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ABSTRACT

Increasing complexity in reactor designs suggests a need to reexamine of methods applied in spent-fuel characterization. The ability to accurately predict the nuclide composition of depleted reactor fuel is important in a wide variety of applications. These applications include, but are not limited to, the design, licensing, and operation of commercial/research reactors and spent-fuel transport/storage systems. New complex design projects such as space reactors and Generation IV power reactors also require calculational methods that provide accurate prediction of the isotopic inventory. New high-fidelity physics methods will be required to better understand the physics associated with both evolutionary and revolutionary reactor concepts as they depart from traditional and well-understood light-water reactor designs. The TRITON sequence of the SCALE code system provides a powerful, robust, and rigorous approach for reactor physics analysis. This paper provides a detailed description of TRITON in terms of its key components used in reactor calculations.

Key Words: SCALE, TRITON, CENTRM, ORIGEN, depletion

1. INTRODUCTION

In response to concerns about the ability to accurately model the burnup of a light-water reactor (LWR) core containing weapons-grade mixed oxide (MOX) fuel assemblies, the Nuclear Regulatory Commission (NRC) has supported the enhancement of the generalized-geometry discrete-ordinates transport code NEWT [1,2]. NEWT provides lattice physics parameters for the Purdue Advanced Reactor Core Simulator (PARCS) code [3] used in the analysis of MOX-fueled LWR cores. Also supported under this work was the enhancement and formal release of TRITON [4] within the Standardized Computer Analyses for Licensing Evaluation (SCALE) system [5]. TRITON is a SCALE *control module* that enables depletion calculations to be performed by coordinating iterative calls between cross-section processing codes, NEWT, and the ORIGEN-S point-depletion code. NEWT is used to calculate weighted burnup-dependent cross sections that are employed to update ORIGEN-S libraries and to provide localized fluxes used for multiple depletion regions. TRITON uses a predictor-corrector approach to perform fuel-assembly burnup and branch calculations and generates a database of cross sections and other burnup-dependent physics data that can be used for full-core analysis.

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Although MOX-fueled cores may represent a challenge to existing analysis methods, next-generation reactor design concepts pose a substantial departure from traditional lattice designs and will initially be built and operated without the benefit of thousands of LWR plant-years of operation that serve as a basis for current design and analysis methods. Thus, high-fidelity computational methods will be necessary to better understand the physics associated with novel reactor concepts as they move from paper to reality. The TRITON sequence, based on the NEWT arbitrary-geometry transport solver, is able to perform two-dimensional lattice calculations for non-traditional lattice designs, including hexagonal arrays, even irregular-lattice (e.g., CANDU ACR-700 design) and non-lattice configurations. The more rigorous treatment of neutron transport available within NEWT, coupled with the accuracy of ORIGEN-S depletion capabilities and SCALE resonance self-shielding calculations within TRITON-driven lattice physics analyses, provides a rigorous first-principles approach for calculation of cross sections for such fuel designs. Additionally, the recent release of TRITON within SCALE 5.1 adds the ability to perform full three-dimensional depletion using the KENO V.a and KENO-VI Monte Carlo packages within SCALE.

This paper provides a description of TRITON, its components, and its capabilities. It also presents results of calculations that demonstrate the accuracy of the TRITON methodology for lattice physics.

2. OVERVIEW OF TRITON

SCALE is a modular system comprised of numerous sets of codes and data, with a broad range of functions and capabilities. Codes are classified as *functional modules, control modules,* or *utilities*. Functional modules include the basic physics codes, such as XSDRNPM (one dimensional [1-D] discrete ordinates), KENO (3-D Monte Carlo for criticality analysis), NEWT (two-dimensional [2-D] arbitrary geometry discrete ordinates), ORIGEN-S (point depletion and decay) and many other codes applicable to criticality, shielding, depletion, and radiation transport. Control modules operate as sequence controllers, preparing input for functional modules, transferring data, and executing functional modules in the appropriate sequence for a particular analysis type. TRITON is a SCALE control module that can be used for problem-dependent cross-sectional weighting, 2-D transport calculations with NEWT, or 2-D depletion calculations through a coupling of NEWT and ORIGEN-S. TRITON also supports 3-D depletion calculations using either KENO V.a or KENO-VI as transport solvers.

In the realm of lattice physics, the TRITON module is used to perform an iterative sequence of calculations. Given mixtures and cell structures defined in input, TRITON drives cross-section processing operations using BONAMI to perform Bondarenko calculations for resonance self-shielding in the unresolved resonance range and CENTRM/PMC for resolved resonance evaluation. Alternatively, a simplified resolved resonance treatment by NITAWL, using the Nordheim Integral Treatment, is available for U.S. Evaluated Nuclear Data Files (pre-ENDV/B-VI) data. The cross-section library and mixing table produced by the sequence are automatically used in the NEWT calculation so that no mixing table need be specified. The transport solution is followed by COUPLE and ORIGEN-S calculations. In depletion mode, NEWT creates a three-group weighted library based on calculated and volume-averaged fluxes for each mixture.

COUPLE updates the ORIGEN-S cross-section library with cross-section data read from the weighted library. Three-group fluxes calculated by NEWT are supplied to ORIGEN-S for depletion calculations. COUPLE/ORIGEN calculations are repeated for each mixture being depleted, as specified in input, using mixture-specific cross-section data and fluxes. Used in conjunction with TRITON, NEWT can generate a library of cross sections as a function of burnup, with a branch capability that provides cross sections at each burnup step for perturbations in moderator density, fuel and moderator temperatures, boron concentration, and control-rod insertion or removal.

Although ORIGEN-S supports the use of time-dependent fluxes, TRITON uses a more rigorous iterative procedure. Because spatial fluxes are burnup dependent, changing with nuclide inventories, and because mixture cross sections will also change with burnup, TRITON uses a predictor-corrector approach to update both fluxes and cross sections as a function of burnup. Such calculations can be considered to consist of two components during this iterative phase: (1) transport calculations (cross-section processing and the transport solution) and (2) depletion calculations. Transport calculations are used to calculate fluxes and prepare weighted cross sections and other lattice physics parameters based on a given set of nuclide concentrations; depletion calculations are used to update nuclide concentrations, which can be used in the following transport calculation.

The predictor-corrector process is introduced by performing a transport solution based on anticipated concentrations at a time halfway into a given cycle. Depletion calculations are then performed over the full cycle using fluxes and cross sections predicted for mid-cycle. Depletion calculations are then extended to a point halfway into the next cycle, followed by another mid-cycle transport calculation. This process is repeated until depletion calculations are completed for all cycles in a depletion case. In order to start the process, a startup case is required using initial isotopic concentrations for the initial transport solution. This solution is used as a basis for the first half-cycle isotopic estimate.

The physics of the depletion problem are solved by the sequential application of the codes described above. TRITON is responsible for processing input, conversion of the input description to appropriate forms, and execution of each module in sequence, with preparation and exchange of required data between modules. The following subsections describe the key components of this analysis sequence.

2.1. Rigorous Two-Dimension Deterministic Transport Calculations Using NEWT

The NEWT transport solver provides a two-dimensional, unstructured-mesh discrete-ordinates solution for multigroup neutron transport calculations for a wide variety of applications. NEWT employs a mesh defined by arbitrary polygons; extended step characteristic (ESC) discretization is applied to calculate transport between sides within each computational cell. Using the ESC solution method, computational cells are developed in the form of arbitrary polygons, which are further divided into trapezoids as a function of angle for a number of angles in a quadrature set. While similar in concept to the Method of Characteristic (MOC) class of transport solution, ESC provides a more rigorous spatial representation of bodies and allows significantly more control

of spatial mesh refinement. The theory of the ESC formalism is described in detail in Ref. 6. ESC differencing is compared and contrasted with MOC methods in Ref. 7.

Using a discrete-ordinates solution to the transport equation on an arbitrary grid, NEWT provides a robust and rigorous deterministic solution for non-orthogonal configurations. The arbitrary-polygon mesh developed by NEWT can be used to closely approximate curved or irregular surfaces to provide the capability to model problems that were formerly difficult or impractical to model directly with discrete-ordinates methods. Automated grid-generation capabilities provide a simplified user input specification in which elementary bodies can be defined and placed within a problem domain.

Specification of the geometric configuration within NEWT is based on a combinatorial geometry approach similar to that used in many Monte Carlo packages. In fact, the geometry package available with the Version 5.1 release of SCALE is based on the SCALE Generalized Geometry Package (SGGP). SGGP was originally developed at ORNL to provide the geometric flexibility necessary to develop extremely complex models for the KENO-VI Monte Carlo code. Although limited to two-dimensional analysis in an (x-y) plane in NEWT, the application of the combinatorial SGGP input structure provides the ability to easily develop models for complex configurations in two dimensions. Because it is limited to two-dimensions, certain bodies (e.g., spheres) cannot be modeled; however, the input flexibility is limited only by the 2-D constraint.

SGGP input is body based, with bodies placed, rotated, translated, and cut as required for the desired configuration. Bodies are placed into “units,” which are basic building blocks for model development. Units may be placed within other units, either in the form of a single placement or in the form of an array; NEWT supports both rectangular and hexagonal array structures.

In terms of body description and placement, the capabilities of the 2-D SGGP implementation in NEWT mirror the capabilities of KENO-VI. However, because the ESC methodology is really an advanced realization of the discrete-ordinates method, a grid structure must be introduced. Traditional discrete-ordinates methods are based on a grid structure of orthogonal cells in which diamond-differencing relationships are used to estimate transport across a computational cell. The ESC method uses a short characteristics approach to calculate transport across arbitrary polygons. Diamond-differencing is an approximation in which the derivatives of the flux are assumed to be linear across a computational cell, requiring cell sizes small enough to approximate this assumption. The characteristic solution is an exact representation of transport across a cell; however, approximations are introduced by the calculation of side-averaged fluxes, which limit the length of cell sides such that the variation of the angular flux along a side may be properly represented by an average. Both traditional and ESC discrete-ordinates methods are also constrained in size by the assumption of a constant source within the cell. Hence, a grid structure must be introduced to further subdivide the problem domain. In NEWT, this is achieved by the introduction of a rectangular grid; computational cells are defined by the intersection of the rectangular grid and placed bodies. The problem domain is characterized by a largely rectangular grid, which has additional polygonal structure at the intersection of bodies and rectangular cells. The rectangular grid need not be continuous, however, and may be locally refined as needed for capture of local effects.

NEWT may be used to model simple or complex configurations. Figure 1 illustrates a simple NEWT representation of a ¼ pin cell, in which the grid structure is easily observed. The user retains control on the number of line segments used to approximate curved surfaces; the code determines the inner radius of the regular polygon such that the volume matches the volume for the original circle. NEWT also provides an option to combine cells when small cells are created but are adjacent to a large cell of the same composition. In this figure, each continuous region, whether rectangular or an irregular polygon, comprises a computational cell.

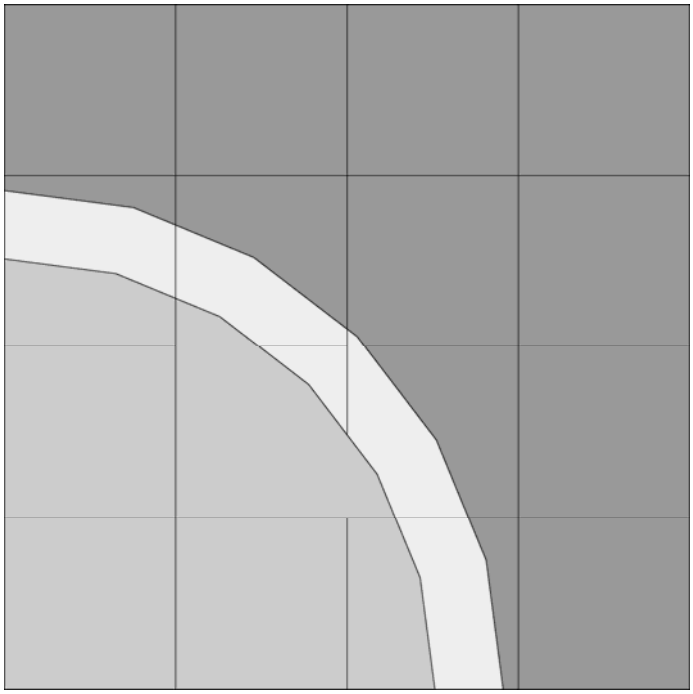


Figure 1. NEWT model for a ¼ pin cell with an underlying 4 × 4 rectangular grid.

In Fig. 2, a significantly more complex model is shown, developed for a code-to-code comparison. In this model, a 2 × 2 array comprised of a mixture of Atrium-10 and GE-10 type boiling water reactor (BWR) fuel lattices is shown. A single model (*unit*) for each of the two assembly designs was created; the SGGP input specification allows for rotation of units and placement using an array. In fact, this 2 × 2 array contains assemblies which themselves were developed using arrays.

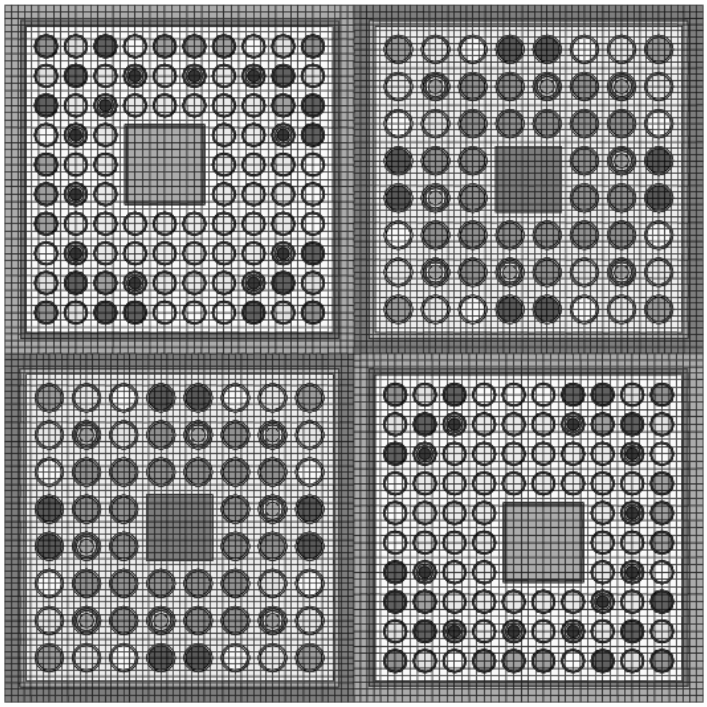
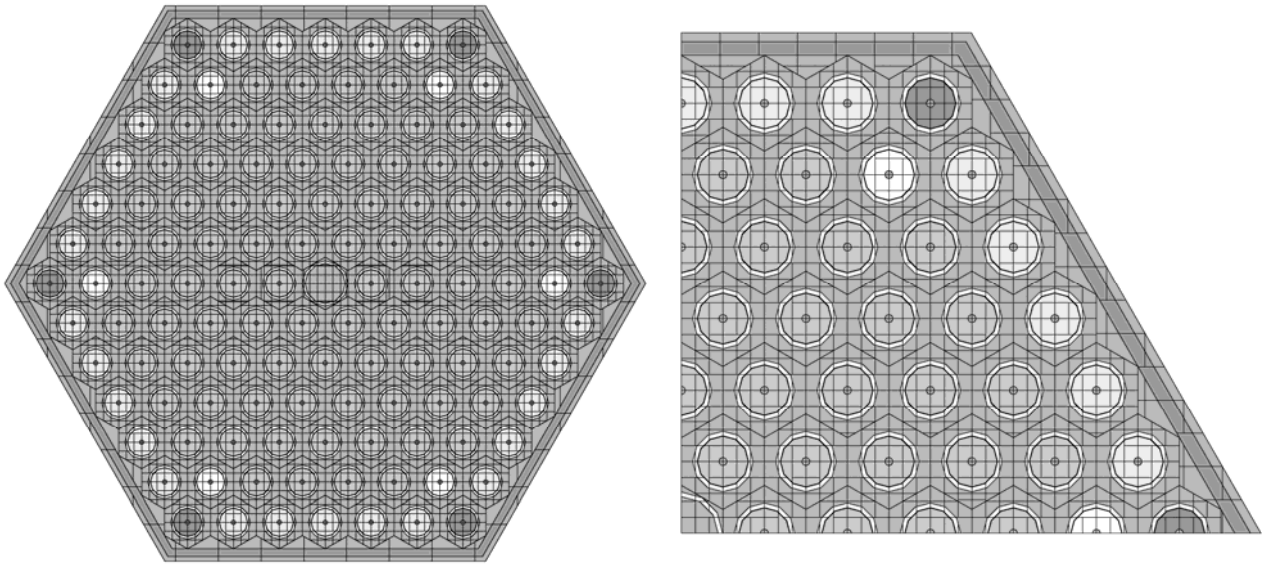


Figure 2. Complex 2 × 2 configuration with ATRIUM-10 and GE-10 BWR fuel lattices.

Finally, Fig. 3a illustrates a model developed for a VVER-400 fuel assembly. This case was also developed using an array; hexagonal arrays are also supported within NEWT. In this model, a more refined rectangular grid structure was used within each pin cell. A slightly more coarse rectangular grid is used in the vicinity of the inter-assembly channel and water gap. For this model, symmetry could have been used to reduce the problem to a ¼ assembly representation, as shown in Fig. 3b.

2.2. Continuous-Energy Resonance Processing via CENTRM/PMC

CENTRM computes continuous-energy neutron spectra in zero- or one-dimensional systems by solving the Boltzmann transport equation using a combination of pointwise and multigroup nuclear data. Several calculational options are available, including discrete ordinates in slab, spherical, or cylindrical geometry; collision probabilities in slab or cylindrical coordinates; and zone-wise or homogenized infinite media. CENTRM is used primarily to calculate problem-specific fluxes on a fine energy mesh (>10,000 points), which may be used to generate self-shielded multigroup cross sections for subsequent transport calculations. The purpose of the code is to provide highly accurate angular fluxes and flux moments for applications that require a detailed description of the fine-structure variation in the neutron energy spectrum. This is accomplished by first performing a CENTRM calculation for a simplified system model (e.g., a pin cell), then utilizing the spectrum as a problem-dependent weight function for multigroup averaging. The multigroup data processing is performed by the PMC code, which reads the



(a) (b)

Figure 3. NEWT representation of a full VVER-400 assembly.

CENTRM continuous-energy flux spectra and cross-section data; calculates problem-dependent, group-averaged cross sections over some specified energy range. The resulting problem-specific multigroup cross-section library can be passed to NEWT for multigroup deterministic calculations. In this approach the multigroup cross-section processing becomes an active component in the overall transport analysis, since the group averaging is tailored specifically to the system being analyzed.

CENTRM solves the Boltzmann transport equation over an energy range that typically varies from ~0 to 20 MeV. A distinguishing feature of the calculation is that a combination of multigroup and pointwise solution techniques allows a continuous spectrum to be computed over the full energy range of interest for reactor physics analysis. The continuous spectrum consists of values for the flux per unit lethargy defined over a discrete energy mesh, for which a linear variation of the flux between energy points is assumed. Several transport approximations are available for both the multigroup and pointwise calculations. Depending on the specified transport approximation, the flux spectrum may vary as a function of space and direction, as well as energy. Spherical harmonic moments of the angular flux, which may be useful in processing matrices for higher order moments of the scattering cross section, are also determined as a function of space and energy mesh. CENTRM solves the fixed-source (inhomogeneous) form of the transport equation, with a user-specified fixed source term. Within TRITON, the neutron source is an internally-computed fission-spectrum energy distribution appropriate for each fissionable mixture.

PMC calculates “multigroup” cross sections using point fluxes calculated in CENTRM and point cross-section data contained in a CENTRM pointwise continuous cross-section library. The purpose of the code is to provide problem-dependent cross-section data in multigroup format.

This task is performed by integrating point cross-section data with problem-dependent point fluxes that contain a detailed description of the fine-structure variation in the neutron energy spectrum. PMC reads the CENTRM continuous-energy flux spectra and cross-section data; calculates problem-dependent, group-averaged cross sections over some specified energy range; and then updates the corresponding data in an input cross-section library. In this manner the original library can be corrected for resonance self-shielding and other spectral effects.

2.3. Comprehensive Depletion/Decay Estimates with ORIGEN-S

A necessary component of accurate fuel calculations is the use of cross-section data libraries that define the nuclear reaction rates during irradiation. The one-group cross sections used by burnup codes change as a function of burnup, enrichment, and operating conditions and depend on the fuel assembly design. It is critical to accurately characterize the cross sections for the specific fuel type being analyzed. Most reactor physics codes are not capable of analyzing the full set of fuel isotopics necessary to accurately calculate all activation and fission products and daughters thereof. ORIGEN-S provides the versatility to use burnup-dependent cross section libraries produced via problem-specific transport calculations, and simultaneously track every activation, fission, and decay event for which data are available.

The ORIGEN codes have become one of the most widely used and internationally recognized class of depletion and decay codes in the nuclear industry for the comprehensive analysis of nuclide compositions, decay heat, and radiation sources from spent nuclear fuel. The modern version is ORIGEN-S, developed at ORNL within SCALE. The ORIGEN-S code currently tracks 1119 individual fission products generated in the fuel during irradiation, 129 actinides, and 698 isotopes associated with structural and/or activation components. The standard cross-section libraries used by ORIGEN-S for its depletion and decay calculations have been extensively updated. This work entailed the development of broad multigroup neutron cross sections for ORIGEN-S from several sources of pointwise continuous-energy cross-section evaluations, including the U.S. Evaluated Nuclear Data Files ENDF/B-VI Release 7, the Fusion Evaluated Nuclear Data Library FENDL-2.0, and the European Activation File EAF-99 [8]. The pointwise cross sections were collapsed to a three-group structure using a continuous-energy neutron flux spectrum representative of the typical neutronic conditions of the type of fuel and formatted for use by ORIGEN-S. In addition, the fission-product library has been expanded to include ENDF/B-VI fission yield data for 30 fissionable actinides. The updates are expected to increase the versatility and accuracy of ORIGEN-S for a wide range of potential applications including decay-heat estimations and advanced reactor design studies, as well as very high-burnup fuel, MOX fuel, and actinide transmutation applications. The procedures used to generate the new cross sections in ORIGEN-S LWR libraries are completely general and can easily be extended to develop new libraries for other reactor designs and design concepts such as the Pebble-Bed Modular Reactor (PBMR), the Advanced High Temperature Gas-Cooled Reactor (AHTGR), and others.

The ORIGEN-S code has been extensively validated against destructive radiochemical measurements for more than 100 spent-fuel samples from domestic and international programs involving older [9] and modern high-burnup MOX and low-enriched uranium (LEU) fuels,

decay heat measurements for more than 120 assemblies spanning cooling times up to 30 years [10], and neutron and gamma radiation measurements. The extensive validation of ORIGEN-S is a critical element in establishing the reliability of the code to support detailed lattice physics analysis with few approximations.

2.4. TRITON depletion using Monte Carlo methods

Despite broad applicability of the 2-D fuel depletion analysis capability of TRITON, there are some domains in which accurate 3-D depletion capabilities are necessary. For example, characterization of commercial spent fuel in transportation and storage is concerned with the positive reactivity effects of low-burnup fuel near the ends of a fuel assembly where axial leakage effects, not captured by 2-D methods, may be important. Deterministic transport methods are also unable to perform full-core analysis in a practical sense because of the computational overhead of such large-scale discretization. Additionally, conceptual advanced reactor designs depart from traditional design attributes to the extent that more robust 3-D methods may be required to track fuel depletion or provide reference solutions for 2-D methods. For these reasons, among others, a 3-D depletion capability has been integrated into TRITON, using the 3-D Monte Carlo–based KENO V.a and KENO-VI codes in SCALE. The TRITON sequence has been modified to accept input for either KENO module in place of NEWT input and to use the Monte Carlo codes as the transport solver component of the sequence. The SCALE utility codes KMART and KMART6, originally developed to post-process KENO V.a and KENO-VI calculations, have been adapted to provide collapsed cross-sections and fluxes required by TRITON for setting up ORIGEN-S depletion calculations. However, while Monte Carlo methods provide powerful geometric modeling capabilities in three dimensional domains, such calculations must typically simulate a large number of neutrons to converge on an accurate system response (e.g., global neutron multiplication). In order to achieve reasonably precise neutron fluxes and power density distributions, significantly more computational effort must be invested. This computational effort is compounded in a depletion calculation, where transport solutions must be repeated in an iterative sequence alternating with depletion calculations to update isotopic cross sections and inventories. And while deterministic solutions are based on fluxes that are converged to a specified degree over the full problem domain, the nature of Monte Carlo simulations makes it extremely difficult to obtain accurate fluxes in locations that are far removed from the most reactive region of an analysis domain. Since the accuracy of the neutron flux is therefore a function of position in a Monte Carlo simulation, the accuracy of the depletion solution is likewise spatially distributed.

Because the uncertainty associated with the generation of lattice physics parameters based on Monte Carlo methods has not been studied, along with other lesser issues, TRITON does not permit use of either Monte Carlo sequence in the generation of collapsed few-group cross sections for subsequent nodal core calculations. Branch calculations, typically needed in core analysis methods, are also not permitted in the KENO-based depletion sequences. Nevertheless, all other applications of TRITON are available, whether the transport solver is NEWT or one of the KENO codes. It is also important to note the fact that the KENO codes use the same multigroup cross section data and interact with ORIGEN-S in exactly the same manner as NEWT, and the KENO sequences provide a method to directly and independently assess the effect of any transport approximations made in NEWT. This has proven invaluable in testing

and validation efforts related to NEWT. The converse is also true – errors associated with Monte Carlo uncertainties can be easily tested by running 2-D analyses calculated by both NEWT and KENO sequences. Lastly, the fact that NEWT and KENO-VI share a common geometry-description paradigm makes it possible to rapidly interchange models between packages.

2.5. TRITON: Pulling the Pieces Together

The rigor of the NEWT solution in estimating angular flux distributions combined with the world-recognized rigor of ORIGEN-S depletion gives TRITON the capability to perform precise burnup-dependent physics data with few implicit approximations, limited primarily by the accuracy of nuclide data. Such rigor may be necessary to capture the unique attributes of advanced, highly heterogeneous fuel-assembly designs being deployed in current-generation reactors, as well as next-generation candidate concepts. As these concepts evolve, however, it becomes clear that 2-D methods may not be feasible or practical in representing configurations that are strongly characterized by 3-D behavior. In such applications, such as the PBMR, Monte Carlo analysis methods may be necessary to properly capture 3-D effects.

TRITON was originally developed for spent-fuel characterization. The SAS2 sequence, long available within SCALE, uses ORIGEN-S with the 1-D discrete-ordinates code XSDRNPM. However, the 1-D approximation works well only with relatively homogeneous assembly lattices and was only able to perform depletion for a single assembly-averaged mixture. Using the 2-D capabilities of NEWT provides the ability to perform a more precise transport solution; TRITON builds on this capability by allowing independent depletion of multiple mixtures through multiple loops over COUPLE/ORIGEN-S. The nature of the ORIGEN-S depletion package provides a broad range of data beyond time-dependent isotopic concentrations for more than 1000 nuclides. Decay heat sources, neutron and gamma spectra and source strengths, and radiotoxicity of fuel, target, and other activated materials are available from ORIGEN-S. Hence, these data are also available from TRITON, either for a specific mixture or as the aggregate for an entire system.

SAS2 has also been used to generate cross section libraries for the ORIGEN-ARP (Automatic Rapid Processing) code within SCALE. This capability is now also available within TRITON. This approach provides an alternative to the full TRITON sequence for spent fuel characterization. In the ARP approach, burnup-dependent libraries are generated for a given reactor design using TRITON. These pre-generated libraries are suitable for use by the ORIGEN-S code. By interpolation among these libraries, point-depletion ORIGEN-S calculations can be carried out for that particular design of reactor without requiring the full complexity of additional TRITON calculations. Depending on requirements or on the reactor design being analyzed, the libraries allow for interpolation on three variables: burnup, enrichment, and water (moderator) density. ARP libraries released with SCALE 5.1 are based on TRITON calculations and represent a broader class of fuel designs than could be modeled using SAS2 [11].

Extensive testing has been performed to verify proper operation of TRITON and to validate its ability to predict a system eigenvalues, both for critical systems and as a function of burnup; to accurately estimate spent-fuel inventories, and its ability to provide properly weighted few-group cross sections for core simulator calculations [10-15]. Calculations have been compared to measured data when available, but code-to-code comparisons have also been used to assess certain capabilities where little or no meaningful measurements are available.

3. COMPARISON OF TRITON TO OTHER CODES

TRITON was selected by the NRC because of the rigor of the transport solution of NEWT; however, TRITON is clearly not limited to MOX fuel analysis. Ongoing validation analyses have included a suite of code-to-code comparisons initiated by the Japan Atomic Energy Research Institute (JAERI) for LWR Next Generation fuels. [16] The benchmark report features results for a variety of analysis codes, methods, and data for pin cells and for both PWR and boiling water reactor (BWR) fuel assemblies, each with both UO_2-only and MOX fuel loadings. Reported results include k_{eff}, pin powers, and reactivity calculations. TRITON calculations have recently been completed for the UO_2 BWR benchmark. Figure 4 shows the performance of TRITON depletion calculations relative to other depletion approaches. TRITON results deviate from the other codes at high burnup by over-predicting reactivity. It is believed that this is due to the inclusion of insufficient numbers of fission products in the transport model. Nevertheless, TRITON results are in excellent agreement with the results submitted by other participants. Void reactivity calculations performed as part of the benchmark are illustrated in Fig. 5. Again, TRITON shows excellent agreement with the results of other participants.

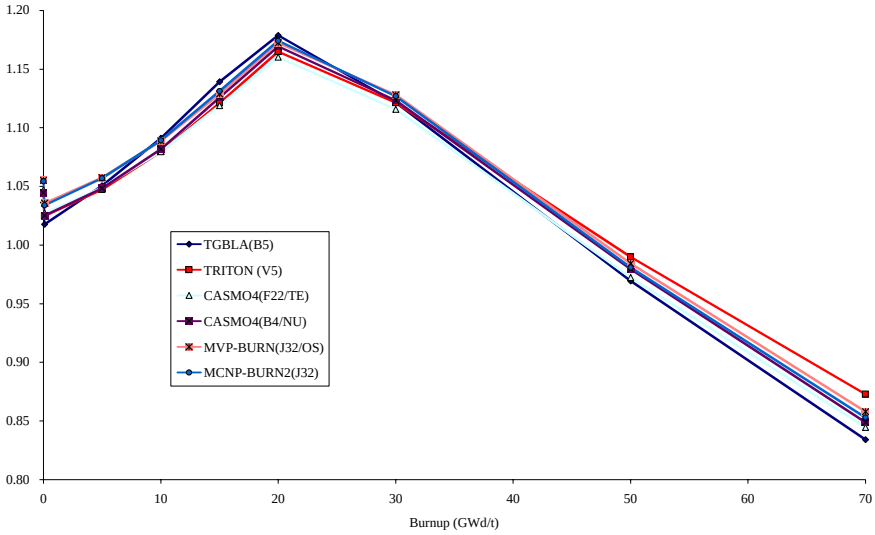


Figure 4. K-infinity predictions as a function of burnup for the JAERI BWR-UO_2 assembly benchmark.

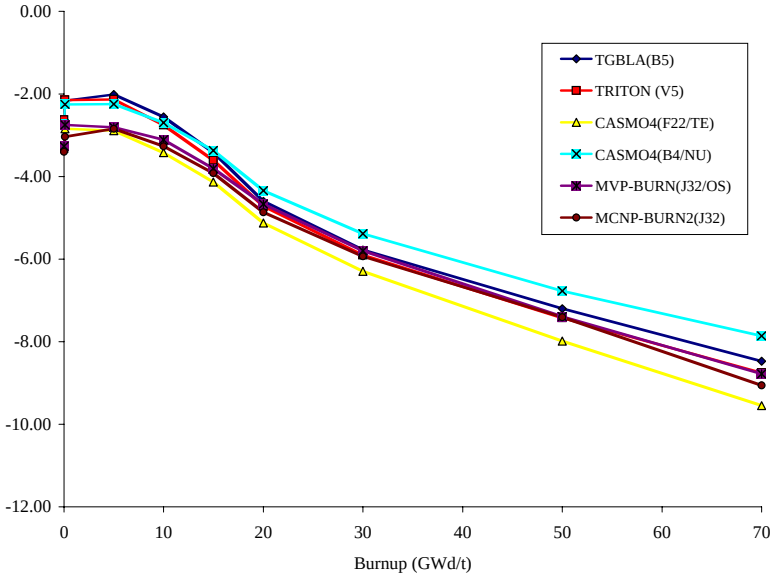


Figure 5. Void reactivity (0% – 70%) as a function of burnup for the JAERI BWR-UO_2 assembly benchmark.

4. CONCLUSIONS

The TRITON module of the SCALE code system provides a powerful and robust approach for transport and depletion analysis of reactor fuel assemblies. Coupled with a core simulator, accurate core-follow calculations can be performed for complex fuel designs. The arbitrary-geometry features of NEWT and the high fidelity of the combined CENTRM/NEWT/ ORIGEN-S sequence provide a powerful tool for the study of advanced designs and nontraditional fuel-bundle concepts. The input geometry specifications for NEWT, based on the SCALE Generalized-Geometry Package (SGGP) combinatorial-geometry format of the KENO-VI Monte Carlo code with SCALE, provide a simple means for rapid model development, ranging from simple pin cells and regular lattices to irregular arrangements of nuclear materials. Additionally, the similarity of the geometry model allows simple translation to and from KENO-VI calculations, in which independent transport solutions can be compared for code-to-code validation.

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