry lattice-physics code for neutron and gamma transport calculations in fuel assemblies and larger heterogeneous systems. [5] The code solves the

transport equation using the current-coupling collision probability method in the energy groups of the cross section library. The HELIOS cross section library is based on ENDF/B-VI nuclear data. The standard application library has 34 neutron and 18 gamma groups. A larger 190-group neutron library is also available with the code.

The APA (ALPHA/PARAGON/ANC) code system being developed by Westinghouse provides an example of an emerging lattice capability for core design calculations. [6] The code system is based on the newly developed lattice code PARAGON and the well established advanced nodal code (ANC). PARAGON is intended to replace the PHOENIX-P code that is currently used in generating the assembly nuclear data for ANC. The ALPHA code is the automated driver of both PARAGON and ANC that aids the user in modeling all types of PWR cores. Similar to other lattice codes, the PARAGON code has primarily four basic modules: resonance self-shielding, flux solver, leakage correction, and depletion. Compared to PHOENIX-P, the PARAGON code provides more accurate angular treatment and geometry modeling flexibilities, including exact cell geometry representation instead of cylinderization, multiple rings and regions within the fuel pin and the moderator cell geometry, and variable cell pitch. [6] With PARAGON, whole-core modeling, two-dimensional baffle/reflector modeling, and rack type calculations for spent fuel pool are now possible. PARAGON is based on the collision probability and interface current coupling methods, and it uses a new resonance self-shielding method SDDM (Space Dependent Dancoff Method) for the treatment of the resonance cross-sections. The code library uses ENDF/B-VI as the source of the basic evaluated data, and it is generated using the NJOY processing code. Although it can work with any number of energy groups specified in the library, the current library has 70 neutron energy groups (and 48 gamma energy groups). All important fission products are explicitly represented in the library.

TGBLA is the lattice code that is generally used by General Electric for the analysis of the BWR assembly. [7] The code performs the thermal neutron spectra calculation by a leakage-dependent integral transport method. An approximate 1-D geometry treatment is used for the resonance integral calculation of the resonant nuclides. The fluxes from the transport theory problem are used to condense group parameters into few groups. A 3-group, 2-D fine-mesh, diffusion theory model is used for determining the lattice multiplication factor, rod-wise fission densities and gamma-smeared power densities, neutron balance, 3-group lattice homogenized cross sections, and flux discontinuity factors. Depletion calculations are performed using a single pseudo-fission product lump and single gadolinium tail lump. [7] The version of the code approved by the USNRC in 1999 was qualified over an expanded range of enrichments and gadolinium poison concentrations. It is noted that different vendor and utilities have developed CASMO-4 models for analysis of BWR cores and that the code has been used as an independent methodology for the verification of the TGBLA approach. A new lattice methodology being developed at Global Nuclear Fuel (GNF) is LANCER. The LANCER code employs a collision probability method for solving the 2-D transport equation with over 30 energy groups.

2.3 Cross Section Generation for Fast Reactor Analysis

The main approach used for fast reactor cross section generation in the 1970s was the Bondarenko self-shielding factor method. [9] In this approach, a generalized cross section set is first prepared by calculating multigroup cross sections for a given material as a function of background cross section and temperature. Then, cross sections for a particular composition at a given temperature are interpolated from the cross sections in the generalized set by calculating the background cross section for each material. Typical codes employing this approach are

MINX/SPHINX and NJOY. [10-12] An alternative approach was developed at ANL based on detailed spectrum calculations for individual compositions, and implemented in the ETOE-2/ MC^2-2/SDX code system. [13-15] Compared to the self-shielding factor method, this approach is more rigorous in energy treatment. Recently, the self-shielding factor method has been improved to allow more detailed energy modeling by utilizing more energy groups. [8] Recent international improvements in this approach include the implementation of multi-dimensional lattice analysis capability in conjunction with sub-group methods (e.g., the ECCO code). [16] This spatial modeling capability explicitly treats the heterogeneity effect, and the sub-group method allows some re-capture of the resonance energy details.

At present, the second approach based on detailed spectrum calculation is most widely used in the U.S. In this approach, the ETOE-2 code is used for processing the ENDF/B data files into cross section libraries utilized by the MC^2-2 and RABANL codes. [14] These libraries include resolved resonance parameters, unresolved resonance parameters, ultra-fine-group smooth cross sections (2082 groups), inelastic and (n,2n) scattering data, fission spectrum parameters, and elastic scattering distributions. The ETOE-2 code was recently updated to process the ENDF/B-VI format by implementing a scheme that converts the Reich-Moore parameters into multi-pole parameters such that they can be Doppler broadened. [17,18]

Using the libraries generated by ETOE-2, ultra-fine group MC^2-2 calculations are performed for specific compositions and temperatures with explicit representation of resonances. The MC^2-2 code solves the neutron slowing-down equation with the P1, B1, consistent P1, or consistent B1 approximation. The multi-group slowing-down equation based on ultra-fine group lethargy structure is solved above the resolved resonance energy, but the continuous slowing down equation is solved below this range. Resolved and unresolved resonances are treated explicitly by the generalized J^* integral formulation based on the narrow resonance approximation including overlapping and Doppler broadening effects. Equivalence theory is used to treat heterogeneity effect. For the resolved resonance range, alternative hyper-fine group integral transport calculation is an option. Other available options include inhomogeneous group-dependent sources, group-dependent buckling, isotope-dependent fission spectrum distributions, and buckling search for criticality.

MC^2-2 calculations provide composition and temperature dependent cross sections in a user specified energy group structure. Broad group cross sections for whole-core calculations can be obtained directly from these MC^2-2 calculations. However, since few group cross sections are generally space-dependent, they are typically determined in multiple steps. In the first, composition and temperature dependent cross sections are generated in an intermediate (~230) group structure from MC^2-2 calculations. Using these intermediate group cross sections, whole-core diffusion or transport calculations are then performed with relatively simple core models. Finally, using the resulting space dependent flux as weighting spectra, the intermediate group cross sections are collapsed into space-dependent broad group cross sections. For the space-dependent flux calculation, 1-D diffusion calculations used to be done with the SDX code. (The SDX code has options for the resonance integral calculation of actinide isotopes and the collision probability unit cell calculation to account for heterogeneity effect.) At present, 2-D transport calculations in RZ geometry are usually performed with the TWODANT code. [19]

3. Whole-Core Calculation Tools

3.1 Advanced Nodal Methods

The whole-core analysis capabilities are not specific to design systems, and can be used for a wide spectrum of systems. There are a number of well-established deterministic tools for diffusion and transport calculations. Most of the current whole-core diffusion theory codes are based on the advanced nodal methods that were developed mainly in early 1980s to replace the expensive pin-by-pin finite difference method. These nodal codes include ANC, DIF3D, NESTLE, PARCS, and SIMULATE-3. [20-24] Using the homogenized assembly parameters obtained from assembly lattice calculations, they solve the few-group diffusion equation for three-dimensional Cartesian and/or hexagonal geometries.

Among dozens of different nodal methods, those based on the transverse integration procedure are most widely used in production analysis. In these methods, the transverse leakages are typically approximated with quadratic polynomials, and the resulting set of one-dimensional equations is generally solved with the nodal expansion method or the analytic nodal methods. However, the final algebraic equations are formulated in various forms and solved with different solution and acceleration schemes including Krylov subspace, domain decomposition, coarse-mesh rebalancing, and non-linear iterative methods.

These advanced nodal methods usually employ discontinuity factors to reduce assembly homogenization errors. Pin powers are recovered by imbedded local calculations or by superposition of nodal and lattice powers. The nodal codes developed for LWR applications treat intra-assembly depletion effects with space-dependent homogenized cross sections and the spectral interactions between assemblies using two-group spectrum as a measure of deviation from the lattice calculations. Various refinements were recently made in these nodal codes for applications to MOX fueled core and extended burnup analyses. A typical accuracy of current advanced nodal method is presented in Table 1. [25] It should be noted that this represents the aggregate accuracy of nuclear data, core representation, and lattice and full-core calculation methods.

Applications of the advanced nodal method were significantly extended in the last decade. Due to the high efficiency of nodal method and advances in computer hardware, the whole-core analysis can now be performed in a fraction of a minute. This has led to the direct use of advanced nodal codes in fuel loading optimization programs and core monitoring systems.

Table 1. Typical Accuracy of Advanced Nodal Method

	PWR	BWR
Operating reactors		
Axially integrated reaction rates	~1.0% rms[a)]	~1.5% rms
3-D reaction rates	~3.0% rms	~3.0-6.0% rms
Pin powers of BOL criticals		
Axially integrated pin powers	~1.0% rms	
2D whore core lattice depletion calculation		
Assembly powers	~1.0% rms	
Pin powers	~1.5% max[b)]	
MOX pin powers	~2.5% max	

[a)] root-mean-square error

[b)] maximum error

Various three-dimensional, spatial kinetics capabilities were also developed based on the advanced nodal methods, and coupled with system thermal-hydraulics codes. These high fidelity kinetics methods are important for core transients involving significant variations of the flux shape.

3.2 Deterministic Transport Codes

In the last decade, whole-core transport theory capabilities were significantly improved in both the first-and second-order formulations of the transport equation. The production codes based on the first-order formulation include ATTILA, PARTISN, PENTRAN, THREEDANT, and TORT. [26-30] These codes solve the multi-group transport equation on two- and three-dimensional orthogonal or unstructured meshes. All of these codes use the discrete ordinates approximation for treating the angular variation of particle distribution. Various spatial discretization schemes are employed including the diamond and weighted diamond difference, linear and exponential discontinuous, discontinuous finite element, nodal, and characteristic methods. The source iteration approach is used to solve the discretized equations with various acceleration schemes such as the diffusion synthesis acceleration, coarse mesh rebalancing, partial current rebalancing, and multi-grid methods. Parallel computation capabilities are available in PARTISN, PENTRAN, and TORT. The time-dependent transport equation can also be solved using the PARTISN code, where the Crank-Nicholson method is used for time differencing.

VARIANT is a typical production code based on the second-order formulation. It solves multi-group transport problems in two- and three-dimensional Cartesian and hexagonal geometries. [31] It is based upon a variational nodal method that guarantees nodal balance and permits refinement using hierarchical complete polynomial trial functions in space and spherical harmonics or simplified spherical harmonics in angle. The even angular parity flux equations are solved within the nodes, and the continuity between nodes is provided by even- and odd-parity flux moments. The final algebraic equations are cast into a response matrix form and solved with red-black partial current iterations. Recent developments include a spatial kinetics capability, a sub-element method to treat the within-node heterogeneities, and a first-order spherical harmonics method to treat voided nodes.

3.3 Monte Carlo Codes

Monte Carlo codes can be used for lattice and whole-core calculations. Such codes include KENO, MCNP, and VIM. [32-34] Additionally, for analyzing accelerator-driven systems, a Monte Carlo code MCNPX has been developed by combining the high-energy physics code LAHET and the Monte Carlo code MCNP. [35,36] These codes allow accurate representation of nuclear data details and treatment of heterogeneity effects and complex geometries. They are however not routinely used at the current time for design calculations because of computational requirement.

As with other physics methods, significant improvements have been made in Monte Carlo codes with the rapid advance of computing power. Parallel computation capabilities have been implemented in most of production codes. Parallel processing has extended the applicability of Monte Carlo simulation to much wider range of problems. Library production approximations to accommodate limited computer memory have been reduced. The number of cross section points has been increased significantly, and the interpolation error thinning criteria have been tightened. To reduce the computational time by achieving acceptable statistics with fewer histories,

efficient global variance reduction techniques have also been developed. For example, the variational variance reduction methods have been developed for eigenvalue calculations. These approaches rely on global estimates of both forward and adjoint functions. The adjoint function is determined from a multigroup Monte Carlo simulation and is used to derive variance reduction parameters for the continuous Monte Carlo simulation of the forward problem. Some of these include hybrid approaches to determine efficiently the biasing parameters by using approximate deterministic calculations or defining accurate functionals for desired responses by using deterministic adjoint and Monte Carlo forward information.

The required Monte Carlo code improvements include the provision of capabilities to perform feedback calculations, the systematic propagation of uncertainties in Monte Carlo depletion analyses, capabilities for easing the task of creating input data for the code, and improving computational efficiency. Additionally, most of the Monte Carlo code packages do not have an adequate set of temperature dependent cross sections and only have a limited set of cross sections for representing fission products. It is noted that LANL has written an auxiliary code that enables the use of cross sections interpolated to user-specified temperature in the MCNP code.

3.4 Developmental Capabilities and Activities

Activities are ongoing in the U.S. to improve the accuracy of reactor physics analysis tools. [37] Efforts are being made to improve the current nodal approaches that employ regional (assembly) homogenization. The quasi-diffusion theory approach is being investigated to capture transport effects and provide theoretical foundation for improving the parameterization of neutron cross sections and homogenization parameters. A high-order boundary condition perturbation method is also being examined within the framework of diffusion theory to address this problem. Heterogeneous deterministic transport theory methods are also being developed based on the variational principle, focused on eliminating the need for the homogenization approximation.

Solution improvements are also being made by using embedded 1-D, pointwise energy formulation and improving the efficiency of transport methods by adapting the mathematical formulations to modern parallel computing platforms. To improve the deficiencies in angular discretization approaches used in discrete ordinates transport methods, the application of spatially adaptive quadrature sets and the wavelet technology are also being investigated. In a project, the spatial kinetics capabilities based on the method of characteristics approach that would eliminate the homogenization step and provide whole-core flux and power distributions is being developed. To represent the continuous refueling and resulting pebble motion in a PBMR reactor, a whole-core diffusion theory code is also being developed by coupling the neutronics solution with a pebble flow model.

4. Depletion Calculation Codes

For LWR analyses, the depletion calculation is generally incorporated with the lattice calculation. Depletion calculations are performed in the assembly calculations using lattice codes (CASMO-4, PHOENIX-P, TGBLA, HELIOS, WIMS8, DRAGON, etc.), and homogenized assembly cross sections are generated as a function of burnup as well as the other state variables. Whole-core nodal calculations are then performed by determining the individual assembly cross sections by interpolating these homogenized cross sections for the corresponding burnup states.

For fast-reactor fuel cycle calculations, depletion calculation is performed in conjunction with the whole-core calculation. At Argonne, the REBUS-3 code is utilized for equilibrium, non-equilibrium, and external cycle analysis. [38] This code uses DIF3D, VARIANT, or TWODANT as the flux solver. [39] Search options for fresh fuel enrichment, control poison density, or reactor burn cycle time are available, allowing the user to achieve a specified multiplication factor or discharge burnup without time consuming (trial and error) repetitions of the analysis. The calculational methods have been validated using EBR-II, FFTF, and other fast reactor data. As an example of the accuracy of the overall computational schemes, Table 2 compares the axial distribution of pin burnups of an IFR (Integral Fast Reactor) test assembly irradiated in EBR-II calculated with the DIF3D nodal flux solution option and reconstruction scheme with measured values. These results represent aggregate tests of nuclear data, cross section generation, core representation, full-core diffusion/depletion method, and reconstruction scheme. Recent code modifications include extension of code capabilities to handle accelerator-driven systems and implementation of MCNP as a flux solver.

Isotopic point depletion tools like ORIGEN and CINDER are typically used for analyzing the detailed radioactivity properties of depleted fuel. [40,41] However, they have recently been coupled to Monte Carlo tools for depletion calculations. Deficiencies in these capabilities are typically related to the nuclear data used in them. The ORIGEN-S code is the current ORNL supported version of the ORIGEN code. [42] Recent updates of cross sections, decay libraries, and neutron/gamma source data are contained in the code. However, updates for actinide fission yields are pending. The validation of these data for the advanced systems might be necessary and their correct utilization for this purpose would require the generation of datasets applicable to these systems.

There has recently been more reliance on Monte Carlo tools for depletion calculations. The method is particularly useful for analysis of specified designs but not sufficiently efficient for use in parametric and trade studies required for developing an optimized design. Coupled Monte Carlo and depletion code systems such as MOCUP, MONTEBURNS, and MCODE have been developed for the analysis of advanced systems. [43-45] These code systems provide linkage capabilities that couple the MNCP Monte Carlo code with the ORIGEN2 depletion code. The Monte Carlo technique is attractive because of the ability to represent accurately nuclear data details and to treat heterogeneity effects and complex geometries. Propagation of the Monte Carlo statistical uncertainty during depletion calculations has not been addressed in these tools and future work to quantify (and hopefully control) this problem is required. Without this, results

Table 2. C/E Comparisons for Burnup of an IFR Test Assembly in EBR-II

Fuel Pin	L/L_o	C/E_{La}	C/E_{Nd}	C/E_{avg}
No.31 (T167) (U-19Pu-10Zr)	0.08	1.063	1.028	1.045
	0.57	1.057	1.054	1.055
	0.94	1.019	0.978	0.998
N. 40 (T13) (U-8Pu-10Zr)	0.12	1.054	1.030	1.042
	0.57	1.069	1.046	1.057
	0.90	0.991	0.982	0.986
No. 13 (T119) (U-10Zr)	0.10	1.018	1.037	1.028
	0.45	1.038	1.025	1.032
	0.92	1.010	1.001	1.006

from such methods would be questioned due to un-quantified uncertainty. Work is also ongoing to develop a continuous-energy version of KENO and KENO/ORIGEN-S.

5. Conclusions

Neutronics tools used for the modeling of current and advanced reactor designs in the U.S.A. have been reviewed. Status of capabilities (NJOY and ETOE-2) for processing the evaluated nuclear data libraries for generating application code libraries was discussed. Current and developmental lattice capabilities used for LWR (CASMO-4, PARAGON, PHOENIX-P, HELIOS, DRAGON, WIMS8) and fast reactor (MC^2-2) modeling were summarized. A review of deterministic (ANC, DIF3D, PARCS, SIMULATE-3, REBUS-3, etc.) and Monte Carlo (MCNP, VIM, KENO, MOCUP, MONTEBURNS, MCODE) whole-core static and depletion codes was performed and discussed in the paper.

The review shows that an extensive amount of validation and qualification activities have been performed for these codes by comparing calculational results to those from higher fidelity methods (typically Monte Carlo codes), and critical experiments and measured plant data. These comparisons indicate that the existing neutronic analysis tools are sufficiently accurate for the design of currently existing power reactors and for early pre-conceptual design development and viability phase evaluations of advanced reactor designs. However, for refined analyses of the advanced systems, they require additional verification and validation tests. Additional improvements and capabilities might be needed in order to reduce computational uncertainties and improve the plant operational economics.

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