ORNL/LTR-2012/559

Reactor and Nuclear Systems Division (94)

LETTER REPORT

GENERATION OF COLLAPSED CROSS SECTIONS FOR HATCH 1 CYCLES 1-3 AND GENERATION OF GENERIC BWR REFLECTOR CROSS SECTIONS

Brian J. Ade

Oak Ridge National Laboratory
P.O. Box 2008
Bldg. 5700, MS-6170
Oak Ridge, TN 37831-6170
adebj@ornl.gov
(865) 574-6116

November 26, 2012

Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555-0001

NRC JCN V6361

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831-6170
managed and operated by
UT-Battelle, LLC
for the U.S. Department Of Energy
under Contract No. DE-AC05-00OR22725

CONTENTS

Page

LIST OF TABLES

Page

LIST OF FIGURES

Page

1. INTRODUCTION

Under NRC JCN V6361, Oak Ridge National Laboratory (ORNL) was tasked to develop and run SCALE/TRITON [1] models for generation of collapsed few-group cross sections and to convert the cross sections to PMAXS format using the GENPMAXS [2] conversion utility for use in PARCS/PATHS [3,4] simulations of Hatch Unit 1, cycles 1-3 (herein referred to as "Hatch") [5,6]. This letter report documents the final models used to produce the Hatch collapsed cross sections.

2. BACKGROUND

This project is similar to recent projects V6222 (generation of SVEA96-Optima2 cross sections) [7,8], N6825 (generation and testing of lattice parameters for an ABWR core) [9,10], V6098 (former N7098) (generation and testing of lattice parameters for MELLLA+ for a BWR [11], and V6182 Task 2 (SCALE/TRITON and PARCS/PATHS code validation for Peach Bottom Unit 2 cycles 1 and 2) [12]. The fuel assembly and reactor design data are very similar to those used in Ref. 12 (original design data from Ref. 13), and the basic models and assumptions were based on the previous work in Ref. 12.

Under NRC JCN V6361, tasks were included for generation of nodal data for all fuel assemblies in Hatch cycles 1-3, and to generate reflector nodal data that could be applied as a general set of reflector nodal data, rather than a specific set of data for a particular reactor and fuel assembly design. At the start of the project, ORNL located the Hatch EPRI reports in the ORNL Research Library. NRC staff member Peter Yarsky found that the data in the Hatch EPRI report for cycles 2 and 3 (NP-2106 [6]) contained inaccurate design data in reference to the type-4 and type-5 fuel assemblies. Ref. 6 indicates in multiple places that the type-4 and type-5 fuel assemblies contain gadolinia-bearing fuel pins, however the content (weight percent of gadolinia) and layout of these fuels pins are not specified in the report. Dr. Yarsky contacted GE-Hitachi, who provided the missing data [14].

Previous work [7–12, 16] highlighted the need to calculate fuel rod Dancoff factors independently of the standard TRITON/MIPLIB process for highly heterogeneous BWR fuel assemblies. The SCALE module MCDancoff [1] was employed to calculate Dancoff factors for all fuel rods each of the models using varying thermal-hydraulic (T-H) conditions. The Dancoff factors where then applied in the SCALE/TRITON input files for the history and branch states. SCALE 6.1 was used for all calculations in this report. Previous work utilized extensive KENO continuous energy (CE) testing prior to production model development [7,8]. However, because the fuel assemblies used in Hatch are very similar to those in Ref. 12 and because the fuel assemblies are far less heterogeneous than those in previous work, detailed testing was not required; the now standard procedure of using MCDancoff-calculated Dancoff factors has been assumed to be sufficient for the current work.

SCALE 6.1 on ORNL computer cluster *cpile2* was used to perform the production calculations. While running the production calculations, partial eigenvalue trajectories were extracted from the TRITON *txtfile16* output files and plotted. A visual inspection revealed no unexpected results. After the SCALE/TRITON calculations were complete, all eigenvalue trajectories were extracted

and plotted in Microsoft Excel. A detailed visual inspection of the eigenvalue trajectories revealed no major issues.

The nodal data contained in SCALE/TRITON *xfile016* output files were converted to PMAXS format using GenPMAXS version 6, obtained from University of Michigan (UM) staff. After converting the nodal data to PMAXS format, various PMAXS files were spot-checked to ensure that the data were properly converted. The nodal data and the spreadsheet used to plot eigenvalue trajectories were encrypted and uploaded to ORNL file servers where they could be downloaded by NRC staff.

3. DESIGN DOCUMENTS

Design data were provided in Refs. 5 and 6, with missing information provided by NRC staff member Peter Yarsky [14]. In addition, the design data from previous projects [7–12] were used when modeling reflector nodal data in order to provide data applicable to many types of BWRs. The actual data from other projects were reviewed to ensure general applicability of the reflector nodal data, but they have been omitted from this report to protect proprietary information. All depletion histories and branch states were chosen based on the recommendations found in Ref. 15.

4. ASSUMPTIONS

Minor shortfalls in the design data led to some minor assumptions in the input files, as documented in the following subsections.

4.1 Reflector Modeling

Various assumptions are required to generate SCALE/TRITON models suitable for generating reflector nodal data. In order to provide a neutron source for reflector calculations, a fuel assembly is required. The radial reflector is modeled using a mid-assembly fuel lattice, while the top and bottom axial reflectors are modeled using a natural uranium-fueled lattice (typical of the ends of modern BWR fuel assemblies). The core shroud is modeled as a flat surface, rather than a cylindrical surface, and is assumed to be located at a distance corresponding to the average distance from the outermost row of fuel to the inner surface of the shroud. The radial reflector is assumed to terminate at the inner surface of the core vessel where a vacuum boundary condition is assumed. There has been no differentiation between radial side and corner reflectors in this analysis. Nodal data have not been generated specifically for corner reflectors, as the adjustment performed by GenPMAXS to generate the corner radial reflector from the radial side reflector data is typically minimal for BWRs.

As previously stated, the bottom and top axial reflectors are modeled using a natural uranium-fueled lattice as a neutron source. The Hatch documentation contains more details pertinent to reflector modeling than previously used design data – various masses of materials such as end plugs, plenum springs, and support plates are provided. These data were used to generate the axial reflector models and should result in increased accuracy compared to previously generated data. In general, the top and bottom axial reflectors have been modeled as 2 homogenous

regions containing a mixture of structural materials located in that region plus moderator (water). The bottom axial reflector was modeled as always having the influence of a control blade because, when fully withdrawn, the control blade will still be present in the region immediately below the fuel assemblies. A particularly wide set of thermal-hydraulic conditions (including borated conditions) were modeled in the reflector models to ensure applicability to many different transient scenarios.

4.2 Control Blade Tip Modeling

The control blades used in Hatch have a rounded tip, however, the blade tip has been modeled as square in this analysis for modeling simplicity. The impact is neutronically insignificant.

4.3 Various Other Assumptions

During model development, various other assumptions were made. The assumptions are listed here with a short description.

1. Cross-section libraries — SCALE's ENDF/B-VII 238-group library was the basis for all calculations. SCALE/TRITON's "parm=weight" option was used to collapse the 238-group master library to a 49-group problem-dependent library at time T = 0.
2. Resonance processing — all calculations use CENTRM/PMC [17], for resonance self-shielding. Non-gadolinium-bearing fuel rods use a lattice cell treatment in CENTRM, while gadolinium-bearing fuel rods use a multiregion treatment with five equal-area rings to capture radial depletion of gadolinium.
3. Depletion — all fuel mixtures were depleted individually. Non-gadolinium fuel rods were depleted by constant power, while gadolinium-bearing rods were depleted by constant flux. To reduce computing time, the SCALE/TRITON "assign" function was used to group rods for resonance processing purposes. Tests have shown that the use of this technique results in less than ~10 pcm of bias in k_{inf} compared with using a CENTRM model for every rod. An optimized depletion step scheme was used that has been shown in internal studies to provide accurate results for gadolinia-bearing fuel assemblies.
4. Transport model — All fuel mixtures and structural material use P_1 scattering, while all moderator mixtures use P_2 scattering. Convergence criterion for the eigenvalue was set at 1.0E-5. Coarse-mesh finite-difference acceleration was used on the global grid. A 4×4 grid mesh was used for all unit cells in the reference calculations.
5. History calculations — void histories of 0%, 40%, 70%, as well as one history corresponding to 40% void with control rods in, were included (recommend in Ref. 15).
6. Branch calculations — Branch calculations were chosen based on recommendations in Ref. 15. Additional branch calculations corresponding to cold conditions were included at the request of the NRC Project Manager.

5. METHODOLOGY

Prior to production model development, MCDancoff models were generated for each unique lattice. These models were used to calculate Dancoff factors for all fuel pins in the lattice at varying thermal-hydraulic conditions. These Dancoff factors were used in production model

development.

Prior to running production calculations, separate 49-group cross-section (XS) libraries were generated for each lattice type (12 total) and each history (4 total). Each 49-group library was generated from the SCALE ENDF/B-VII 238-group XS library by weighting with the problem-dependent flux at BOL. Using a 49-group master library represents a factor of ~4 improvement in the computational time required for transport calculations. The problem-dependent 49-group libraries introduce a bias that is typically less than 100 pcm versus the 238-group library. Gadolinium-bearing fuel rods were modeled with five equal-area rings in order to accurately capture the time- and space-dependence of the depletion of gadolinium in those pins. Internal studies have found five equal-area rings provide accurate results with reasonable computer time. Calculations were performed using SCALE/TRITON's parallel capabilities that allow a user to spread the calculations of branch data over many different computer processors.

6. HATCH FUEL ASSEMBLIES

Hatch cycles 1-3 contain five different fuel assemblies. The initial (cycle 1) fuel assemblies are GE-designed 7x7 fuel assemblies (assembly types 1-3), and the reload assemblies (cycles 2 and 3) are GE-designed 8x8 fuel assemblies (assembly types 4 and 5). All fuel assemblies contain a channel that separates in-channel boiling water from out-channel saturated liquid water. Assembly types 1-3 are essentially identical in physical design, but assembly types 4 and 5 both contain different physical design features from the other assemblies. In addition to having an 8x8 fuel pin layout, assembly type 4 contains a single, small water rod filled with saturated liquid water. Assembly type 5 is an 8x8 fuel design that contains two small water rods filled with saturated liquid water. A SCALE/KENO representation of the three major assembly types can be found in Figure 1.

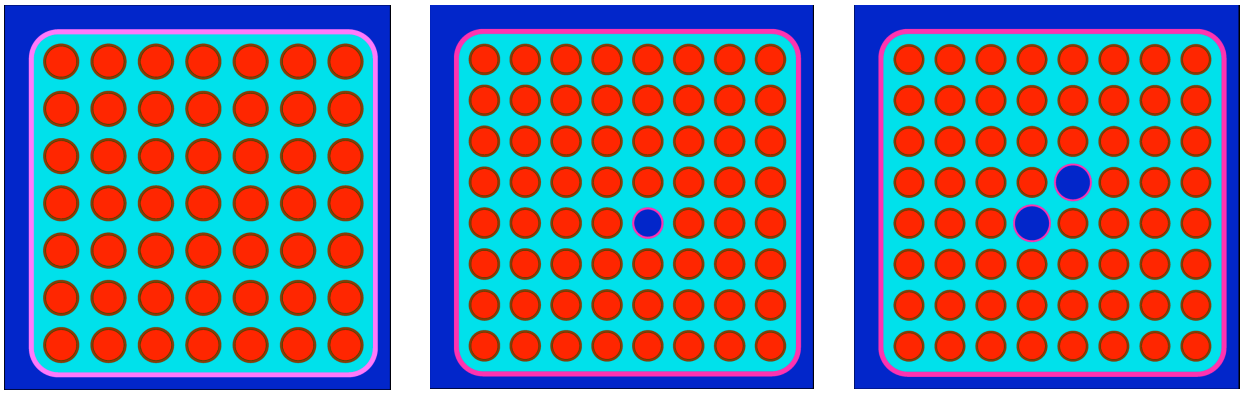


Figure 1: SCALE/KENO-VI representations types 1-3 (left), type 4 (middle), and type 5 (right) Hatch fuel assemblies.

Figure 2 is a SCALE/TRITON representation of the bottom level of the type 1 fuel bundle. The SCALE/TRITON model uses different mixtures for each fuel pin, resulting in many different colored fuel pins in the figure. The SCALE/TRITON model has been developed such that the control blade is modeled in the northwest corner of the lattice. A detailed description of each lattice modeled can be found in Section 10.

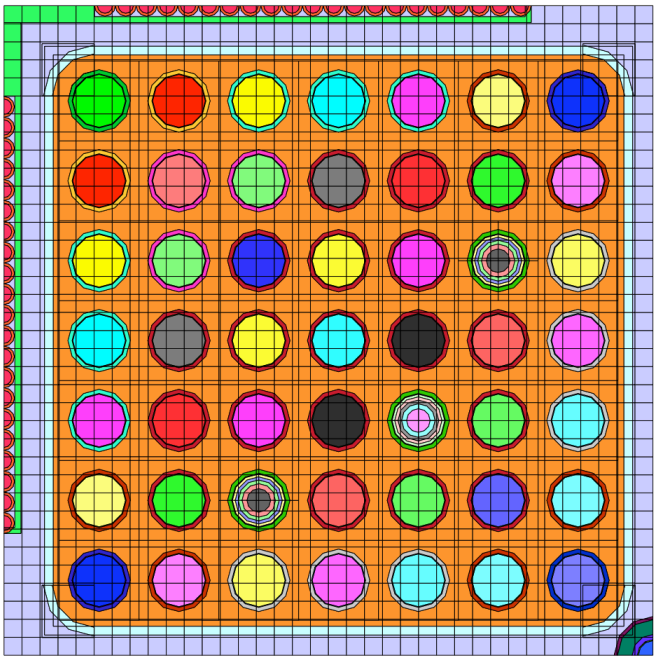


Figure 2: SCALE/TRITON-NEWT representation of the bottom region (lattice 1) of the type 1 fuel assembly.

7. MODELING PARAMETERS

The following sections describe the parameters used in the various assembly models.

7.1 Assembly Dimensions

The Hatch 1 power plant has an assembly pitch of 6.0 inches (15.2 cm) and used GE-designed 7x7 and 8x8 fuel assemblies in the first three cycles. Basic fuel assembly dimensions are provided in Table 1.

Table 1. Assembly dimensions

Quantity	Dimension (cm)
Lattice	7x7 and 8x8
Assembly pitch	15.24 (6.0 in.)
Fuel/tube pitch	1.87452 (7x7) and 1.62560 (8x8)

7.2 Fuel Pin Dimensions

Hatch cycles 1-3 utilized five different fuel assembly types. The initial fuel assemblies (types 1-3) utilize fuel pins with identical dimensions. The reload fuel assemblies, type 4 and type 5, each utilize unique fuel pin dimensions. The fuel pin dimensions are summarized in Table 2.

Table 2. Radii for fuel cells

Quantity	Dimension (cm)
Pellet radius	0.60579 (types 1-3) 0.52832 (type 4) 0.52070 (type 5)
Clad inner radius	0.60579 (types 1-3) 0.53975 (type 4) 0.53213 (type 5)
Clad outer radius	0.71501 (types 1-3) 0.62611 (type 4) 0.61341 (type 5)

7.3 Control Blade Dimensions

The control rod blade assembly dimensions used in all fuel assembly models are summarized in Table 3. The Hatch control rod blades contain a rounded tip that has been modeled as square in this analysis.

Table 3. Dimensions for the Hatch control rod blade

Quantity	Dimension (cm)
Blade + central support span	24.76500
Central support span	3.97002
Absorber tube inner radius	0.41402
Absorber tube outer radius	0.47752
Sheath thickness	0.14224

7.4 Fuel Assembly Channel, Water Rods, and Assembly Gap

The Hatch fuel assemblies each contain a fuel channel that separates saturated liquid water (out-channel) from saturated boiling (2-phase) water (in-channel). In addition, the type 4 and type 5 fuel assemblies contain one and two water rods, respectively. Like the out-channel region, the water rods are filled with saturated liquid water and are also referred to as out-channel. Relevant channel dimensions are summarized in Table 4.

Table 4. Various channel dimensions

Quantity	Dimension (cm)
Channel width (outside)	13.8125 (all lattices)
Channel thickness	0.2032 (all lattices)
Water gap half-thickness	0.95250 (wide) 0.47498 (narrow)
Water rod inner radius	N/A (types 1-3) 0.53975 (type 4) 0.67437 (type 5)
Water rod outer radius	N/A (types 1-3) 0.62611 (type 4) 0.75057 (type 5)

7.5 Materials

In each fuel lattice, the fuel composition was assumed to consist of isotopes ^{234}U, ^{235}U, ^{236}U, and ^{238}U, where the ^{234}U component is 0.89% of the ^{235}U concentration, and the ^{236}U component is 0.46% of the ^{235}U concentration [18]. A fixed UO_2 density of 10.42 g/cm^3 was used for each non-burnable absorber fuel mixture [5,6]. A number of fuel pins utilizing Gd_2O_3 as a burnable absorber are used in each of the fuel assemblies. The Gd-bearing fuel rods contain different concentrations of Gd_2O_3 ranging from 1.5 wt% to 4.0 wt% Gd_2O_3 [5,6,14]. All fuel densities are summarized in Table 5.

Table 5. Fuel density in subject fuel types

Fuel Type	Fuel Density (g/cm^3)
UO_2	10.42
98.5 wt% UO_2 + 1.5 wt% Gd_2O_3	10.35
97.5 wt% UO_2 + 2.5 wt% Gd_2O_3	10.32
97.0 wt% UO_2 + 3.0 wt% Gd_2O_3	10.28
96.0 wt% UO_2 + 4.0 wt% Gd_2O_3	10.26

Fuel cladding was modeled as Zircaloy-2 with a density of 6.56 g/cm^3 (the SCALE default density). The water rods, the fuel channel, and a portion of the fuel spacer material were modeled as Zircaloy-2 at a density of 6.56 g/cm^3 (the SCALE default density).

Water moderator density and temperatures were based on the saturation temperature and densities for an operating pressure of 7.03 MPa (1035 psia). The density of the two-phase water-steam mixture was calculated as

$$\rho_{2-phase} = \rho_{liquid}(1 - VF) + \rho_{gas}(VF),$$

where,

$\rho_{2-phase}$ = density of the two-phase mixture;
ρ_{liquid} = density of the saturated water;
ρ_{gas} = density of the steam;
VF = void fraction.

The resulting moderator densities are 0.7373 g/cm^3 (0% void, hot), 0.4573 g/cm^3 (40% void), 0.2473 g/cm^3 (70% void), and 0.1073 g/cm^3 (90% void). Moderator density corresponding to 0% void, hot conditions were applied for all out-channel moderator mixtures (including water rods, if applicable) regardless of the in-channel moderator density. In addition to these conditions, moderator density branches of 0.9982, 0.9922, 0.9837, and 0.9718 g/cm^3, corresponding to 293 K, 313 K, 333 K, and 353 K, respectively, were included [19].

The fuel pin gaps are filled with helium whose density was calculated using the ideal gas law assuming a backfill conditions of 10 psi and 70°F [5,6]. The gap was assumed constant during depletion.

The control blades are composed of B_4C and SS304. The B_4C inside the absorber tubes is assumed to be natural boron at 70% theoretical density (1.765 g/cm^3) [5,6]. SS304 at the SCALE default density is used for all control blade structural materials. SCALE standard compositions for B_4C and SS304 were utilized.

The Hatch fuel lattices were modeled with a neutron detector in the southeast corner of the assembly in order to generate detector data to be used in follow-on calculations. SCALE standard compositions at default densities were used for the materials in the detector tube. Further detail regarding detector modeling information can be found in Sect. 12.

For the reflector calculations, homogenized regions were utilized for the bottom and top axial reflector. These homogenized regions contain a mixture of many different materials further detailed in Sect. 11. SCALE standard compositions at the default densities were used for the materials in the homogenized reflector regions.

7.6 Temperatures

Nominal fuel and moderator temperatures used in the analysis are documented in Table 6.

The average fuel temperature was calculated using equation 8-119 in Ref. 20, based on the moderator temperature. The moderator temperature is equal to the saturation temperature of the water at 7.03 MPa. The clad temperature was calculated using a weighted average of the moderator and fuel temperatures ($0.8*T_{mod} + 0.2*T_{fuel}$). The gap temperature was calculated as the average temperatures of the fuel and clad. All control rod blade assembly material temperatures were assumed equal to the moderator temperatures. All other structural and detector materials were also assumed equal to the moderator temperature.

Table 6. Temperatures in different regions of the assembly

Region	Temperature (K)
Fuel	948.45
Gap	792.26
Clad	560.29
Water	560.29
Structural materials	560.29
Control blade materials	560.29

7.7 Power Density

The average power density, 23.5 W/g (or MW/MTU) was calculated using the rated reactor power and the number of fuel assemblies (and the weight of uranium in those assemblies) in the reactor.

7.8 Fuel Exposure

A maximum exposure of 61.5 GWd/MTU was used for all simulations.

8. BRANCH AND HISTORY CALCULATIONS

Based on the recommendations outlined in Ref. 15, depletion histories corresponding to the following conditions were used to generate nodal data:

T_{fuel}	948.45 K
T_{mod}	560.29 K
ρ_{mod}	0.7373 g/cm^3 (0% void), CR out
	0.4573 g/cm^3 (40% void), CR out
	0.2473 g/cm^3 (70% void), CR out
	0.4573 g/cm^3 (40% void), CR in.

In order to provide cross sections for anticipated thermal-hydraulic states during steady-state depletion, branch calculations are required. TRITON provides the capability to perform branch calculations for fuel and moderator temperatures, moderator density, soluble boron concentration, control rod state (in or out), and Dancoff factors for associated conditions.

As recommended by Ref. 15, moderator density branches corresponding to 0%, 40%, 70%, and 90% void were performed. In addition, at the request of NRC staff who will utilize the cross sections various calculations, 0% void branches with temperatures of 293, 313, 333, and 353 K were performed. In these four cold states, both moderator and fuel temperatures were modified to obtain data for a true cold state. In the cold states, only the fuel and moderator temperatures were modified – gap, clad, and other structural materials were omitted from the temperature change. The bias introduced due to this treatment is expected to be small because Doppler feedback is small for all structural materials in the assemblies. Fuel temperature branches of 500.00, 948.45 and 1500.00 K were also performed. Rods-in calculations corresponding to the four hot moderator densities and the four cold moderator densities were included. This resulted in a total of 24 branch states; the nominal state is duplicated in one of the branch states.

Table 7 lists all branches used in the SCALE/TRITON calculations.

Table 7. Branch calculation states for 0% void tree

Branch no.	Void Fraction (%)	Moderator density (g/cm^3)	Fuel temperature (K)	Moderator temperature (K)	Control rod state (0=out, 1=in)
1	0	0.7373	948.45	560.29	0
2	40	0.4573	948.45	560.29	0
3	70	0.2473	948.45	560.29	0
4	90	0.1073	948.45	560.29	0
5	0	0.7373	948.45	560.29	1
6	40	0.4573	948.45	560.29	1
7	70	0.2473	948.45	560.29	1
8	90	0.1073	948.45	560.29	1
9	0	0.7373	500.00	560.29	0
10	40	0.4573	500.00	560.29	0
11	70	0.2473	500.00	560.29	0
12	90	0.1073	500.00	560.29	0
13	0	0.7373	1500.00	560.29	0
14	40	0.4573	1500.00	560.29	0
15	70	0.2473	1500.00	560.29	0
16	90	0.1073	1500.00	560.29	0
17	0	0.9982	293.15	293.15	0
18	0	0.9922	313.15	313.15	0
19	0	0.9837	333.15	333.15	0
20	0	0.9718	353.15	353.15	0
21	0	0.9982	293.15	293.15	1
22	0	0.9922	313.15	313.15	1
23	0	0.9837	333.15	333.15	1
24	0	0.9718	353.15	353.15	1

9. BURNUP STEPS

As previously mentioned, a maximum exposure of ~60 GWd/MTU was chosen. Based on this maximum burnup, a depletion step scheme was developed based on a parabolic approximation of the depletion of gadolinium (^{155}Gd and ^{157}Gd). I.e., the depletion step size increases parabolically following the depletion of gadolinium. Depletion steps start at ~0.3 GWd/MTU and increase up to 3 GWd/MTU (after peak reactivity). The depletion points utilized in the TRITON calculations can be found in Table 8.

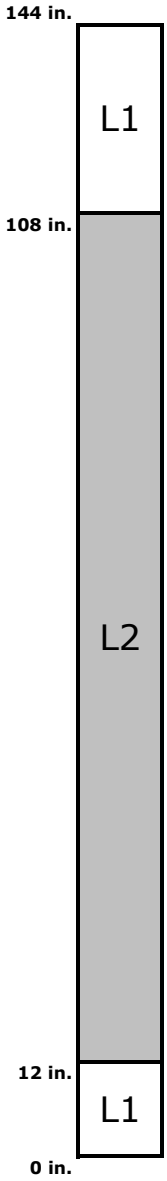
Table 8. Burnup steps

Time (days)	Burnup (GWd/MTU)	Time (days)	Burnup (GWd/MTU)	Time (days)	Burnup (GWd/MTU)
0.00	0.00	245.95	5.78	643.07	15.11
0.01	0.00	260.43	6.12	677.07	15.91
6.69	0.16	275.19	6.47	715.04	16.80
19.97	0.47	290.23	6.82	758.43	17.82
33.12	0.78	305.60	7.18	809.79	19.03
46.17	1.08	321.32	7.55	874.31	20.55
59.15	1.39	337.43	7.93	966.13	22.70
72.09	1.69	353.97	8.32	1085.34	25.51
85.02	2.00	370.98	8.72	1213.00	28.51
97.95	2.30	388.52	9.13	1340.66	31.51
110.90	2.61	406.64	9.56	1468.32	34.51
123.90	2.91	425.41	10.00	1595.98	37.51
136.96	3.22	444.90	10.46	1723.64	40.51
150.10	3.53	465.20	10.93	1851.30	43.51
163.34	3.84	486.42	11.43	1978.95	46.51
176.70	4.15	508.70	11.95	2106.61	49.51
190.20	4.47	532.17	12.51	2234.27	52.51
203.85	4.79	557.06	13.09	2361.93	55.51
217.68	5.12	583.60	13.71	2489.59	58.51
231.71	5.45	612.12	14.38	2617.25	61.51

10. LATTICE DESIGNS

Based on the information in the design documents [5,6,14], five fuel assembly designs were identified for this project. The five fuel assembly designs are simply named types 1-5, as in the Hatch documentation. The fuel lattice layouts provided in this section specify the composition of fuel pins in the lattice and the axial layout of those lattices in the fuel assembly. Each box in the figures corresponds to a fuel pin location (or water rod location). A box containing a single number corresponds to the fuel enrichment for that pin location. If the box contains two numbers, the top number corresponds to the fuel enrichment and the bottom number corresponds to the gadolinia content in weight percent. A box that contains the letters "WR" corresponds to a water rod. In each of the fuel assembly layouts in this section, the axial regions are shown on the left, the lattice layouts near the middle of the figure with a summary table of the fuel pins, and the SCALE/TRITON representation of the lattice is shown on the right.

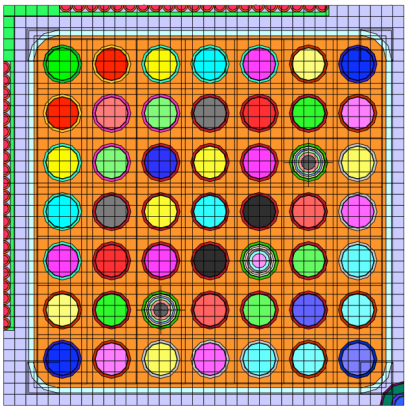
The type 1 fuel assembly contains three axial regions. However, the top and bottom axial regions are identical, so there are two unique axial regions. The lattice layout for the type 1 assembly can be found in Figure 3.



Type 1, L1

1.27	1.27	1.69	1.69	1.69	1.94	1.69
1.27	1.94	1.94	2.79	2.79	2.79	1.94
1.69	1.94	2.79	2.79	2.79	2.79 4.0	2.79
1.69	2.79	2.79	2.79	2.79	2.79	2.79
1.69	2.79	2.79	2.79	2.79 4.0	2.79	2.79
1.94	2.79	2.79 4.0	2.79	2.79	2.79	1.94
1.69	1.94	2.79	2.79	2.79	1.94	1.94

Enerichment (wt% ^{235}U)	Gd_2O_3 (wt%)	No. of Rods
2.79	--	25
1.94	--	10
1.69	--	8
1.27	--	3
2.79	4.0	3



Type 1, L2

1.27	1.27	1.69	1.69	1.69	1.94	1.69
1.27	1.94	1.94	2.79	2.79	2.79	1.94
1.69	1.94	2.79 4.0	2.79	2.79	2.79 4.0	2.79
1.69	2.79	2.79	2.79	2.79	2.79	2.79
1.69	2.79	2.79	2.79	2.79 4.0	2.79	2.79
1.94	2.79	2.79 4.0	2.79	2.79	2.79	1.94
1.69	1.94	2.79	2.79	2.79	1.94	1.94

Enerichment (wt% ^{235}U)	Gd_2O_3 (wt%)	No. of Rods
2.79	--	24
1.94	--	10
1.69	--	8
1.27	--	3
2.79	4.0	4

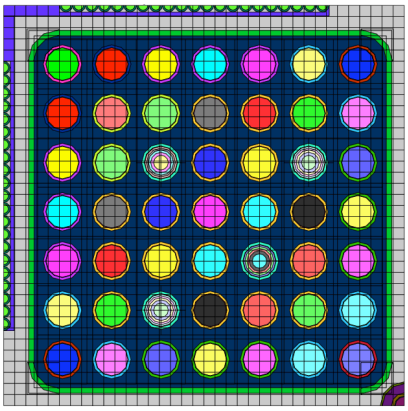


Figure 3: Lattice layout for the type 1 fuel assembly.

Assembly type 2 contains five axial regions, but two regions are identical to other regions in the assembly, so there are only three unique axial regions. In addition, two of these axial regions (lattices) are identical to lattices in the type 1 fuel assembly, so only one additional SCALE/TRITON model is required for this case. The lattice layout for the type 2 fuel assembly can be found in Figure 4.

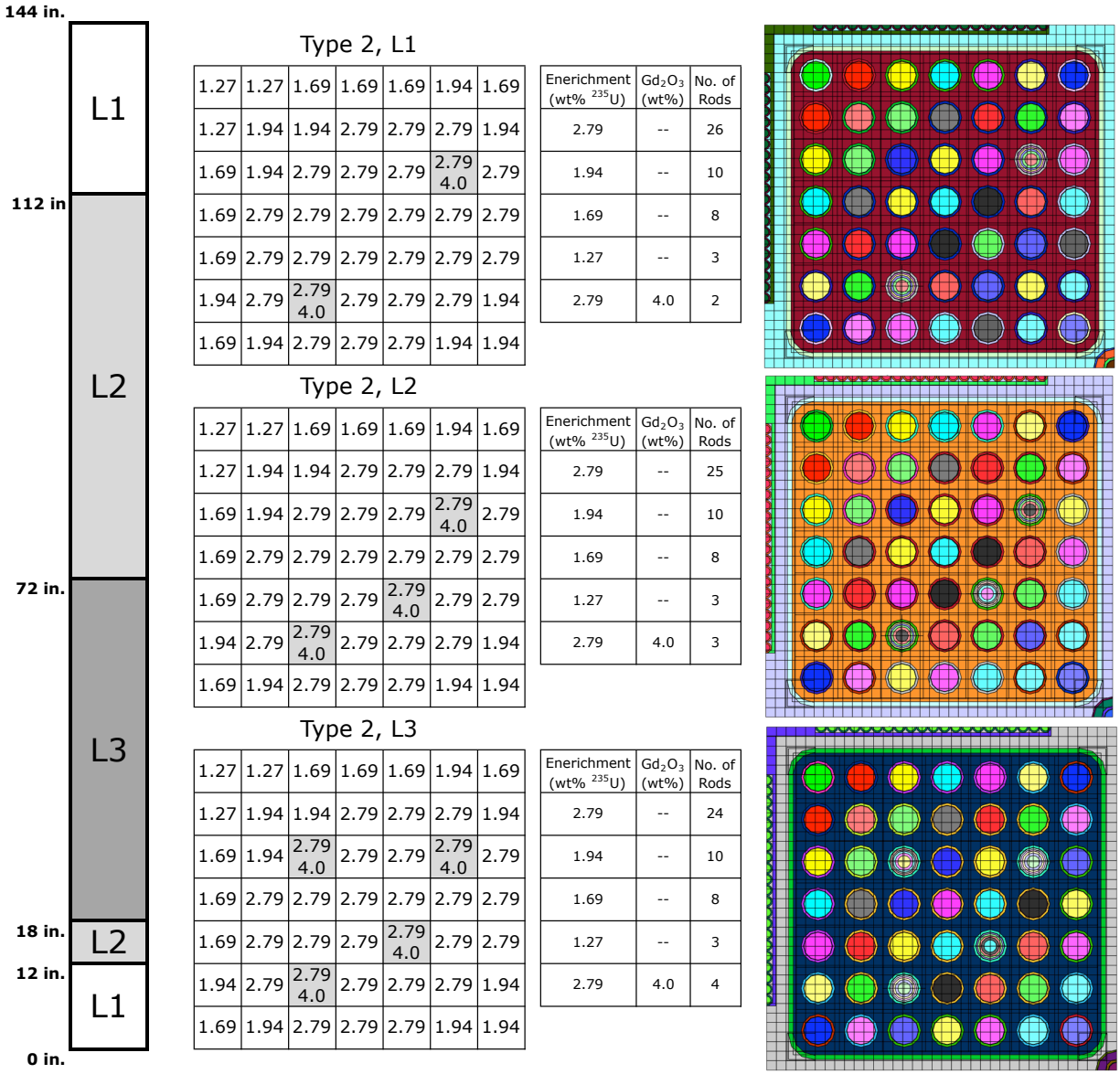
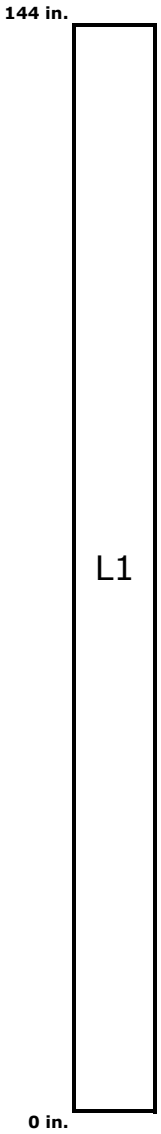


Figure 4: Lattice layout for the type 2 fuel assembly.

Assembly type 3 contains only a single axial region. The lattice layout for assembly type 3 can be found in Figure 5.



Type 3, L1

1.27	1.27	1.69	1.69	1.69	1.94	1.69
1.27	1.94	1.94	2.79	2.79	2.79	1.94
1.69	1.94	2.79	2.79	2.79	2.79 2.5	2.79
1.69	2.79	2.79	2.79	2.79	2.79	2.79
1.69	2.79	2.79	2.79	2.79	2.79	2.79
1.94	2.79	2.79 2.5	2.79	2.79	2.79	1.94
1.69	1.94	2.79	2.79	2.79	1.94	1.94

Enerichment (wt% ^{235}U)	Gd_2O_3 (wt%)	No. of Rods
2.79	--	26
1.94	--	10
1.69	--	8
1.27	--	3
2.79	2.5	2

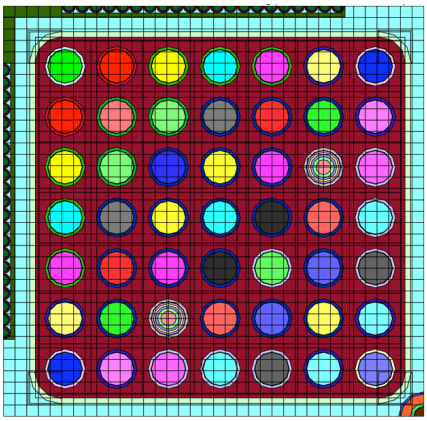


Figure 5: Lattice layout for the type 3 fuel assembly.

Assembly type 4 is an 8x8 design, rather than the 7x7 design used in assembly types 1-3. Like assembly type 3, assembly type 4 contains only one axial level. In addition, assembly type 4 contains a single water rod that occupies the location of a single fuel pin. The lattice layout for assembly type 4 can be found in Figure 6.

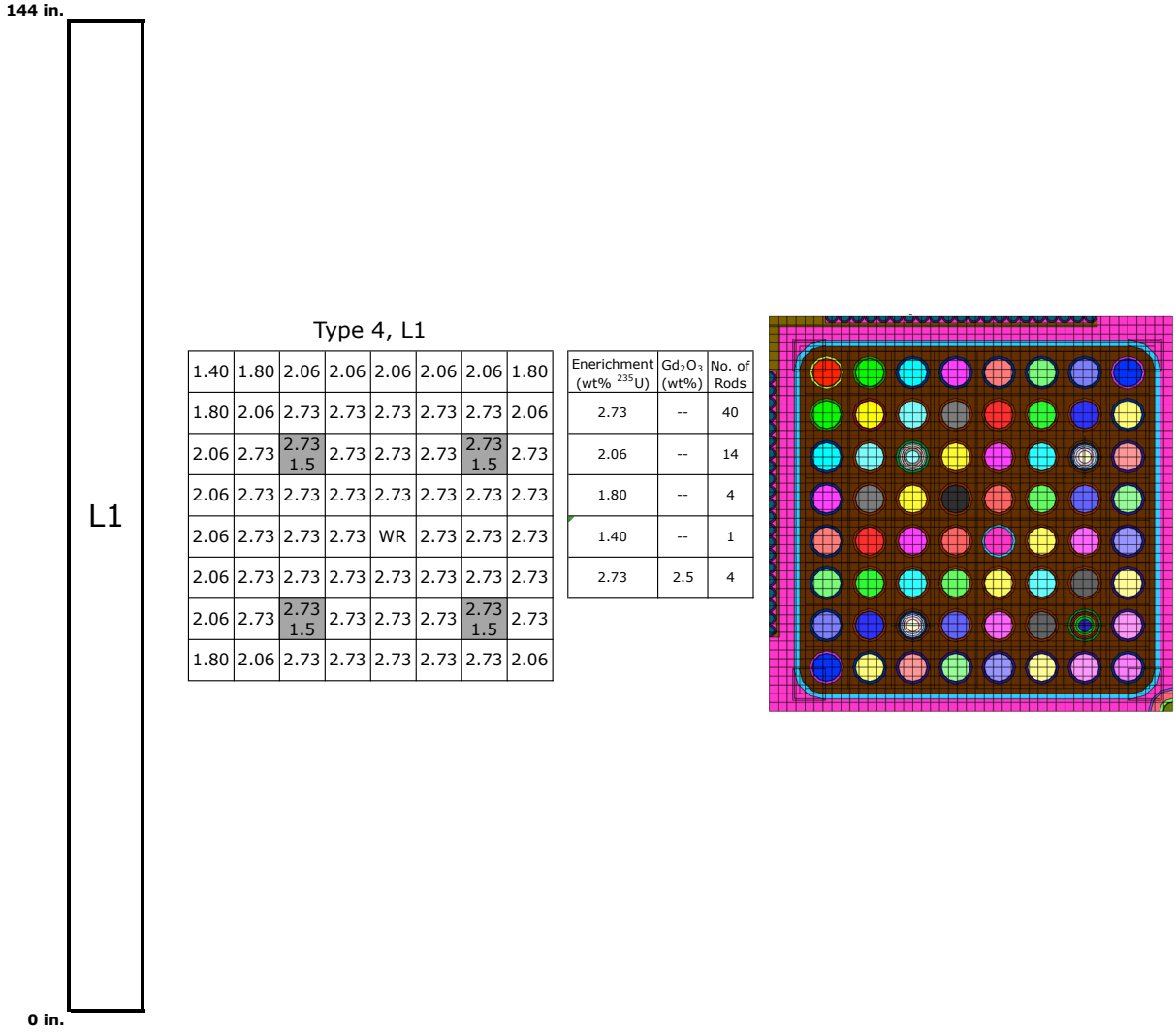


Figure 6: Lattice layout for the type 4 fuel assembly.

Assembly type 5 is an 8x8 fuel assembly, similar to the type 4 fuel assembly. However, assembly type 5 contains two water rods that displace a single fuel pin each. In addition, the top and bottom lattice in the type 5 fuel assembly contains natural uranium fuel pins. The lattice layout for assembly type 5 can be found in Figure 7.

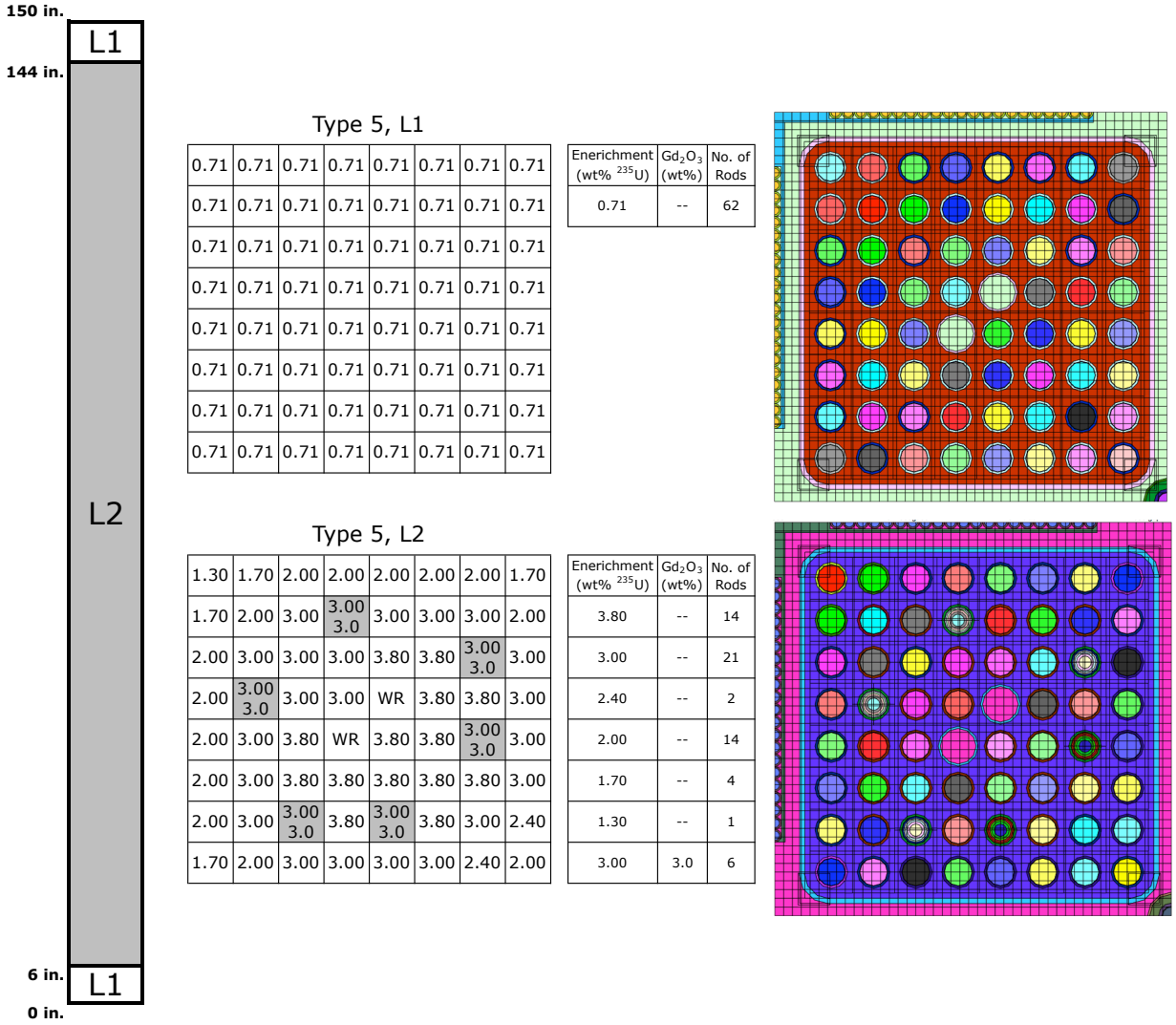


Figure 7: Lattice layout for the type 5 fuel assembly.

The input file naming convention for this project used the following format:

hatch.tN.lL.XXV.crC.inp,

where

N is the assembly name (*t1*, *t2*, etc.);
L is the lattice name (*l1*, *l2*, etc.);
XX is the void fraction (*00*, *40*, or *70*); and
C is the control rod state (*0*/*1* for out/in).

However, there is an exception to this naming convention. As previously stated, the type 4 fuel assembly contains a single axial level, but at the time of model development, it was believed that the type 4 fuel assembly contained a top and bottom axial region of natural uranium, like assembly type 5. This is not the case, but the fuel lattice naming convention was used throughout, so the SCALE/TRITON input files for the type 4 fuel assembly all use "*l2*" for lattice two, rather than "*l1*".

11. REFLECTOR MODELING

In previous projects, reflector nodal data were generated for a specific fuel and reactor design. In this project, given previous knowledge of other types of fuel and reactor types, ORNL was tasked with developing general reflector data, that could be applied to a number of different fuel assemblies and reactor types. A fuel assembly is used in the reflector calculation only to provide a representative neutron source spectrum for a particular reactor type, so the reflector nodal data will be relatively insensitive to the fuel assembly used in the calculation. However, the reflector calculations are more sensitive to the size of the regions modeled for the reflector and the materials that are used in these regions.

In previous work [7-11] a number of assumptions for the size of radial and axial reflector regions have been used. A common practice is to utilize a region that is the size of a single fuel assembly. Some previous work utilized a region for the radial reflector that terminates at the inner surface of the vessel wall [7]. In this work, using a model originally developed for Ref. 7, a sensitivity calculation for the size of the radial reflector was performed. It was found that the although XS data were fairly insensitive the to the size of the radial reflector region, the assembly discontinuity factors (ADFs) did show some sensitivity to the size of the region. The current work uses a reflector region that is as large as reasonable for each reflector (radial, top, and bottom).

11.1 Radial Reflector

It is not clear that the Hatch Unit 1 reactor contains a baffle and vessel that is similar to current-design BWRs, and it is difficult to tell the exact dimensions of the baffle and vessel from the Hatch documentation. For this reason, the same geometric model for the radial reflector as used in Ref. 7 was applied for this case, with the exception of the neutron source (fuel assembly). A mid-level fuel lattice (type 1, lattice 2) was used as a neutron source for these calculations. ORNL staff believe these assumptions are generally acceptable for many types of current generation and near-term future BWRs. The radial reflector is composed of three regions: a region of water, followed by a stainless steel region corresponding to the core baffle, followed by another region of water that terminates at the inner surface of the vessel. For most of the T-H states modeled, the density of the water inside the baffle is modified corresponding to a saturated two-phase mixture, but the density of the water outside the baffle remains saturated liquid. In the low-temperature states (293-353 K), both the water inside and outside the baffle are modified corresponding to that temperature. The SCALE/TRITON representation of the radial reflector model can be found in Figure 8.

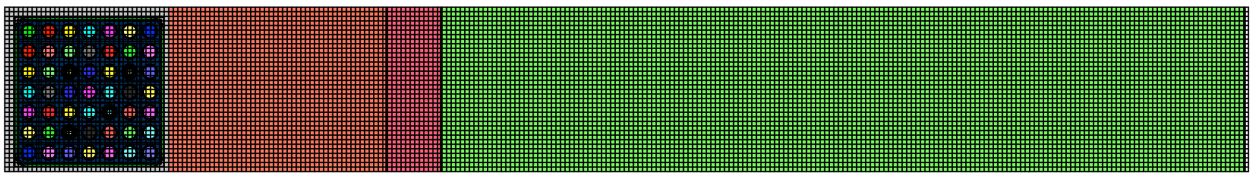


Figure 8: SCALE/TRITON representation of the radial reflector model.

11.2 Bottom Axial Reflector

In previous projects, a homogenized region the size of a single assembly was used to generate data for the bottom axial reflector. However, the Hatch design documents contain detailed information that is particularly useful for modeling the bottom axial reflector region in more detail. The axial length, materials, and associated masses of the fuel rod end plugs, lower tie plate, and core support structure are given in the documentation. The materials below the fuel assembly have been broken down into two regions: a region that contains the fuel rod end plugs and lower tie plate, and a region that contains the core support structure. In addition, it has been assumed that the control blade is always present in the bottom axial reflector below the fuel assembly. As such, control blade materials have been homogenized in both bottom axial reflector regions. For the bottom axial reflector, a type 5, lattice 1 model was used as the neutron source. This lattice is a natural-uranium fueled lattice, which is common in modern fuel assemblies.

The first reflector region contains the fuel rod end plugs (Zircaloy-2), the lower tie plate (SS304 and Inconel), $1/4^{th}$ of a control blade, with water filling the remaining volume. The length of this region is assumed to be 5 inches, as noted in Figure 13 of Ref. 5. The masses of the end plugs and tie plate are given on a per assembly basis, and using the SCALE default densities, a volume for each material is calculated. The control blade volumes were also calculated and added to the homogenized mixture. The volumes of each material and the total volume of the region were used to calculate the volume fractions of the materials, which were used in the SCALE material definition.

The second region contains the core support structure and control blade materials, with the remaining volume filled with water. The Hatch design documentation gives the approximate mass of the core support structure (17,500 lbs.), the length of this region (3.74 in.), and the assumed diameter of the core support structure (172.5 in., Figure 13 of Ref. 5), which were used to approximate the volume of support structural material (assumed SS304) per assembly. Similar to the first region, volumes of the core support structure, the control blade, and water were used to construct the homogenized mixture for the second region. The SCALE/TRITON representation of the bottom axial reflector model can be found in Figure 9.

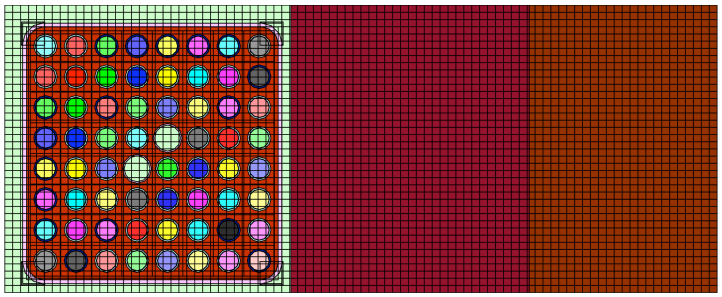


Figure 9: SCALE/TRITON representation of the bottom axial reflector model.

11.3 Top Axial Reflector

The top axial reflector has been modeled in a similar fashion as the bottom axial reflector. Like the bottom axial reflector, the top axial reflector contains two different regions consisting of different homogenized materials. Like the bottom axial reflector model, type 5, lattice 1 was chosen as the neutron source, which is a natural uranium lattice. Again, it is common to have a natural uranium lattice for the top and bottom top of the active fuel. Unlike the bottom axial reflector, it has been assumed that this region will never experience the effect of a control blade, so the volume of those materials have been assumed to be water.

The first region, which corresponds to the fuel plenum region, is a homogenized mixture of a fuel assembly that is devoid of fuel. In place of the fuel, the fuel getter (Zircaloy), plenum spring (SS304), and helium are homogenized. The masses of the getter and plenum spring are given in the Hatch documentation. The length of the fuel plenum region has been assumed to be 12 inches, which was chosen following a survey of current and previously analyzed fuel assemblies. Using the volumes of each material and the total volume of the region, a homogenized mixture was constructed using the volume fractions of the various materials in the mixture.

The second region consists of the upper tie plate material (SS304), the upper fuel end plugs (Zircaloy-2), with the remaining volume occupied by water. The size of this region corresponds to the size of the upper tie plate (6.65 in.) given in Figure 6 of Ref. 5. The SCALE/TRITON representation of the top axial reflector can be found in Figure 10.

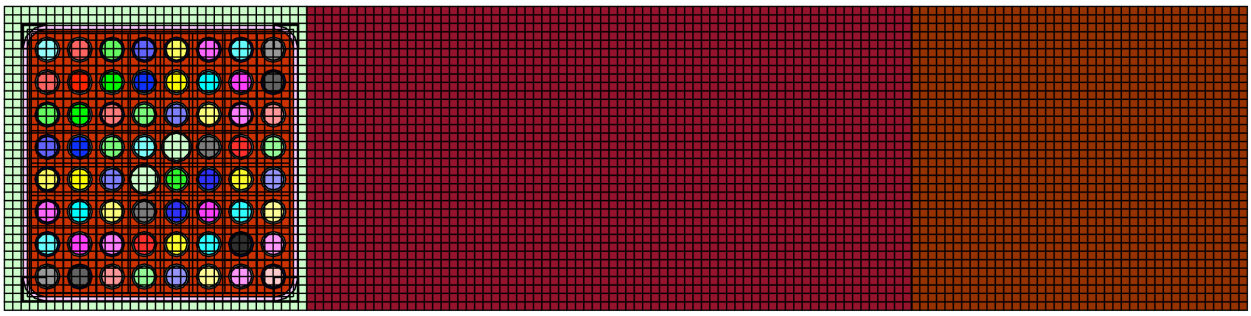


Figure 10: SCALE/TRITON representation of the bottom axial reflector model.

11.4 Reflector Thermal-Hydraulic States

Similar to the lattice calculations, in order to provide nodal data at various T-H states, branch calculations are required. In the current work, branch calculations were not performed in a single TRITON job (as is done in the lattice calculations), rather, many different single state point calculations were performed at different T-H conditions. The data from each of these separate calculations was converted to PMAXS format, then assembled to generate the final PMAXS file. Each of the reflectors (radial, top, and bottom) use the same branch states. Similar to the lattice data, calculations were performed for moderator densities corresponding to 0%, 40%, 70%, and 90% void fraction. In addition, as in the lattice calculations, 0% void cold conditions corresponding to 293, 313, 333, and 353 K were also modeled. Because the reflector data will likely be used outside of the current project, borated conditions were also added corresponding to 600 and 1500 ppm natural boron.

12. DETECTOR MODELING

As part of data generation for this project, traveling in-core probe (TIP) data were requested. The Hatch documentation contains a detailed description of the local power range monitor (LPRM) tube, TIP detector, and other associated materials (Figure 16 of Ref. 5). In previous work [12], the TIP detector had been modeled as a single cylindrical region of water mixed with a trace amount of fissile ^{235}U, which was the size of the LPRM tube. In this project, a more detailed representation of the LRPM and TIP detector was used. Five concentric cylindrical regions were used to represent the TIP detector and LPRM tube and associated materials (southeast corner of Figure 11). The innermost region contains ^{235}U and Argon (typical fill gas for fission chambers [21]). The second region is a thin layer of water that acts as a coolant for the TIP detector. The third region is the TIP detector tube composed of SS304. The actual LPRM detectors make up a relatively small axial length of the LPRM tube, however the LPRM cables are present throughout the axial length of the LPRM tube. In order to account for the LPRM cables in the fourth region, the material of the LPRM cables (SS304 and Al_2O_3) and water were homogenized. The fifth and final region is the LPRM tube composed of SS304.

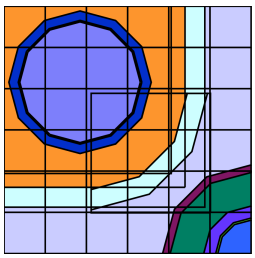


Figure 11: SCALE/TRITON representation of the LRPM tube and TIP detector model.

The detector is modeled in all fuel lattice models, although it is physically only present for ~1/16 of the fuel assemblies in the core. Modeling the detector in TRITON does displace a small amount of bypass water, which is not expected to introduce a significant bias in the nodal data.

In TRITON, unlike some other lattice physics tools, the user must physically construct a region containing ^{235}U to represent the active portion of the detector, then specify this mixture in the TRITON parameters. During the transport calculation, the flux and ^{235}U fission cross-section is

collapsed to 2 groups and written in the *xfile016* file. GenPMAXS then reads the flux and fission cross sections and calculates the detector response, which is printed in the PMAXS data files.

13. CROSS-SECTION CALCULATION PROCEDURE

The process to generate BWR assembly cross sections with SCALE/TRITON for use in TRACE/PARCS calculations is typically performed in four steps for each assembly: (1) generation of pin-by-pin Dancoff factors, (2) generation of 49-group libraries, (3) depletion calculations with branch states, and (4) conversion of *xfile016* files to PMAXS format.

Prior to performing production calculations, lattice models were developed using the aforementioned specifications. Due to the heterogeneity of BWR fuel assemblies, special care was taken to ensure that the cross-section self-shielding methodology for each fuel pin was adequate. Similar to previous BWR cross-section generation projects [7-9,11,12], specialized Dancoff factors calculated using MCDancoff were applied to the corner and edge fuel pins to account for increased moderation of neutrons due to the proximity of the pins to saturated liquid water features (bypass). The type 4 and type 5 fuel assemblies contain saturated liquid water rod(s) near the center of the fuel assembly. The single water rod in the type 4 fuel assembly does not have a significant impact on the Dancoff factor for the neighboring fuel pins. However, assembly type 5 contains two smaller water rods, whose combined effect does have an appreciable impact on the Dancoff factor for fuel pins that are directly adjacent to both water rods. The Dancoff factor for these particular pins is on the same order as the edge fuel pins, so these fuel pins that are adjacent to both water rods have been given an Dancoff factor that is typical for an edge fuel pin.

The Dancoff factors from the MCDancoff calculations are applied in the TRITON model during the CENTRM cross-section processing step by using the "dan2pitch" option and are modified by SCALE/TRITON in the branch calculations to predetermined values corresponding to the Dancoff factor for a certain void fraction.

After model development was complete, the next step was to generate problem-dependent 49-group cross-section libraries from the 238-group master cross-section library. The depletion-ready input files were modified to two-step depletion input files (remaining depletion steps and all branches were removed) and the TRITON "parm=weight" option was added to the input files to generate the problem-dependent 49-group libraries. A unique problem-dependent library was generated for each input file.

The depletion calculations with branch cases were then performed in parallel mode with SCALE/TRITON. The SCALE/TRITON depletion methodology is discussed in depth in the SCALE manual. The TRITON depletion calculations were run using SCALE 6.1 on ORNL compute cluster *cpile2*.

In the final step, the cross-section files created by TRITON (*xfile016* files) for all lattice calculations were converted to PARCS format using the GenPMAXS (version 6, obtained March 2012) utility developed and maintained at the University of Michigan. Due to the method used to build the SCALE/TRITON input files for the reflector cases, a different version of GenPMAXS

(version 6, obtained November 2012) was used for conversion of the reflector nodal data to PMAXS format. In the reflector input files for the current project, the reflector data were printed as the first block of data in the *xfile016* files, rather than the second (which is what GenPMAXS expects). This is due to the way these particular SCALE/TRITON input files were constructed. The issue was communicated to UM staff who provided a version of GenPMAXS that reads the reflector data in the format generated for this particular project. Under NRC JCN F6028, ORNL and UM staff are working on a more robust way to ensure that the proper cross sections for the reflector are chosen by GenPMAXS, among other TRITON/GenPMAXS consistency efforts. The PMAXS files for the reflectors were verified by checking data in the PMAXS files with data in the TRITON *txtfile16* files.

14. RESULTS

NRC staff were provided with PARCS cross sections in PMAXS format via a zip file uploaded to the ORNL file upload server on November 14, 2012. In addition, a spreadsheet that can be used to plot all eigenvalue trajectories was provided.

In total, 12 PMAXS files were provided: nine PMAXS files for the fuel lattices and three PMAXS files for the reflectors. All SCALE/TRITON input files, GenPMAXS input files, and other relevant data are provided via a CD that acts as the final deliverables for project JCN V6361.

At the start of the project, NRC staff provided ORNL with PARCS core-follow input files for Hatch cycles 1-3. The PMAXS file naming used in the sample PARCS input files was used to name the PMAXS files delivered to NRC staff. The PMAXS files are as follows with a description in parenthesis (as noted in Sect. 10):

- hatch-t1a.PMAX (type 1, L1)
- hatch-t1b.PMAX (type 1, L2)
- hatch-t2a.PMAX (type 2, L1)
- hatch-t2c.PMAX (type 2, L2)
- hatch-t2d.PMAX (type 2, L3)
- hatch-t3.PMAX (type 3, L1)
- hatch-t4.PMAX (type 4, L1)
- hatch-t5a.PMAX (type 5, L1)
- hatch-t5b.PMAX (type 5, L2)
- hatch-refl-bot.PMAX (Bottom axial reflector)
- hatch-refl-rad.PMAX (Radil reflector)
- hatch-refl-top.PMAX (Top axial reflector)

Upon completion of the review and approval of the SCALE/TRITON input files, the production calculations were submitted using SCALE 6.1 on *cpile2*. On September 12, 2012, the production calculations were initiated. While calculations were running, eigenvalue trajectories were extracted from the *txtfile16* files and plotted in Excel. No major issues were identified at that point, so production calculations continued. Production calculations were completed on November 2, 2012.

After the production calculations had completed, ORNL staff extracted and plotted the eigenvalue trajectories for every branch of every lattice. Close visual inspection of the eigenvalue trajectories revealed no major inconsistencies. The eigenvalue trajectories for the nominal history states for each lattice are plotted in Appendix A.

After the calculations had completed, ORNL generated PMAXS files from the TRITON data using GenPMAXS. In the data conversion process, GenPMAXS outputs a "*.kinf" file that contains eigenvalue differences between the SCALE/TRITON transport calculation and the eigenvalue calculated using the broad-group data. A large bias between these two values would be an indication of an error in the process. All eigenvalue differences were less than 0.05 pcm, indicating proper conversion of the data from TRITON format to PMAXS format. In addition, ORNL staff spot-checked some of the data to ensure that selected cross sections and other nodal data were equal to the values in the TRITON *txtfile16* files.

After completion of data conversion using GenPMAXS, ORNL utilized the Hatch 1 cycles 1-3 PARCS models provided by NRC staff to perform limited testing. The results are typically below unity for all three cycles, and especially low for much of cycle 1 ($k_{eff} \approx 0.975$). However, NRC staff noted that the results are as expected and are comparable with results generated using other broad-group data for Hatch cycles 1-3 [22].

ORNL has completed generation of PMAXS cross sections for use in PARCS core-follow calculations of Hatch Unit 1 cycles 1-3. In addition, ORNL generated reflector nodal data that can be used as a general data set for many different BWRs operating with a wide variety of T-H conditions.

15. RECOMMENDATIONS FOR FUTURE WORK

NRC JCN V6361 progressed smoothly due to application of the lessons learned from past cross-section generation projects. There are a number practices that enabled the work to proceed as planned.

1. *Clarity of design data and frequent communication with NRC staff* – Once the design data had been obtained, ORNL and NRC staff continued weekly conversations about the project status and the project plans. ORNL staff were clear on what design data were being used and how the SCALE/TRITON models would be developed. As a result, there were no major surprises to either ORNL or NRC staff.
2. *Plotting eigenvalue trajectories as data are generated* – As with previous projects, the eigenvalue trajectories for each lattice were extracted as they were generated. Although no major issues were uncovered in this project, plotting eigenvalue trajectories enabled ORNL to catch issues prior to full data generation in previous projects.
3. *Allowing sufficient time for generation of reflector data* – Generation of reflector nodal data were more extensive in this project than in previous projects. Generating homogenized mixtures of many materials can be tedious, time-consuming, and prone to errors. In addition, large reflector models that have a neutron source at one side of the model and a vacuum boundary condition at the other side can be slow to converge. By

allotting sufficient time for reflector modeling, the modeler will be able to be more methodical about generating reflector materials, and will have sufficient time to troubleshoot any convergence issues.

REFERENCES

1. *SCALE 6.1: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design*, ORNL/TM-2005/39, Version 6.1 (June 2011). Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-00785.
2. Yunlin Xu and Thomas Downar, "GenPMAXS-V5: Code for Generating the PARCS Cross Section Interface File PMAXS", User Manual, University of Michigan, December 2009.
3. T. Downar, Y. Xu, and V. Seker. "PARCS V3.0, U.S. NRC Core Neutronics Simulator." User Manual, University of Michigan, 2010.
4. B. Collins, L. Li, S. Stimpson, D. Jabaay, A. Ward, Y. Xu, T. Downar, and D. Wang. "PATHS: a Steady State Two-Phase Thermal-hydraulic Solver for PARCS Depletion." NURETH-14, Toronto, CA, 2011.
5. N.H. Larsen and J.G. Goudey, "Core Design and Operating Data for Cycle 1 of Hatch 1", EPRI Report NP-562, Jan. 1979.
6. G.L. Holloway, J.E. Fawks, and B.W. Crawford, "Core Design and Operating Data for Cycles 2 and 3 of Hatch 1", EPRI Report NP-2106, Feb. 1984.
7. Brian J. Ade and Harold J. Smith, "Generation and Testing of Lattice Parameters for SVEA-96 Optima2 Fuel – Task 1b", Oak Ridge National Laboratory, under NRC Contract JCN V6222, ORNL/LTR-2012/172, May 2012.
8. Brian J. Ade and Harold J. Smith, "Generation and Testing of Lattice Parameters for SVEA-96 Optima2 Fuel – Task 1a", Oak Ridge National Laboratory, under NRC Contract JCN V6222, ORNL/LTR-2012/224, July 2012.
9. Brian J. Ade, "Generation and Testing of Updated Lattice Parameters for an ABWR Core", Oak Ridge National Laboratory, under NRC Contract JCN N6825, ORNL/LTR-2011/529, January 2012.
10. Matthew A. Jessee, Mark D. DeHart, and Dan Ilas, "Generation of Lattice Parameters for an ABWR Core", Oak Ridge National Laboratory, under NRC Contract NRC JCN N6825, Task 2 Letter Report, May 2010.
11. Brian J. Ade, Matthew A. Jessee, "Generation and Testing of Lattice Parameters for Maximum Extended Load Line Limit Analysis Plus for a BWR", ORNL/LTR-2011/32, November 2011.
12. Brian J. Ade et al., "Code Validation with BWR Steady-State Plant Operating Data", ORNL Letter Report, June 2011.
13. N.H. Larsen, "Core Design and Operating Data for Cycles 1 and 2 of Peach Bottom 2", EPRI NP-563, Electric Power Research Institute, June 1978.
14. P. Yarsky, "Calculation Notebook for Hatch Unit 1 PARCS/PATHS Model," August 1, 2012 (ADAMS Accession No. ML12228A528).
15. Jose March-Leuba and Dean Wang, "Cross Section Generation Guidelines for TRACE-PARCS", ORNL/TM-2012/518, NUREG/CR, to be published, 2012.
16. "Use of MCDANCOFF When Modeling Boiling Water Reactors," *SCALE Newsletter,* Number 42, 2010.
17. M. L. Williams, "Resonance Self-Shielding Methodologies in SCALE 6", *Nuclear Technology*, 174 (2), 149, 2011.
18. S. M. Bowman, O. W. Hermann, and M. C. Brady, *Sequoyah Unit 2 Cycle 3, Scale-4 Analysis of Pressurized Water Reactor Critical Configurations*, ORNL/TM-12294/V2, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1995.

19. Steam Tables Online, http://www.steamtablesonline.com/, data originally from The International Association for the Properties of Water and Steam, 2012.
20. N.E. Todreas and M.S. Kazimi, "Nuclear Systems I: Thermal Hydraulic Fundamentals", Taylor and Francis Group, New York, 1990.
21. G.F. Knoll, Radiation Detection and Measurement, 2nd ed., John Wiley and Sons, 1989.
22. P. Yarsky, "Assessment of HELIOS and PARCS V3.2M06 Against Hatch Unit 1 Data," August 24, 2012 (ADAMS Accession No. ML12250A758).

APPENDIX A. Selected Eigenvalue Trajectories

In this section, eigenvalue trajectories for each history state of each lattice have been plotted.

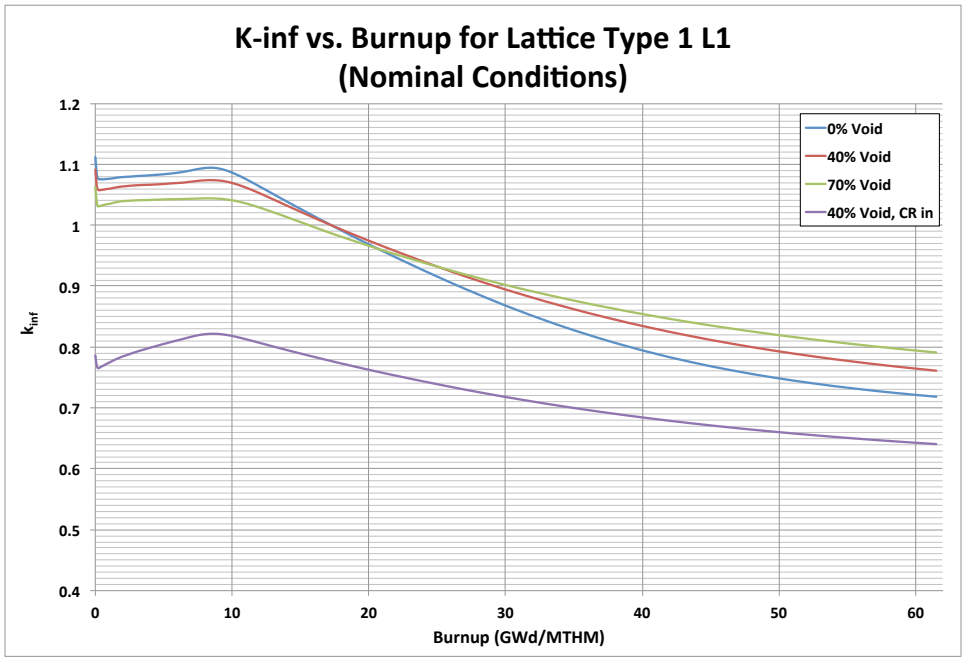


Figure 1: Assembly type 1, lattice 1 eigenvalue trajectory for the history states.

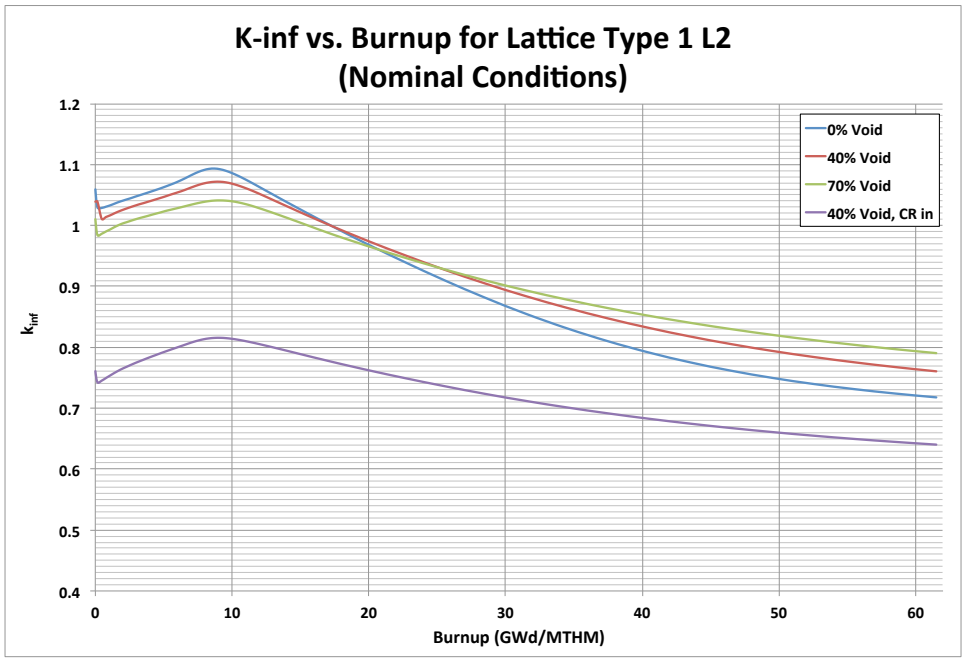


Figure 2: Assembly type 1, lattice 2 eigenvalue trajectory for the history states.

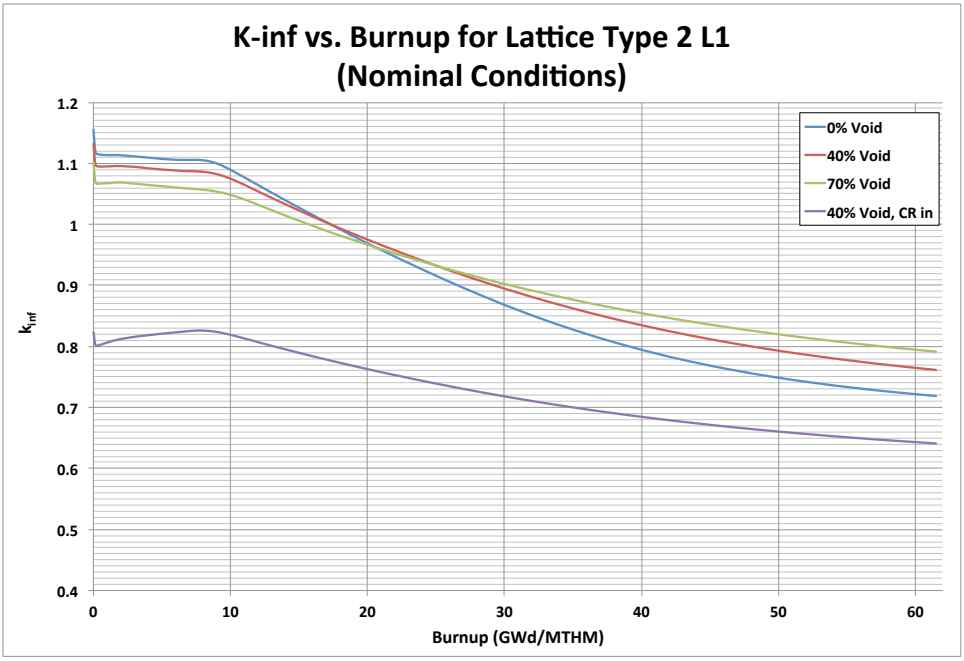


Figure 3: Assembly type 2, lattice 1 eigenvalue trajectory for the history states.

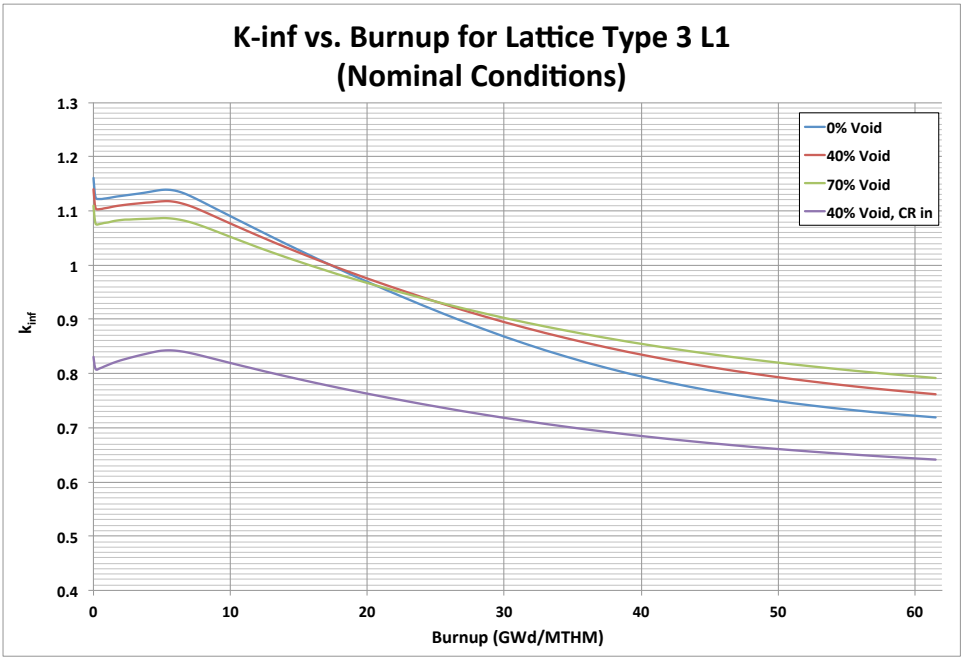


Figure 4: Assembly type 3, lattice 1 eigenvalue trajectory for the history states.

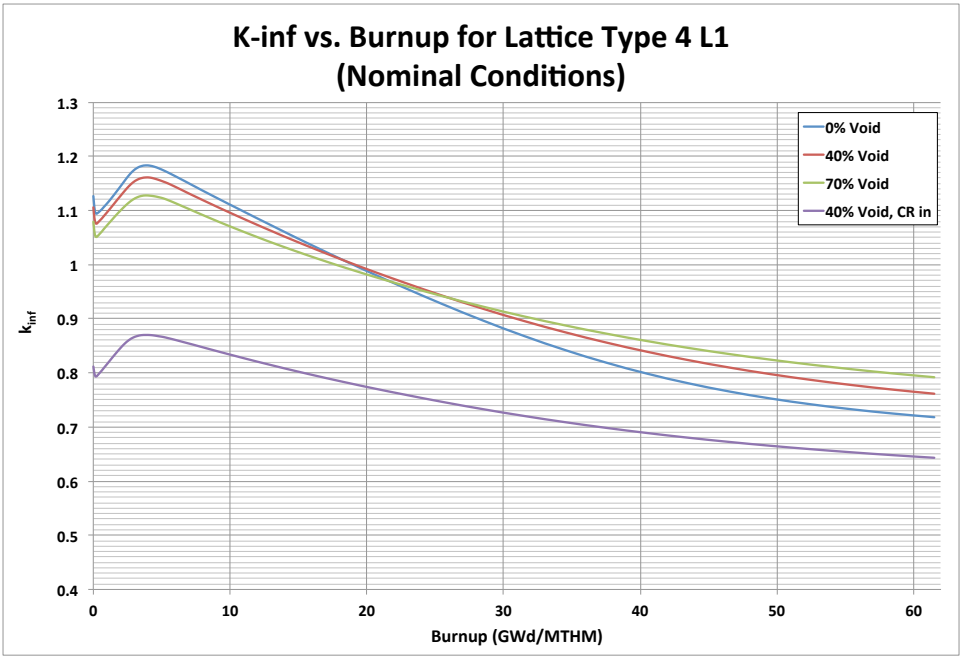


Figure 5: Assembly type 4, lattice 1 eigenvalue trajectory for the history states.

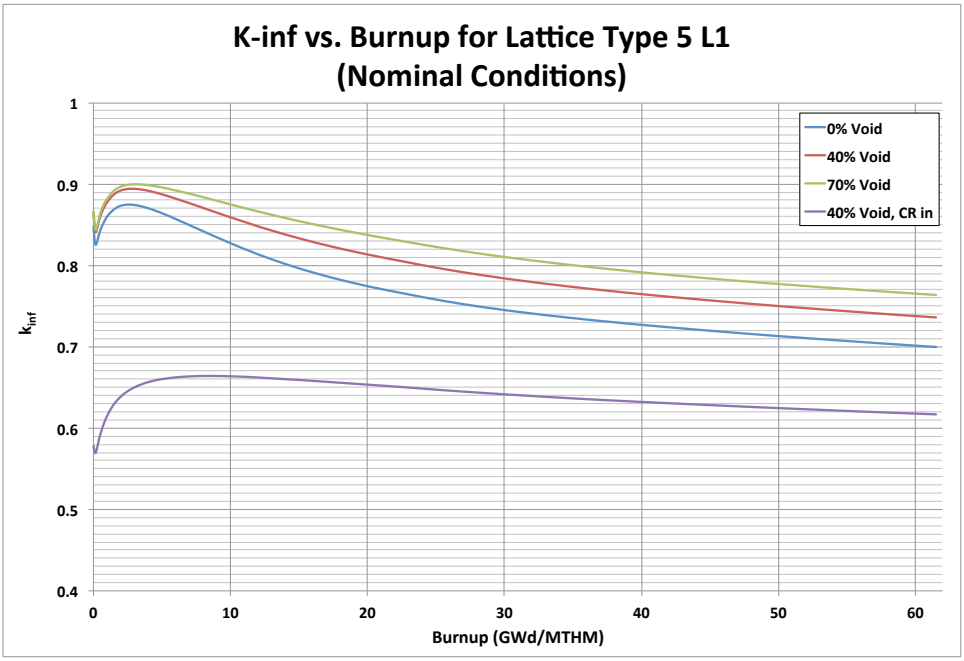


Figure 6: Assembly type 5, lattice 1 eigenvalue trajectory for the history states.

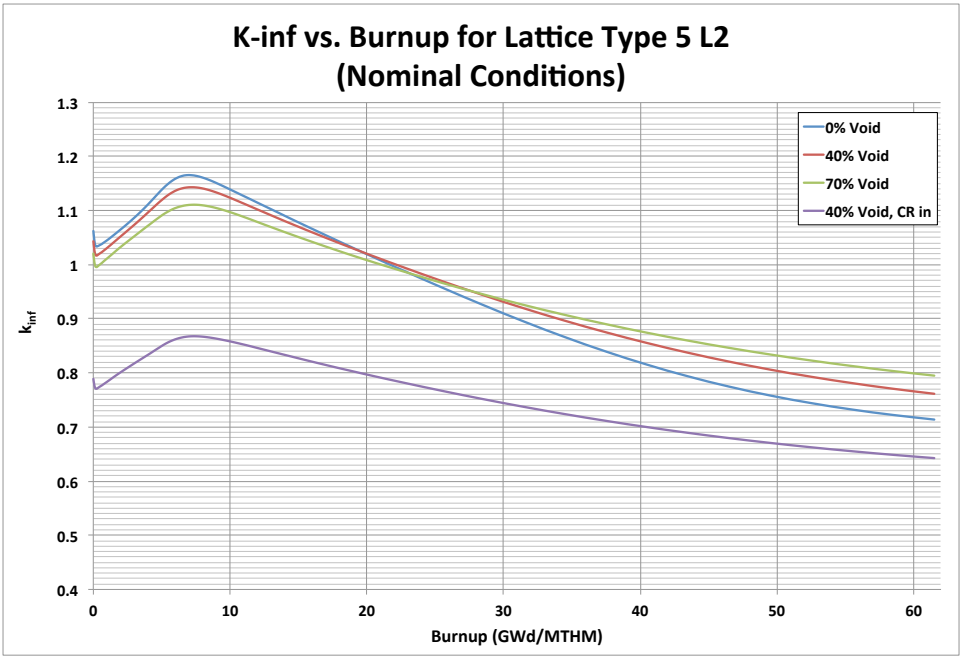


Figure 7: Assembly type 5, lattice 2 eigenvalue trajectory for the history states.