

Informal Documentation

**Cross Section Libraries and Preliminary Modeling
for the Reference UMLRR LEU Core Configuration**

**Dr. John R. White
Chemical and Nuclear Engineering Department
University of Massachusetts Lowell**

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Introduction

A series of studies over the last few years have been performed to establish a set of computational physics models for the UMass-Lowell Research Reactor (UMLRR).¹ These models focussed on the current high enriched uranium (HEU) core geometry with a variety of 2-D XY configurations suitable for use with the VENTURE and DORT codes.²⁻³ These finite difference codes use the multigroup diffusion theory approximation (VENTURE) and discrete ordinates transport theory techniques (DORT) to solve the appropriate particle balance equations over the space and energy (and angular) domains of interest. VENTURE is used primarily to give criticality information and few group fluxes and power density distributions within the core region. DORT, in contrast, is used mainly to characterize the multigroup neutron and gamma radiation environment in the core and excore regions, with a focus on the experimental irradiation locations in the UMLRR. Additional analyses with the DORT-generated neutron and gamma fluxes and material-specific kerma factors can also give energy deposition estimates in various locations and experimental arrangements. All these recent analyses have addressed the UMLRR HEU core configuration in 2-D XY geometry.

With the pending conversion (hopefully in late Summer 1999) to low enriched uranium (LEU) fuel assemblies and a completely new arrangement for the fuel, graphite, and radiation basket assemblies, it becomes essential that a similar set of computational models and overall analytical capability be made available for the expected LEU configuration within the UMLRR. A first step towards this goal is the development of appropriate multigroup cross section libraries for use in subsequent VENTURE and DORT analyses for various LEU core models. In particular, the development of an application-specific 2-group library for use in VENTURE and a coupled 47 neutron group and 20 gamma group library for use with DORT was the real objective of the current project. In addition, the development of a preliminary VENTURE XY model that uses the new 2-group library and establishes the library's credibility for use in future studies was also a specific goal of the current work.

This report documents these two tasks:

- I. The development of two LEU-specific libraries for use in VENTURE and DORT, and
- II. The use of the VENTURE-compatible 2-group library for preliminary verification of the overall cross section generation procedure.

Both subjects are treated in some detail since they will serve as the basis for much of the subsequent computational analyses that will be performed for the new LEU core in the future.

The methods and models documented here for the LEU core are very similar to those developed in previous studies for the HEU configuration.¹ Much of the actual documentation

also follows portions of Refs. 1 quite closely, but it is reproduced here with specificity to the LEU core for completeness. It is expected that this report will serve as the sole reference for the base LEU cross section development and as a reference to the first iteration in the development of computational models that will be used in future analytical studies of the LEU-fueled UMass-Lowell Research Reactor (UMLRR). Thus, a complete account of this brief study is important for future reference, and a little redundancy with previous documentation is justifiable.

This report is broken into two relatively separate components that focus on the two main tasks of this study. The first part highlights the procedures and models associated with the LEU cross section generation process, and the second part overviews the development and testing of a preliminary VENTURE XY model of the reference LEU core for the UMLRR. Combined, these two mini-documents should serve as a foundation for future modeling efforts for the UMLRR LEU core.

Development of the LEU Cross Section Libraries

Computational Procedure for the LEU Cross Sections

Generating effective group constants for performing reactor physics computations is a complicated process. Most 2-D and 3-D core and shielding calculations use the multigroup approximation and they use homogenized spatial regions to describe the real heterogeneous geometric configuration of interest. The “effective” nuclear data used in such calculations must account for both the fine-group resonance effects and the real heterogeneous geometry of the system. This is achieved by performing *resonance calculations* to treat the spectral self shielding caused by large variations in the resonance cross sections and by performing *assembly or cell calculations* to account for the spatial self shielding associated with heterogeneous geometries. In the present work, these computations are accomplished with the BONAMI and XSDRN modules of the SCALE 4.3 system.⁴ Other modules (such as AJAX, NITAWL, WAX, etc.) are also used to select appropriate isotopes from the main ENDF/B-VI VITAMIN-B6 199/42 group coupled library,⁵ to convert the cross section format to a form compatible with the different codes, and to merge individual isotopes into a final library for use in subsequent 1-D and 2-D core calculations.

The specific computations for the generation of the UMLRR LEU cross sections required a set of the three assembly calculations for the primary core elements -- one each for the full fuel assembly, partial fuel assembly, and control blade. These sequences treat the resonance self shielding for the specific isotope densities and geometries present in these elements, and they approximate the spatial self shielding with a detailed heterogeneous 1-D model of a unit assembly. The resultant effective cross sections, still at the 199/42 group level, are stored in the working library format⁴ for subsequent use in XSDRN (with dataset names **fassy.lib**, **passy.lib**, and **cntl.lib**).

An infinite medium homogeneous model for the materials outside the fuel and control regions (water, aluminum, graphite, etc.) is also required. This calculation does not require an “assembly model” in XSDRN because the materials are actually contained in large homogeneous regions. Thus, the spatial self shielding that is associated with heterogeneous geometries is not of concern here. The resonance self shielding could still be important (for some isotopes) so the BONAMI step is still needed. The NITAWL code is also used here to convert the master

library⁴ output from BONAMI into the working library format for subsequent use in XSDRN (saved with file name **homo.lib**).

The cross sections in the four individual libraries noted above were combined into a single working library using the WAX module of the SCALE 4.3 system.⁴ This working library, referred to as **leuxs.worlib**, still has the original fine-group representation (199 neutron groups and 42 gamma groups). This level of energy detail is suitable for 1-D core calculations, but it has too much fine structure for reasonable 2-D computational times (even with today's fast computers). Thus, a final XSDRN calculation, using a generic 1-D model of the reactor, was performed to determine a typical fine-group weighting function and to collapse the fine-group library to the broad-group level for subsequent multidimensional calculations. This step was performed twice -- once to give a 2-group library for relatively simple VENTURE diffusion theory calculations (used for computing k_{eff} and power distributions) and a second time to create a 47/20 group coupled neutron-gamma library for DORT transport theory analyses (typically used to determine neutron and gamma spectra in 2-D configurations using a fixed source determined from a diffusion theory run). The cross sections output from this final XSDRN step, after a little additional post processing, are stored in the **leu2gxs.asnlib** and **leu67gxs.asnlib** ANISN-formatted libraries. These two libraries can be used for VENTURE and DORT computations, respectively.

In summary, this work generated three nuclear data libraries that can be utilized in subsequent UMLRR LEU core analyses, as follows:

- LEUXS.WORLIB** - 199/42 group working library for use with 1-D XSDRN core models⁴
- LEU2GXS.ASNLIB** - 2 group data in ANISN format for VENTURE models²
- LEU67GXS.ASNLIB** - 47/20 group data in ANISN format for DORT calculations³

The LEU2GXS dataset is utilized in the second part of this study, where a preliminary VENTURE model is discussed and used to show that the above procedure produces a reasonable set of data that gives consistent and expected results for the reference LEU core configuration (see below for more details). The LEU67GXS library has not been tested yet, but it is also expected to give reasonable performance within DORT.

The above discussion gives a good overview of the processing steps required to produce the two application-specific libraries for subsequent use with the VENTURE and DORT codes. The remainder of this portion of this report addresses the details of each of the assembly models needed in the above calculations. In particular, the following three subsections detail the Fuel Assembly Models, the Control Cell Model, and the generic 1-D UMLRR Model of the full system. The details of these models, although not necessary for use of the final LEU cross section libraries, do provide good documentation of the overall process (and they could be quite useful should any modifications be required in the future).

The Fuel Assembly Models

As outlined above, a detailed representation of each assembly is necessary to properly take into account the spatial self shielding associated with the specific heterogeneous geometry of the element. Since the multidimensional models in VENTURE and DORT use homogenized representations of the detailed heterogeneous geometry, the spatial self shielding effects must be incorporated within the effective homogenized cross sections used in the multidimensional

calculations. This spatial averaging process is approximated with 1-D assembly models within the XSDRN code. To develop these models, one needs an accurate representation of the actual heterogeneous geometry of the element of interest.

The details of the LEU fuel assembly models (full and partial assemblies) are discussed in this subsection. Most of the physical data given here come directly from Refs. 6 and 7. A sketch of a typical LEU fuel assembly is given in Fig. 1. The fuel is of the flat plate type. Each assembly consists of two aluminum side plates and 18 equally spaced thin fuel/aluminum plates. Note that the current HEU assembly design contains high enriched uranium fuel in all 18 plates, but the proposed LEU design uses low enriched uranium silicide fuel in only the central 16 plates, with the two end plates containing pure aluminum. The new LEU element is referred to as the LEU1618 design to represent the fact that only 16 of the 18 plates contain fuel.

The meat of the new LEU fuel is an U_3Si_2 -Al alloy. The U_3Si_2 contribution is about 67 w/o. The uranium in the LEU fuel is enriched to about 20 w/o U235. Each plate contains 12.5 g of U235. The active fuel height is 59.69 cm (23.5 in). The LEU fuel plate design, including the planar plate dimensions, is given within Fig. 1. Note that the overall LEU fuel plate design is fundamentally the same as for the current HEU fuel plate, but the meat, clad, and coolant thicknesses are substantially different. The detailed physical data for the proposed LEU1618 assembly design are summarized in Table 1. For comparison purposes, the corresponding HEU assembly data are also included in Table 1. These data, given with more significant figures than indicated in the above descriptions, were used to construct the 1-D XSDRN assembly model.

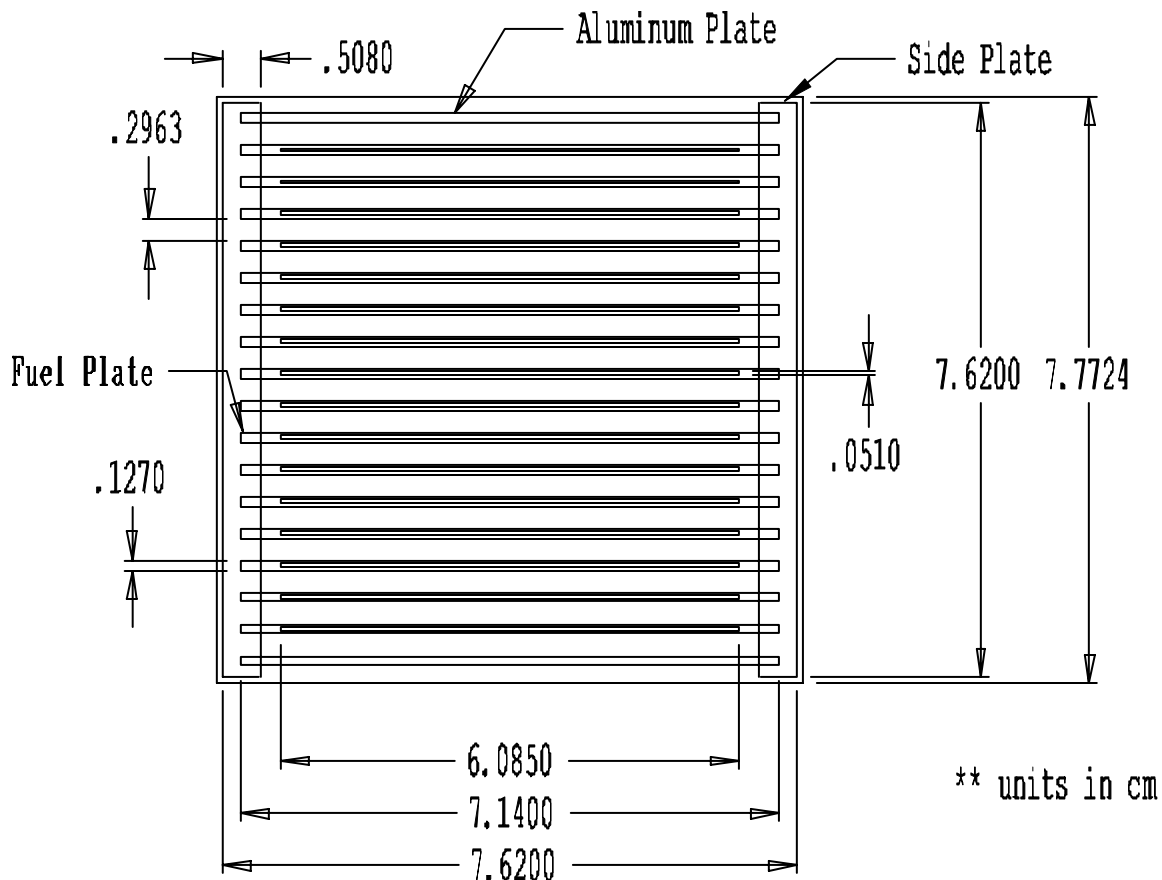


Fig. 1 Standard LEU1618 fuel assembly geometry (modified slightly from Refs. 6-7).

The challenge associated with developing an appropriate assembly model lies in the decisions made when converting a real 2-D assembly into a 1-D computational model. For the fuel assembly developed here, the central 6.0850×7.7724 cm² fueled region (see Fig. 1) was considered as the fuel lattice and everything else as the non-lattice region. A half-symmetric explicit representation of a 1-D cut through the lattice region, parallel to the side plates, gives the geometry used in the XSDRN assembly models -- see Tables 2 and 3 for a detailed description of the FULL and PARTIAL assemblies, respectively. These models contain a sequence of water/clad/fuel/clad/water regions for nine plate sequences, with a reflected boundary condition used on both sides of the 1-D slab geometry to represent an infinite array of such assemblies. For the LEU1618 design, the end plates are pure aluminum and this is treated in the model simply by placing clad material into that particular region (Zone 35 in Tables 2 and 3).

The material compositions used in the XSDRN models are given in Table 4. The region densities for the fuel, clad, and water zones were computed using physical data from Table 1 with all the materials at room temperature. These densities are nearly identical with those from Ref. 6 except that the ¹⁰B impurity in the aluminum clad is included here in only trace amounts.

Table 1 Physical data for the HEU and LEU1618 assembly designs.

Assembly Geometry Data	HEU Assembly	LEU Assembly
fuel plates per element	18	16
aluminum plates per element	0	2
U235 loading (g/element)	135	200
side plate thickness (cm)	0.475	0.508
channel thickness (cm)	0.2718	0.2963
square assembly dimension (cm)	7.62	7.62
square assembly dim. with gap (cm)	7.7724	7.7724
total assembly area (cm ²)	60.4102	60.4102
Plate Geometry Data	HEU Plate	LEU Plate
fuel type	U-Al alloy	U ₃ Si ₂ -Al alloy
U fraction or U ₃ Si ₂ fraction (w/o)	0.24	0.6754
enrichment fraction (w/o)	0.93	0.1975
U235 loading (g/plate)	7.5	12.5
plate width (cm)	7.046	7.14
fuel meat width (cm)	5.461	6.085
plate thickness (cm)	0.1524	0.1270
fuel meat thickness (cm)	0.0305	0.0510
plate height (cm)	63.5	63.5
fuel meat height (cm)	60.96	59.69
fuel meat volume (cm ³)	10.154	18.524

Table 2 Geometry and material layout for the LEU1618 FULL fuel assembly model.

Zone #	Description	Width (cm)	Total Distance (cm)	# of Mesh	Material #
1	water	0.14817	0.1482	3	4
2	clad	0.03800	0.1862	2	2
3	fuel	0.05100	0.2372	3	1
4	clad	0.03800	0.2752	2	2
5	water	0.29633	0.5715	5	4
6	clad	0.03800	0.6095	2	2
7	fuel	0.05100	0.6605	3	1
8	clad	0.03800	0.6985	2	2
9	water	0.29633	0.9948	5	4
10	clad	0.03800	1.0328	2	2
11	fuel	0.05100	1.0838	3	1
12	clad	0.03800	1.1218	2	2
13	water	0.29633	1.4182	5	4
14	clad	0.03800	1.4562	2	2
15	fuel	0.05100	1.5072	3	1
16	clad	0.03800	1.5452	2	2
17	water	0.29633	1.8415	5	4
18	clad	0.03800	1.8795	2	2
19	fuel	0.05100	1.9305	3	1
20	clad	0.03800	1.9685	2	2
21	water	0.29633	2.2648	5	4
22	clad	0.03800	2.3028	2	2
23	fuel	0.05100	2.3538	3	1
24	clad	0.03800	2.3918	2	2
25	water	0.29633	2.6881	5	4
26	clad	0.03800	2.7261	2	2
27	fuel	0.05100	2.7771	3	1
28	clad	0.03800	2.8151	2	2
29	water	0.29633	3.1115	5	4
30	clad	0.03800	3.1495	2	2
31	fuel	0.05100	3.2005	3	1
32	clad	0.03800	3.2385	2	2
33	water	0.29633	3.5348	5	4
34	clad	0.03800	3.5728	2	2
35	clad	0.05100	3.6238	3	2
36	clad	0.03800	3.6618	2	2
37	water	0.22437	3.8862	4	5

Table 3 Geometry and material layout for the LEU1618 PARTIAL fuel assembly model.

Zone #	Description	Width (cm)	Total Distance (cm)	# of Mesh	Material #
1	water	0.14817	0.1482	3	4
2	clad	0.05075	0.1989	2	2
3	fuel	0.02550	0.2244	2	1
4	clad	0.05075	0.2752	2	2
5	water	0.29633	0.5715	5	4
6	clad	0.05075	0.6222	2	2
7	fuel	0.02550	0.6477	2	1
8	clad	0.05075	0.6985	2	2
9	water	0.29633	0.9948	5	4
10	clad	0.05075	1.0456	2	2
11	fuel	0.02550	1.0711	2	1
12	clad	0.05075	1.1218	2	2
13	water	0.29633	1.4182	5	4
14	clad	0.05075	1.4689	2	2
15	fuel	0.02550	1.4944	2	1
16	clad	0.05075	1.5452	2	2
17	water	0.29633	1.8415	5	4
18	clad	0.05075	1.8922	2	2
19	fuel	0.02550	1.9177	2	1
20	clad	0.05075	1.9685	2	2
21	water	0.29633	2.2648	5	4
22	clad	0.05075	2.3156	2	2
23	fuel	0.02550	2.3411	2	1
24	clad	0.05075	2.3918	2	2
25	water	0.29633	2.6882	5	4
26	clad	0.05075	2.7389	2	2
27	fuel	0.02550	2.7644	2	1
28	clad	0.05075	2.8152	2	2
29	water	0.29633	3.1115	5	4
30	clad	0.05075	3.1622	2	2
31	fuel	0.02550	3.1877	2	1
32	clad	0.05075	3.2385	2	2
33	water	0.29633	3.5348	5	4
34	clad	0.05075	3.5856	2	2
35	clad	0.02550	3.6111	2	2
36	clad	0.05075	3.6618	2	2
37	water	0.22437	3.8862	4	5

Table 4 Material composition data (atom/b-cm) for the XSDRN fuel assembly models.

Material	Matl #	²³⁵ U	²³⁸ U	²⁷ Al	²⁸ Si	¹⁶ O	¹ H	¹⁰ B
fuel	1	1.729-3	6.937-3	3.954-2	5.777-3	—	—	—
clad	2	—	—	6.026-2	—	—	—	1.00-12
water	3	—	—	—	—	3.343-2	6.686-2	—
modified water (interior)	4	—	—	1.626-2	—	3.766-2	7.531-2	—
modified water (edge)	5	—	—	1.460-2	—	3.723-2	7.445-2	—

The two “modified water” compositions given in Table 4 have increased densities to account for the non-lattice materials that are not treated explicitly in the 1-D representation of the assembly. Based on the geometry given in Fig. 1, the non-lattice component accounts for nearly 22 % of the assembly planar area. The aluminum and water in this non-lattice region, with region volume fractions of 0.681 and 0.319, respectively, were included in the 1-D model by artificially increasing the moderator material densities. This was done by conserving the number of atoms (nuclide density times volume) in the non-lattice and moderator regions. Note also that the last water region in the model (Zone 37) includes the assembly gap dimension. Thus, since its volume is slightly different from the other moderator zones, this zone was treated separately when computing the non-lattice contribution to the effective densities given for Material #5 in Table 4.

Although the above modeling procedure is approximate, it does maintain the actual 1-D geometry of the repeated water/clad/fuel/clad/water lattice. It also retains the assembly average water to heavy metal ratio which is key to computing the proper energy spectrum in the system. Although placement of the non-lattice material within the water gaps is clearly nonphysical, it appears to be a reasonable tradeoff -- since simply ignoring the non-lattice material is not an acceptable alternative.

The above modeling technique was used for both the full and partial assembly designs. The partial assembly is identical to the full assembly except it has one-half the U235 loading in each plate. This is accomplished by reducing the fuel meat thickness by a factor of two, with a corresponding increase in clad thickness to maintain the same water gap dimension between plates. Comparison of the geometry data in Table 2 for the full LEU1618 assembly and Table 3 for the partial LEU1618 assembly reflects these meat and clad thickness changes, with all other dimensions remaining constant between the two assemblies.

The geometry and composition data for these two assembly models were converted into XSDRN format and 1-D P₃S₈ transport calculations were made, with the goal of generating effective assembly cross sections for both the full and partial fuel elements. The k_{∞} values for these LEU1618 assembly models, compared to similar results for the HEU assemblies, are listed below:

	HEU k_{∞}	LEU1618 k_{∞}
Full Fuel Assembly	1.558	1.628
Partial Fuel Assembly	1.272	1.410

As apparent, even with only 16 fuel plates, the LEU assemblies are more reactive than the corresponding HEU assemblies. The higher reactivity assemblies lead to an overall core size that is smaller for the LEU versus HEU configurations, since the increased leakage in a smaller core compensates for the higher reactivity of the individual fuel elements.

The end result of the fuel assembly computations was the generation of the **fassy.lib** and **passy.lib** cross section libraries. These datasets contain properly shielded cross section data that are used in several additional intermediate calculations as well as in the final 1-D core computation that generates the final libraries for use in VENTURE and DORT (see below).

The Control Cell Model

The UMass-Lowell Research Reactor (UMLRR) has four large control blades for gross reactivity control and a single low-worth regulating rod for fine reactivity adjustments. In both cases, the neutron poison material is boral -- a solid mixture of B_4C and aluminum -- and the thicknesses are such that each poison region is nearly a black absorber for thermal neutrons (i.e. thermal neutrons entering these regions generally do not leave).

Unfortunately, diffusion theory, which is used primarily to give reactivity information and core power distributions, does not perform very well when there are large abrupt changes in the absorption cross section from one region to another. Often, a relatively small physical region containing poison may “look” quite large to a neutron in terms of the neutron mean free path. This situation is not easily treated with diffusion theory techniques since the control blades represent sinks for thermal neutrons.

However, in a manner similar to the fuel assembly models just discussed, one can perform 1-D transport theory computations and spatially average the poison cross section over the homogenized poison regions that will be modeled in the multidimensional calculations. In this way, the poison is smeared over a slightly larger region and the spatial self shielding effect is treated properly. The resultant poison cross sections are usually reduced such that the homogenized geometry calculations maintain approximately the same reaction rates as determined from the detailed geometry computation. This process also gives less of a mismatch in the region-dependent cross sections for the diffusion theory calculations, which tends to improve the base diffusion approximation and give more consistent poison reactivity worths relative to transport theory.

The top view of a single UMLRR control blade and corresponding control shroud is shown in Fig. 2. Based on this geometry and the above rationalization, a 1-D computational model can be envisioned that cuts through the boral, clad, water, shroud, and outer water gap to explicitly represent the control cell geometry. In particular, a half-symmetric representation of this geometry, beginning in the center of the boral region and extending outward through the shroud and water gap was chosen for the control cell model. A homogenized region containing the equivalent of 1.5 fuel assemblies is appended to the end of the control cell. With reflected boundary conditions on both sides of the full slab geometry, we have a 1-D system that models

an infinitely repeating array of control cell/3 fuel assemblies/control cell/3 fuel assemblies etc.. Again, although this is not exactly the geometry we desire, it does provide a reasonable approximation that retains much of the explicit physical geometry of interest.

The computational model just described qualitatively is summarized in detail in Tables 5 and 6. In particular, Table 5 gives the 1-D geometric details and Table 6 gives the composition data for the four different materials within the model. Most of these data were obtained from the physical data in Table 1 or directly from Ref. 6. Where needed, the original UMLRR FSAR (Ref. 8) and some facility blueprints were also used to validate and/or resolve ambiguous data.

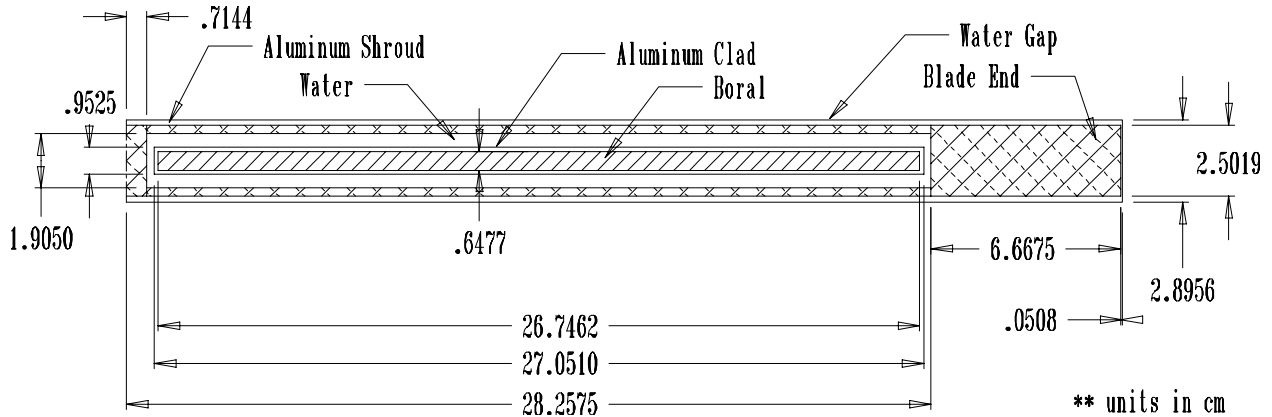


Fig. 2 UMLRR control blade and shroud geometry (modified slightly from Ref. 6).

Table 5 Geometry and material layout for the control cell model.

Zone #	Description	Width (cm)	Total Distance (cm)	# of Mesh	Material #
1	boral	0.32385	0.3239	6	1
2	clad	0.15240	0.4763	3	2
3	water	0.47675	0.9530	9	3
4	shroud	0.29845	1.2515	5	2
5	water	0.19685	1.4483	4	3
6	fuel	11.6586	13.107	60	4

Table 6 Material composition data (atom/b-cm) for the XSDRN control cell model.

Material	Matl #	²³⁵ U	²³⁸ U	²⁷ Al	²⁸ Si	¹⁶ O	¹ H	¹⁰ B	¹² C
boral	1	—	—	3.830-2	—	—	—	7.567-3	1.158-2
clad	2	—	—	6.026-2	—	—	—	—	—
water	3	—	—	—	—	3.343-2	6.686-2	—	—
fuel	4	1.421-4	5.702-4	2.108-2	4.749-4	2.079-2	4.158-2	—	—

The data from Tables 5 and 6 were used to construct the control cell model in XSDRN. A P_3S_8 calculation with 87 fine mesh intervals was made. The dataset from the LEU full fuel assembly calculation (**fassy.lib**) was merged with the BONAMI output for a homogeneous-zone resonance computation for the boral region to form the intermediate library for the XSDRN computation. The inner cell weighting option was used to spatially collapse the cross sections over only the inner two regions of the model. This option was chosen because the geometry for the control regions in the VENTURE and DORT 2-D models will include only the boral region and the control blade cladding or sheath. Thus, the shielded cross section should only be averaged over the domain to be used in subsequent computations. The rest of the control cell model is required to get the proper space and energy flux distribution for the two inner regions.

The k_∞ value for the LEU control cell computation was 1.358 (this can be compared to $k_\infty = 1.258$ for the HEU control cell model). The essential output from this computation was the **cntl.lib** library, which contains the desired inner cell weighted poison cross sections at the 199/42 fine-group level. These data are used in the final 1-D UMLRR core calculation to generate the desired broad-group VENTURE and DORT cross section libraries (see below).

A Generic 1-D UMLRR LEU Core Model

The XSDRN configurations discussed thus far represent “assembly models” that try to account for the detailed heterogeneity that actually occurs within the full and partial fuel assemblies and the control blade cell. These models generate cross sections that have been spatially averaged over the appropriate regions to be used in subsequent “homogeneous” computational models. These data still have the original fine-group energy structure as the base VITAMIN-B6 library, and a final step is needed to collapse these data to the desired broad-group level of interest.

The purpose of this section is to document the final model in the sequence -- that is a generic 1-D full core configuration that is used to determine a weight function for the energy collapse of the cross sections to some broad-group structure (2 groups and 47/20 groups in this work). This 1-D XSDRN model represents a slice through a “typical” LEU core in the direction parallel to the two rows of control blades. However, the model is not specific to any planned operational configuration. Instead, it tries to incorporate all of the geometric situations that may be encountered. It includes all the key components such as water reflector material, graphite reflectors, radiation baskets, partial and full fuel assemblies, control material, a lead shield, and a large graphite thermal column.

The “region weighting” option is used in XSDRN to perform the energy collapse. This means that the energy-dependent flux to be used in the cross section averaging process is spatially averaged over all regions where a particular nuclide resides. For example, since separate ^{235}U datasets for the partial and full assemblies have been retained, two ^{235}U cross section sets are produced -- one using the flux averaged over the partial fuel assembly and another using the spectrum averaged over the several full fuel assemblies that occur in the model. Similarly, several different aluminum cross sections are generated; one each for the partial fuel assembly, the full fuel assembly, the control region, and one that represents several generic structure regions throughout the system (radiation baskets, graphite reflectors, core box structure, etc.). These different cross sections need to be used appropriately, since different energy spectra and different spatial self-shielding models have been used. This level of detail is important in

some cases, and it represents common practice when generating broad-group cross sections to be used throughout several regions of a whole core configuration (as planned for subsequent VENTURE and DORT calculations).

The actual 1-D XSDRN geometry, with corresponding zone dimensions and descriptions, is summarized in Table 7, and the material concentrations used within the model are given in Table 8. The model is roughly consistent with the x-direction of the UMLRR configuration with fresh LEU fuel loaded into the central core region. All but one of the nine grid locations along the x-direction represents a single homogenized zone containing either a radiation basket (RB), graphite reflector (GR), partial LEU fuel assembly (PA), or full LEU fuel assembly (FA). The partial element in the UMLRR has the same macroscopic geometry as the full fuel assembly, except that it only has half the ^{235}U loading in each fuel plate. Note also that the regulating rod position (grid location #8) is broken into three regions -- a control zone that contains a homogenized mix of boral sandwiched between aluminum clad, and a zone which surrounds the control material containing a mixture of aluminum and water. This overall 1-D geometry accounts for all the essential features present in the real system and it allows a set of reasonable weight functions to be determined for collapsing the fine-group data to some broad-group level.

Table 7 Geometry details for 1-D XSDRN model used for the LEU cross section collapse.

Zone Description	Zone Width (cm)	Total Distance (cm)	Zone #	Material # (Reference)	# of Mesh
water	40.0000	40.0000	1	1	40
water+core box	1.5875	41.5875	2	2	2
grid location #1	7.7724	49.3599	3	5	12
grid location #2	7.7724	57.1323	4	3	12
grid location #3	7.7724	64.9047	5	6	12
grid location #4	7.7724	72.6771	6	7	12
grid location #5	7.7724	80.4495	7	7	12
grid location #6	7.7724	88.2219	8	7	12
grid location #7	7.7724	95.9943	9	7	12
grid location #8a	3.4100	99.4043	10	4	5
grid location #8b	0.9524	100.3567	16	12	3
grid location #8c	3.4100	103.7667	10	4	5
grid location #9	7.7724	111.5391	11	5	12
core box+water+clad	1.5875	113.1266	12	8	2
lead shield	7.6200	120.7466	13	9	8
clad+water+clad	2.8575	123.6041	14	10	3
thermal column	60.0000	183.6041	15	11	45

Table 8 Material compositions (atom/b-cm) for the 1-D XSDRN LEU core model.

Material	Matl ID	²³⁵ U	²³⁸ U	²⁷ Al	²⁸ Si	¹⁶ O	¹ H	¹² C	¹⁰ B	^{nat} Cd	Pb**
Water Refl	1	—	—	—	—	3.343-2	6.686-2	—	—	—	—
Water+Core Box	2	—	—	2.410-2	—	2.006-2	4.012-2	—	—	—	—
Rad Basket+Air	3	—	—	1.366-2	—	2.066-2	4.133-2	—	1.000-12	1.000-12	—
Rad Basket+Water*	4	—	—	1.164-2	—	2.697-2	5.394-2	—	—	—	—
Graphite Reflector	5	—	—	2.998-3	—	2.318-3	4.636-3	7.068-2	—	—	—
Partial Fuel	6	7.106-5	2.851-4	2.193-2	2.374-4	2.079-2	4.158-2	—	1.000-12	—	—
Full Fuel	7	1.421-4	5.702-4	2.108-2	4.749-4	2.079-2	4.158-2	—	1.000-12	—	—
Core Box+Water+Clad	8	—	—	3.614-2	—	1.337-2	2.674-2	—	—	—	—
Lead Shield	9	—	—	—	—	—	—	—	—	—	3.296-2
Clad+Water+Clad	10	—	—	4.686-2	—	7.428-3	1.486-2	—	—	—	—
Thermal Column	11	—	—	—	—	—	—	8.023-2	—	—	—
Boral+Clad	12	—	—	4.532-2	—	—	—	7.874-3	5.145-3	1.000-12	—

* This homogeneous material models a water-filled radiation basket. The basket with an air-filled bayonet is modeled as Matl #3. In the current model, the outer regulating rod regions (without control) are modeled as a radiation basket filled with water (i.e. Matl #4).

** This density is the sum of the individual isotope densities comprising lead, where ²⁰⁶Pb = 7.910-3, ²⁰⁷Pb = 7.251-3, and ²⁰⁸Pb = 1.780-2 atom/b-cm.

The above computational model was run twice with the goal of generating a 2-group library, **leu2gxs.asnlib**, and a coupled neutron-gamma library with 47 neutron groups and 20 gamma groups, which is denoted as **leu67gxs.asnlib**. These two ANISN-formatted libraries will be used for different end uses (one for VENTURE calculations and one for DORT analyses). Some final processing of these two libraries is necessary before they can be used directly. However, this final step is done within either the VENTURE or DORT systems as appropriate. For example, for VENTURE, one must convert the ANISN formatted file into ISOTXS format and then into GRUPXS format before actual use.² For DORT, one uses the GIP code to put the cross sections into group-ordered format before use in DORT (this code also does the cross section mixing to get macroscopic data).³

Full validation of the new LEU libraries is not complete. In fact, at present, the 67-group library for use in DORT has not been used at all. Some initial testing of the 2-group library with a preliminary VENTURE 2-D XY model of the reference LEU core configuration has been completed, and comparisons to UMLRR LEU core calculations made several years ago show good consistency.⁷ The details of the preliminary VENTURE model and some reactivity comparisons with Ref. 7 are highlighted in the next section of this report.

Preliminary Verification of the 2-Group LEU Cross Section Library

As a test of the library development process, a preliminary VENTURE model of the reference LEU core was constructed and some criticality calculations were made. This section of the report documents this VENTURE model in some detail since this will serve as the base for future computational models of the LEU-fueled UMLRR. A brief comparison to some calculational results from Ref. 7 is also given.

The 2-D Reference LEU Configuration

The UMass-Lowell Research Reactor (UMLRR) contains a 9x7 grid of fuel assemblies, graphite reflector elements, radiation baskets, and corner posts. It also has two grid locations reserved for an external neutron source and a low-worth regulating rod for fine reactivity control. Four large control blade assemblies are used for gross reactivity control and for flux shape adjustments. From the top view, the reactor is enclosed by an aluminum core box and a large pool of demineralized water surrounds the system on three sides, with a 3" lead shield and large graphite thermal column on the remaining side. A specific arrangement of fuel elements, graphite reflector blocks, and radiation baskets make up a particular core configuration.

The basic layout for the proposed reference LEU core arrangement, including the beam ports and thermal column, is pictured in Fig. 3, and an expanded view of the core region is given in Fig. 4. The reference configuration contains 20 fresh LEU full fuel assemblies arranged symmetrically in the center of the 9x7 grid. Directly in the middle of the core is a central irradiation zone known as the flux trap. The flux trap is similar to a radiation basket, except that the region between the inner irradiation tube and the outer aluminum can is filled with graphite.

Putting together a computational model for any reactor system requires precise information about the geometry and material composition of the actual system. It also requires a number of decisions by the model builder concerning the mesh grid layout, the zone placement,

and the degree of detail that can be modeled efficiently and accurately within the limitations of the available computer tools. These decisions often require that tradeoffs be made in the model details, and these usually result in the consolidation of the fine heterogeneous geometry details into larger homogeneous regions. The homogeneous zones must use properly averaged material densities to approximate the actual materials present in the real system. They also require a problem-dependent set of microscopic cross sections that account for the heterogeneous detail that is lost when developing a homogeneous model -- and a suitable library was already generated in the first part of this project.

In the current model, each interior grid location within the UMLRR geometry (the central 7x5 grid array) was broken into a 5-region, 3-zone symmetric configuration. Each grid location has two sides (top and bottom), two edges (left and right), and a center region. This zone layout and the specific dimensions used are consistent with the LEU fuel assembly geometry. This arrangement allows explicit modeling of the side plate and central fueled regions of the fuel assembly. It also allows for treatment of the aluminum end plates and the slightly increased water content associated with the assembly gap. This explicit 5-region, 3-zone model for each interior grid location is apparent in the LEU01 configuration shown in Figs. 3 and 4. The remaining structures (control blades, corner posts, lead shield, thermal column, beam ports, etc.) are also displayed explicitly in these diagrams.

Getting the actual geometry and homogenized material information into the VENTURE code requires specification of three key arrays: a region grid layout, a zone by region map, and a material by zone array. In particular, the region boundaries with the desired mesh spacing for the UMLRR geometry are specified in Tables 9 and 10, respectively, for the x-direction and y-direction region boundaries. The present model uses a 65x65 region layout and a fine-mesh grid containing 175x170 mesh points, spanning 225 cm in the x direction and 200 cm in the y direction.

A zone by region map is then given which identifies a specific zone ID with each region in the 2-D grid structure. For the current model, the zone numbers range from 1 to 176 (for ease in modeling, some zones numbers were not used). Attention to detail in specifying the zone and mesh map allows one to build a relatively flexible geometry that can handle a variety of material configurations. This modeling step is quite important, because a little added flexibility here can save lots of analysis time if several material layouts need to be considered.

The final zone by region map for the current model was passed through a short Matlab program to plot the resultant zone layout. The result of this exercise gave the zone and material maps for the LEU01 configuration shown in Figs. 3 and 4. This process helps to debug the geometry setup and to visualize the details of the final model.

Note that the angled beam ports were modeled with a relatively coarse stair-step approximation to minimize the number of region boundaries specified within VENTURE. Since the regions of interest here occur within the core or at the beam port/core box interface, the jagged edges used to represent the beam port sides should not seriously affect the current analysis.

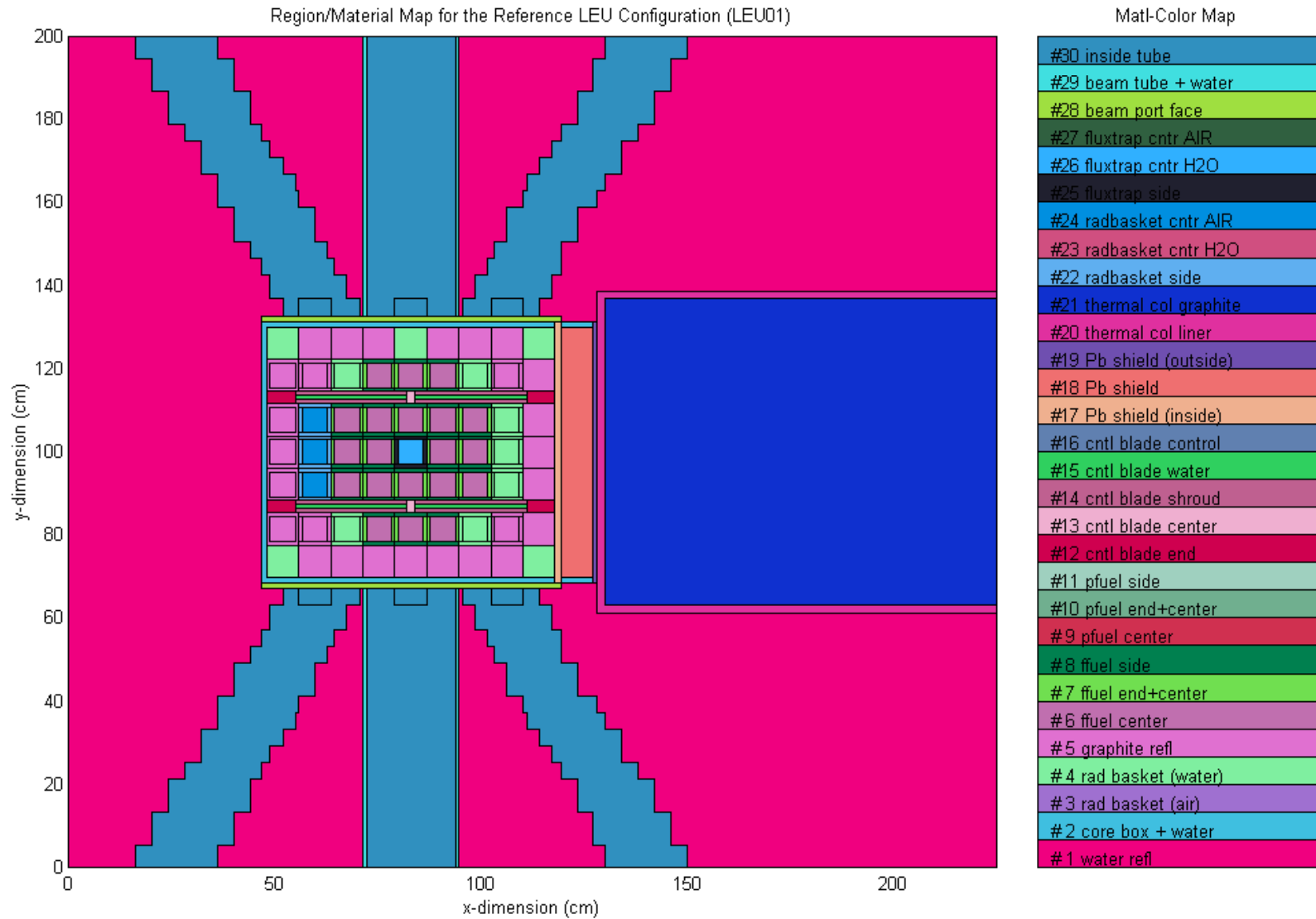


Fig. 3 Basic geometry and material layout for the reference LEU core configuration (LEU01 model).

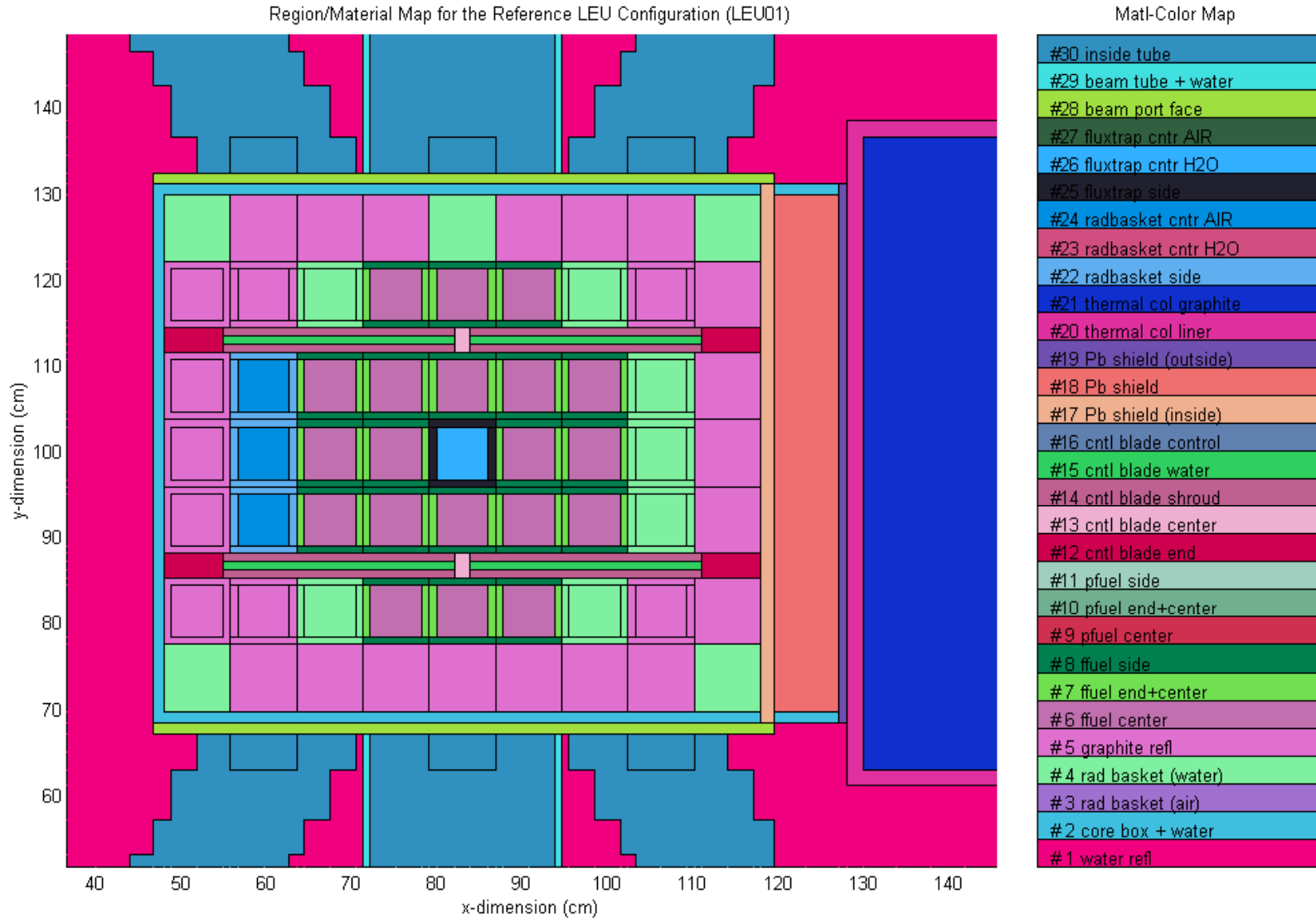


Fig. 4 Expanded view of the core region for the reference LEU configuration (LEU01 model).

Table 9 X-direction region boundaries for the UMLRR LEU geometry (LEU01 model).

Primary Region Description	Secondary Region Description	Dimensions in cm		Dimensions in inches		# Mesh	Total Mesh	
		Thickness	Total Dist.	Thickness	Total Dist.			
water reflector	water + beam ports	4.0000	4.0000	1.5748	1.5748	3	3	
	water + beam ports	4.0000	8.0000	1.5748	3.1496	3	6	
	water + beam ports	4.0000	12.0000	1.5748	4.7244	3	9	
	water + beam ports	4.0000	16.0000	1.5748	6.2992	3	12	
	water + beam ports	4.0000	20.0000	1.5748	7.8740	3	15	
	water + beam ports	4.0000	24.0000	1.5748	9.4488	3	18	
	water + beam ports	4.0000	28.0000	1.5748	11.0236	3	21	
	water + beam ports	4.0000	32.0000	1.5748	12.5984	3	24	
	water + beam ports	4.0000	36.0000	1.5748	14.1732	3	27	
	water + beam ports	4.0000	40.0000	1.5748	15.7480	3	30	
	water + beam ports	4.0000	44.0000	1.5748	17.3228	3	33	
	water + beam ports	2.7300	46.7300	1.0748	18.3976	2	35	
	water refl + core box	water(0.5) + Al(0.5)	1.2700	48.0000	0.5000	18.8976	1	36
	grid location #1	rad basket or graphite refl	0.8437	48.8437	0.3322	19.2298	1	37
		rad basket or graphite refl	3.0425	51.8862	1.1978	20.4276	3	40
rad basket or graphite refl		3.0425	54.9287	1.1978	21.6255	3	43	
7.7724	basket/refl & control shroud end	0.8437	55.7724	0.3322	21.9576	1	44	
grid location #2	edge(0.5921) + center (0.4079)	0.8437	56.6161	0.3322	22.2898	1	45	
	center	3.0425	59.6586	1.1978	23.4876	3	48	
	7.7724	center	3.0425	62.7011	1.1978	24.6855	3	51
	edge(0.5921) + center (0.4079)	0.8437	63.5448	0.3322	25.0176	1	52	
grid location #3	fuel edge(0.5921) + fuel center (0.4079)	0.8437	64.3885	0.3322	25.3498	1	53	
	fuel center	3.0425	67.4310	1.1978	26.5476	3	56	
	7.7724	fuel center	3.0425	70.4735	1.1978	27.7455	3	59
	fuel edge(0.5921) + fuel center (0.4079)	0.8437	71.3172	0.3322	28.0776	1	60	
grid location #4	fuel edge(0.5921) + fuel center (0.4079)	0.8437	72.1609	0.3322	28.4098	1	61	
	fuel center	3.0425	75.2034	1.1978	29.6076	3	64	
	7.7724	fuel center	3.0425	78.2459	1.1978	30.8055	3	67
	fuel edge(0.5921) + fuel center (0.4079)	0.8437	79.0896	0.3322	31.1376	1	68	
grid location #5	fuel edge(0.5921) + fuel center (0.4079)	0.8437	79.9333	0.3322	31.4698	1	69	
	7.7724	fuel center	2.0820	82.0153	0.8197	32.2895	2	71
		fuel center + control center structure	1.9210	83.9363	0.7563	33.0458	2	73
	fuel center	2.0820	86.0183	0.8197	33.8655	2	75	
	fuel edge(0.5921) + fuel center (0.4079)	0.8437	86.8620	0.3322	34.1976	1	76	
grid location #6	fuel edge(0.5921) + fuel center (0.4079)	0.8437	87.7057	0.3322	34.5298	1	77	
	fuel center	3.0425	90.7482	1.1978	35.7276	3	80	
	7.7724	fuel center	3.0425	93.7907	1.1978	36.9255	3	83
	fuel edge(0.5921) + fuel center (0.4079)	0.8437	94.6344	0.3322	37.2576	1	84	
grid location #7	fuel edge(0.5921) + fuel center (0.4079)	0.8437	95.4781	0.3322	37.5898	1	85	
	fuel center	3.0425	98.5206	1.1978	38.7876	3	88	
	7.7724	fuel center	3.0425	101.5631	1.1978	39.9855	3	91
	fuel edge(0.5921) + fuel center (0.4079)	0.8437	102.4068	0.3322	40.3176	1	92	
grid location #8 (no reg rod)	fuel edge(0.5921) + fuel center (0.4079)	0.8437	103.2505	0.3322	40.6498	1	93	
	fuel center	3.0425	106.2930	1.1978	41.8476	3	96	
	7.7724	fuel center	3.0425	109.3355	1.1978	43.0455	3	99
	fuel edge(0.5921) + fuel center (0.4079)	0.8437	110.1792	0.3322	43.3776	1	100	
grid location #9 (no reg rod)	basket/refl & control shroud end	0.8437	111.0229	0.3322	43.7098	1	101	
	rad basket or graphite refl	3.0425	114.0654	1.1978	44.9076	3	104	
	rad basket or graphite refl	3.0425	117.1079	1.1978	46.1055	3	107	
7.7724	rad basket or graphite refl	0.8437	117.9516	0.3322	46.4376	1	108	
core box+gap+shield clad	water(0.400) + Al(0.600)	1.5875	119.5391	0.6250	47.0626	1	109	
lead shield	pure lead	3.8100	123.3491	1.5000	48.5626	3	112	
lead shield	pure lead	3.8100	127.1591	1.5000	50.0626	3	115	
shield clad + water gap	water(0.6666) + Al(0.3334)	0.9525	128.1116	0.3750	50.4376	1	116	
Thermal column clad	Al clad	1.9050	130.0166	0.7500	51.1876	2	118	
Thermal column	graphite + water + beam ports	3.9350	133.9516	1.5492	52.7369	3	121	
	graphite + water + beam ports	4.0000	137.9516	1.5748	54.3117	3	124	
	graphite + water + beam ports	4.0000	141.9516	1.5748	55.8865	3	127	
	graphite + water + beam ports	4.0000	145.9516	1.5748	57.4613	3	130	
	graphite + water + beam ports	4.0000	149.9516	1.5748	59.0361	3	133	
	graphite + water + beam ports	4.0000	153.9516	1.5748	60.6109	3	136	
	graphite + water + beam ports	4.0000	157.9516	1.5748	62.1857	3	139	
	graphite + water + beam ports	4.0000	161.9516	1.5748	63.7605	3	142	
	graphite + water + beam ports	4.0000	165.9516	1.5748	65.3353	3	145	
	graphite + water + beam ports	59.0484	225.0000	23.2474	88.5827	30	175	

Table 10 Y-direction region boundaries for the UMLRR LEU geometry (LEU01 model).

Primary Region Description	Secondary Region Description	Dimensions in cm		Dimensions in inches		# Mesh	Total Mesh
		Thickness	Total Dist.	Thickness	Total Dist.		
water reflector	water + beam ports	1.3600	1.3600	0.5354	0.5354	1	1
	water + beam ports	4.0000	5.3600	1.5748	2.1102	3	4
	water + beam ports	4.0000	9.3600	1.5748	3.6850	3	7
	water + beam ports	4.0000	13.3600	1.5748	5.2598	3	10
	water + beam ports	4.0000	17.3600	1.5748	6.8346	3	13
	water + beam ports	4.0000	21.3600	1.5748	8.4094	3	16
	water + beam ports	4.0000	25.3600	1.5748	9.9843	3	19
	water + beam ports	4.0000	29.3600	1.5748	11.5591	3	22
	water + beam ports	4.0000	33.3600	1.5748	13.1339	3	25
	water + beam ports	4.0000	37.3600	1.5748	14.7087	3	28
	water + beam ports	4.0000	41.3600	1.5748	16.2835	3	31
	water + beam ports	4.0000	45.3600	1.5748	17.8583	3	34
	water + beam ports	4.0000	49.3600	1.5748	19.4331	3	37
	water + beam ports	4.0000	53.3600	1.5748	21.0079	3	40
	water + beam ports	4.0000	57.3600	1.5748	22.5827	3	43
	water + beam ports	4.0000	61.3600	1.5748	24.1575	3	46
water refl or graphite refl clad	water or Al clad	1.9050	63.2650	0.7500	24.9075	2	48
water refl or thermal col	water or graphite	2.0975	65.3625	0.8258	25.7333	2	50
water refl or thermal col	water or graphite	2.0975	67.4600	0.8258	26.5591	2	52
low den water + beam port plate	water(0.1409) + Al(0.2953)	1.2700	68.7300	0.5000	27.0591	1	53
water refl + core box	water(0.5) + Al(0.5)	1.2700	70.0000	0.5000	27.5591	1	54
grid location A	rad basket or graphite refl	7.7724	77.7724	3.0600	30.6191	8	62
grid location B	side	0.8437	78.6161	0.3322	30.9512	1	63
7.7724	center	6.0850	84.7011	2.3957	33.3469	6	69
	side	0.8437	85.5448	0.3322	33.6791	1	70
control location	control shroud	0.9716	86.5164	0.3825	34.0616	1	71
2.8957	control center	0.9525	87.4689	0.3750	34.4366	1	72
	control shroud	0.9716	88.4405	0.3825	34.8191	1	73
grid location C	fuel side	0.8437	89.2842	0.3322	35.1513	1	74
7.7724	fuel center	6.0850	95.3692	2.3957	37.5469	6	80
	fuel side	0.8437	96.2129	0.3322	37.8791	1	81
grid location D (no reg rod)	fuel side	0.8437	97.0566	0.3322	38.2113	1	82
7.7724	fuel center	6.0850	103.1416	2.3957	40.6069	6	88
	fuel side	0.8437	103.9853	0.3322	40.9391	1	89
grid location E	fuel side	0.8437	104.8290	0.3322	41.2713	1	90
7.7724	fuel center	6.0850	110.9140	2.3957	43.6669	6	96
	fuel side	0.8437	111.7577	0.3322	43.9991	1	97
control location	control shroud	0.9716	112.7293	0.3825	44.3816	1	98
2.8957	control center	0.9525	113.6818	0.3750	44.7566	1	99
	control shroud	0.9716	114.6534	0.3825	45.1391	1	100
grid location F	fuel side	0.8437	115.4971	0.3322	45.4713	1	101
7.7724	fuel center	6.0850	121.5821	2.3957	47.8670	6	107
	fuel side	0.8437	122.4258	0.3322	48.1991	1	108
grid location G	rad basket or graphite refl	7.7724	130.1982	3.0600	51.2591	8	116
water refl + core box	water(0.5) + Al(0.5)	1.2700	131.4682	0.5000	51.7591	1	117
low den water + beam port plate	water(0.1409) + Al(0.2953)	1.2700	132.7382	0.5000	52.2591	1	118
water refl or thermal col	water or graphite	2.0975	134.8357	0.8258	53.0849	2	120
water refl or thermal col	water or graphite	2.0975	136.9332	0.8258	53.9107	2	122
water refl or graphite refl clad	water or Al clad	1.9050	138.8382	0.7500	54.6607	2	124
water reflector	water + beam ports	4.0000	142.8382	1.5748	56.2355	3	127
	water + beam ports	4.0000	146.8382	1.5748	57.8103	3	130
	water + beam ports	4.0000	150.8382	1.5748	59.3851	3	133
	water + beam ports	4.0000	154.8382	1.5748	60.9599	3	136
	water + beam ports	4.0000	158.8382	1.5748	62.5347	3	139
	water + beam ports	4.0000	162.8382	1.5748	64.1095	3	142
	water + beam ports	4.0000	166.8382	1.5748	65.6843	3	145
	water + beam ports	4.0000	170.8382	1.5748	67.2591	3	148
	water + beam ports	4.0000	174.8382	1.5748	68.8339	3	151
	water + beam ports	4.0000	178.8382	1.5748	70.4087	3	154
	water + beam ports	4.0000	182.8382	1.5748	71.9835	3	157
	water + beam ports	4.0000	186.8382	1.5748	73.5583	3	160
	water + beam ports	4.0000	190.8382	1.5748	75.1331	3	163
	water + beam ports	4.0000	194.8382	1.5748	76.7080	3	166
	water + beam ports	4.0000	198.8382	1.5748	78.2828	3	169
	water + beam ports	1.1618	200.0000	0.4574	78.7402	1	170

Table 11 Material composition (atom/b-cm) for the 2-D UMLRR LEU models.

Material	Matl ID	²³⁵ U	²³⁸ U	²⁷ Al	²⁸ Si	¹⁶ O	¹ H	¹² C	¹⁰ B	Pb**
Water Refl	1	—	—	—	—	3.343-2	6.686-2	—	—	—
Water+Core Box	2	—	—	3.013-2	—	1.672-2	3.343-2	—	—	—
Rad Basket+Air	3	—	—	1.366-2	—	2.066-2	4.133-2	—	—	—
Rad Basket+Water*	4	—	—	1.164-2	—	2.697-2	5.394-2	—	—	—
Graphite Refl	5	—	—	2.998-3	—	2.318-3	4.636-3	7.068-2	—	—
LEU Full Fuel Center	6	2.083-4	8.358-4	1.558-2	6.961-4	2.340-2	4.680-2	—	—	—
LEU Full Fuel Center+End	7	8.497-5	3.409-4	1.542-2	2.839-4	2.431-2	4.862-2	—	—	—
LEU Full Fuel Side	8	—	—	4.103-2	—	1.067-2	2.134-2	—	—	—
LEU Partial Fuel Center	9	1.042-4	4.179-4	1.683-2	3.480-4	2.340-2	4.680-2	—	—	—
LEU Part Fuel Center+End	10	4.249-5	1.705-4	1.593-2	1.420-4	2.431-2	4.862-2	—	—	—
LEU Partial Fuel Side	11	—	—	4.103-2	—	1.067-2	2.134-2	—	—	—
Control Blade End	12	—	—	5.027-2	—	5.533-3	1.106-2	—	—	—
Control Blade Center	13	—	—	4.190-2	—	1.018-2	2.036-2	—	—	—
Control Blade Shroud	14	—	—	1.851-2	—	2.316-2	4.632-2	—	—	—
Control Blade Water	15	—	—	—	—	3.343-2	6.686-2	—	—	—
Control Blade Control	16	—	—	4.549-2	—	—	—	7.785-3	5.087-3	—
Core Box+Water+Clad	17	—	—	3.614-2	—	1.337-2	2.674-2	—	—	—
Lead Shield	18	—	—	—	—	—	—	—	—	3.296-2
Clad+Water	19	—	—	2.000-2	—	2.229-2	4.458-2	—	—	—
Thermal Column Liner	20	—	—	6.024-2	—	—	—	—	—	—
Thermal Column	21	—	—	—	—	—	—	8.023-2	—	—
Rad Basket Side	22	—	—	2.350-2	—	2.039-2	4.078-2	—	—	—
Rad Basket Center (water)	23	—	—	4.146-3	—	3.113-2	6.226-2	—	—	—
Rad Basket Center (air)	24	—	—	7.448-3	—	2.084-2	4.167-2	—	—	—
Flux Trap Side	25	—	—	2.350-2	—	5.987-3	1.197-2	3.457-2	—	—
Flux Trap Center (water)	26	—	—	4.146-3	—	1.600-2	3.200-2	3.631-2	—	—
Flux Trap Center (air)	27	—	—	7.448-3	—	5.703-3	1.141-2	3.631-2	—	—
Beam Port Face	28	—	—	1.779-2	—	4.710-3	9.421-3	—	—	—
Beam Tube Sides	29	—	—	4.535-2	—	8.271-3	1.654-2	—	—	—
Inside Beam Tube	30	—	—	—	—	6.686-3	1.337-2	—	—	—

* This homogeneous material models a water-filled radiation basket. The basket with an air-filled bayonet is modeled as Matl #3. The 2-region radiation basket model uses Matls #22 - #24.

** This density is the sum of the individual isotope densities comprising lead, where ²⁰⁶Pb = 7.910-3, ²⁰⁷Pb = 7.251-3, and ²⁰⁸Pb = 1.780-2 atom/b-cm.

Also, it should be noted that the beam port regions contain a water-air (void) mix with the water volume occupying 20% of the total. This is clearly a rough approximation that tries to account, in a 2-D model, for the real water-air mix in the 3-D system. The modeling also acknowledges the fact that the diffusion theory approximation in VENTURE cannot handle voided regions. The 20/80 water-air mix is clearly a compromise, but again, because our interest is not in radiation transport down the beam port, the assumptions made here should give a reasonable approximation of reality near the beam port/core box interface.

With a flexible geometry properly modeled, one now simply specifies a particular material composition to be placed in each zone (i.e. the so-called material by zone map). Homogenized atom densities for 30 materials were computed based on the nominal region densities and region volume fractions within the given homogeneous zone. These materials include the primary core and reflector components (fuel, radiation baskets, graphite reflectors, core box, water reflector, flux trap, etc.) as well as mixtures that represent the excore structures as described by a 2-D geometric model. The actual isotope densities for each of these homogeneous material zones are given in Table 11.

The geometry and material composition information make up a large part of the actual VENTURE input file. With the addition of suitable boundary conditions, the system power level, and specification of a few switches to set various code options, the model building process is complete. One is now ready to do the actual neutronics calculations in the VENTURE code.

Some Reactivity, Power Density, and Flux Distribution Results

With a proper material by mesh description and a set of few group cross sections (described previously), the VENTURE code was used to compute k_{eff} , the power distribution, and the few-group flux profiles for the reference LEU system. In addition, several configurations with various control blades inserted were simulated. The computed reactivity results for the reference and control-in cases are summarized in Table 12, along with similar data from the FSAR Supplement for the conversion to LEU fuel (see Ref. 7). Selected power density profiles and flux distribution results for the reference control-out configuration are also summarized in Figs. 5-13. These data represent a snapshot of several important physics parameters for the new reference LEU core. They also give a consistent picture relative to the expected behavior of the modeled LEU configuration, with overall results very similar to those presented in Ref. 7.

Focusing first on Table 12, we see that the current model predicts a larger excess reactivity relative to the FSAR model. It also estimates consistently larger blade worths on the order of 0.20 - 0.25 % $\Delta k / k$ for each blade, giving about 0.92 % $\Delta k / k$ additional worth for the full set of four blades. The worth distribution is very uniform as expected for the nearly symmetric reference configuration and this is consistent with the data in the FSAR Supplement. Note also that the 1 % $\Delta k / k$ increase in excess k_{eff} is nearly offset by the increase in blade worth, keeping the shutdown margins nearly constant for the two models (with the most reactive blade stuck out). Finally, we note that the FSAR Supplement uses a 1.5 % $\Delta k / k$ uncertainty bias, and the data quoted here use only raw calculated results from the code.

Table 12 Reactivity data for the reference LEU core.

	New Model (LEUV01)	FSAR Model (Ref. 7)
excess reactivity (% $\Delta k / k$)	2.54	1.49
blade worths (% $\Delta k / k$)		
#1	2.99	2.78
#2	2.93	2.68
#3	2.94	2.69
#4	3.00	2.79
sum #1 - #4	11.86	10.94
worth of all blades (% $\Delta k / k$)	13.16	---
shutdown margin (% $\Delta k / k$)	6.32	6.66

Figures 5-7 give some information about the power density distribution in the new LEU core. The x-directed and y-directed profiles through the peak power density location are given in Figs. 5 and 6. These profiles are similar to those computed in recent HEU analyses,¹ except for the y-profile where the central flux trap region has no power generation. Also note that the peak power density of about 30 W/cm³ for a 20 element LEU core is roughly consistent with similar HEU 29 element calculations with a peak power density of about 20 W/cm³ (i.e. a rough 50% decrease in assemblies translates to a 50% increase in power per assembly for a constant 1 MW total core power level).

Finally, Fig. 7 shows a gray-scale/color representation of the 2-D power density distribution. The symmetric distribution of energy generation shown here is consistent with the symmetric fuel layout and it is apparent that the peak power density regions surround the central flux trap -- due to the large thermal flux peak in this region.

Figures 8-13 show various 1-D flux profiles and 2-D flux distributions for the reference LEU core configuration, and they give a good overview of the expected neutron distribution in the system. The 2-group VENTURE model has the fast/thermal energy boundary at 1 eV. Thus, the fast fluxes displayed here represent an integral over energy from 1 eV to 20 MeV, and the thermal flux includes all neutrons below 1 eV. Also, for display here, all the fluxes computed by VENTURE were multiplied by a factor of 1.385. This value is an estimate of the axial peaking factor for the LEU core from Ref. 7. Thus, the XY distributions here should be representative of the peak fluxes that are expected to occur a little below the mid-plane of the core (the peak occurs below center because of the control blades in the upper portion of the reactor).

The relatively odd-shaped profiles in Figs. 8-11 are as expected and they are quite similar to the data given in Ref. 7. The peaks and valleys coincide with the fuel/water material variations within the system. For example, the x-directed profiles in Figs. 8 and 9 clearly show the large thermal peak and corresponding fast neutron depression expected within the central flux trap. Similar thermal peaking is expected at the assembly edges in the aluminum plate/water cell and in the radiation baskets on both sides of the fueled regions, and these peaks are clearly visible in the plots. Note that the linear scale used in Fig. 8 is a little misleading because it implies that the excore fluxes attenuate fairly symmetrically on both sides of the core in the x direction. However, Fig. 9, which represents the same data from Fig. 8 plotted on a logarithmic

scale, clearly shows that the fast and thermal neutron attenuation is much greater on the water-reflected left side. As expected, the mean free path in the graphite thermal column on the right side is quite large and, as desired, the slope is significantly reduced relative to the water reflector.

Much of this same behavior is shown again in Figs. 10 and 11, only this time the profiles are plotted along two different y-directed axes -- one goes directly through the central flux trap and the other passes through grid column #6 which is just to the right of the flux trap. These figures again show the characteristic thermal peak that is expected in water-filled regions. In Fig. 10, the three large central peaks coincide with the two water-filled control blade locations and the central flux trap. In contrast, Fig. 11 shows two smaller thermal peaks that result from the side plate zones between the two large control blade regions.

Finally, the 2-D distributions for the fast and thermal fluxes in Figs. 12 and 13 show, qualitatively, the overall distribution expected for the reference LEU configuration. Although the gray-scale representation is not as clear as the actual color maps generated directly from the plotting programs, these figures still show the fast flux decreasing gradually as one moves away from the central core region (note that there is a fast flux dip directly in the flux trap location). The thermal flux also decreases with distance from core center, but there is local peaking in the flux trap, water-filled control blade locations, and radiation baskets surrounding the core. Also, as expected on a linear scale, the flux distributions appear quite symmetric because of the near perfect symmetry within the full model.

Validation of the Cross Section Processing Sequence

In addition to the development of a base computational model for future analytical studies of the LEU configuration, the current VENTURE model also represents a test configuration for use of the cross sections generated in the first part of this study. Overall, the results summarized in the previous subsection are quite reasonable and they show all the qualitative behavior that was expected for this model. In addition, the comparisons of excess reactivity, blade worths, and the generic flux profiles with data from the FSAR Supplement⁷ were all favorable. The noted differences are small enough to be attributed to small changes in the geometry and material compositions, and to the differences in the base libraries and methods used in the generation of the few-group cross sections. These results suggest that the 2-group cross sections are indeed reasonable for the given application, and they give some degree of credibility to the overall procedure for generating the problem-dependent broad-group libraries. Although these arguments only represent a qualitative verification, this is the only approach to validation that can be made until actual experimental testing becomes possible (once the new LEU core is operational).

Thus, the cross section libraries generated in this study have passed the first level of validation, and they will be used for subsequent VENTURE and DORT analyses of the LEU core. Once the LEU configuration is operational, further validation tests against experimental measurements will be made.

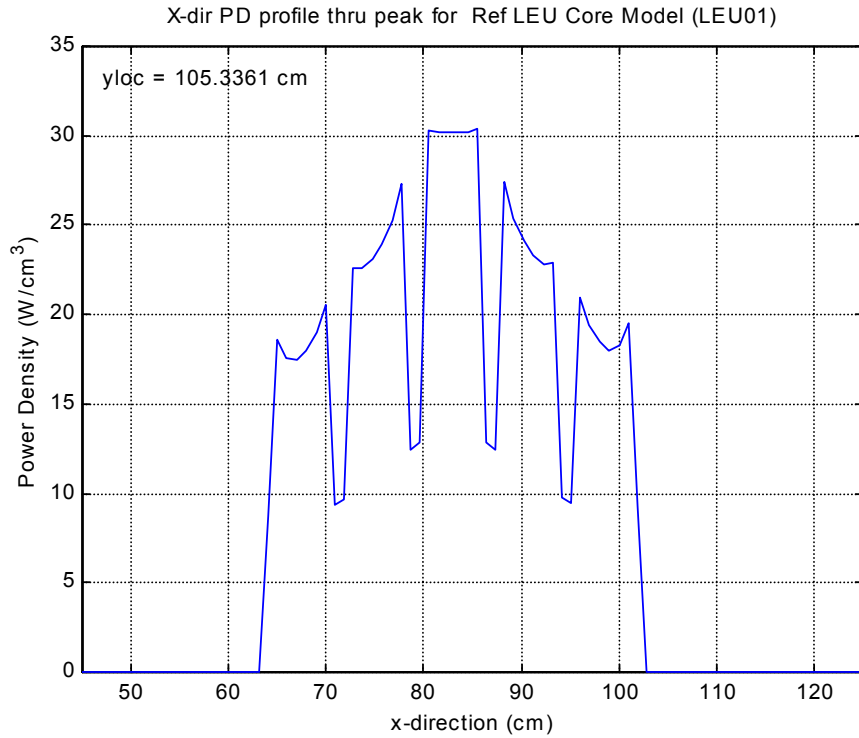


Fig. 5 Power density profile through peak in x-direction (LEUV01 core).

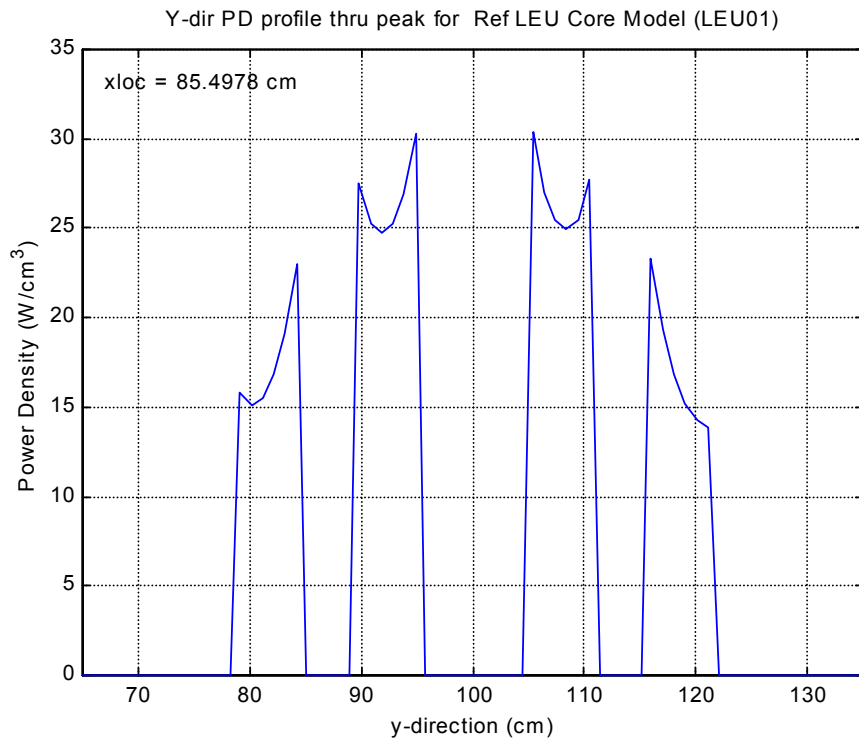


Fig. 6 Power density profile through peak in y-direction (LEUV01 core).

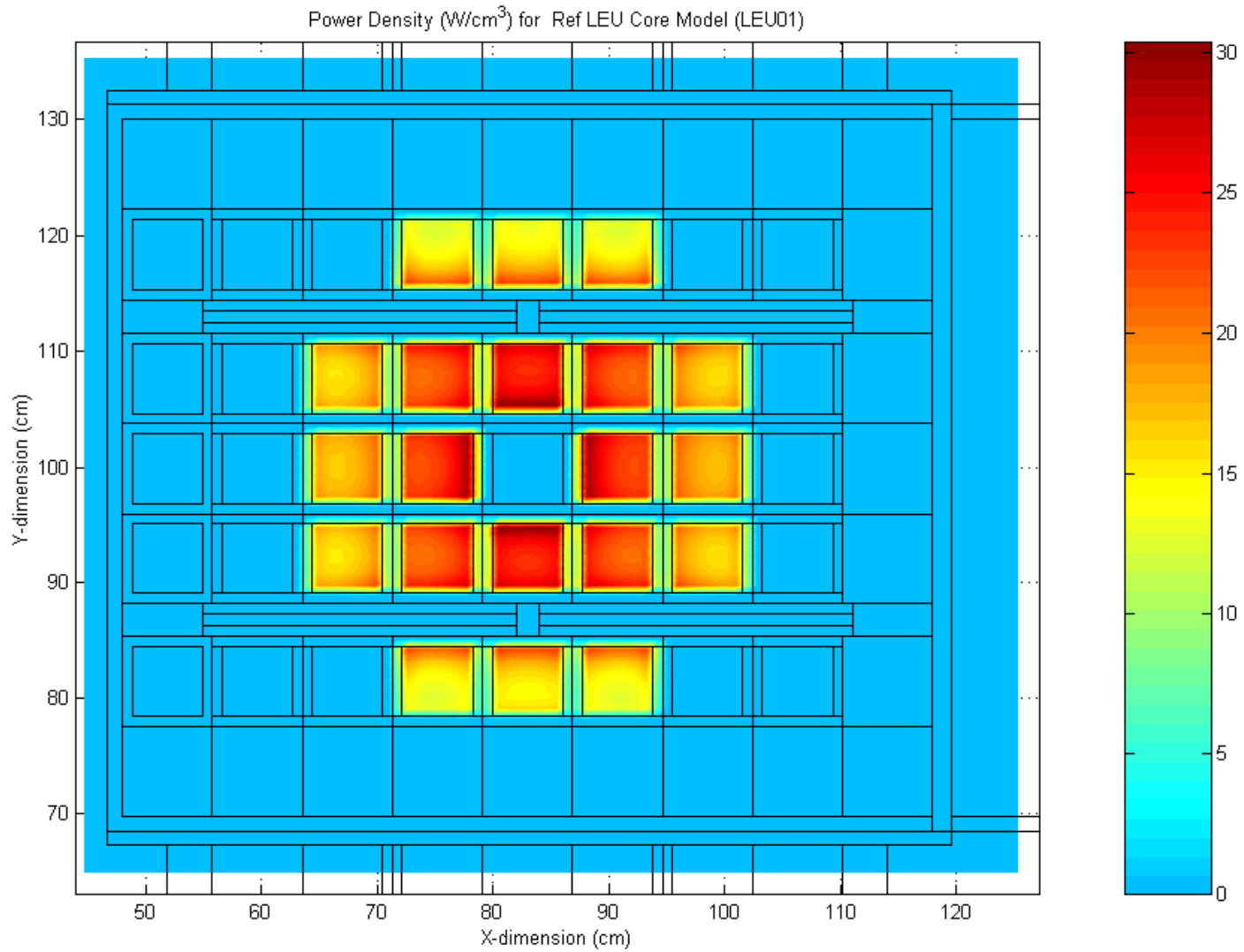


Fig. 7 Two-dimensional power density distribution in the LEU01 model.

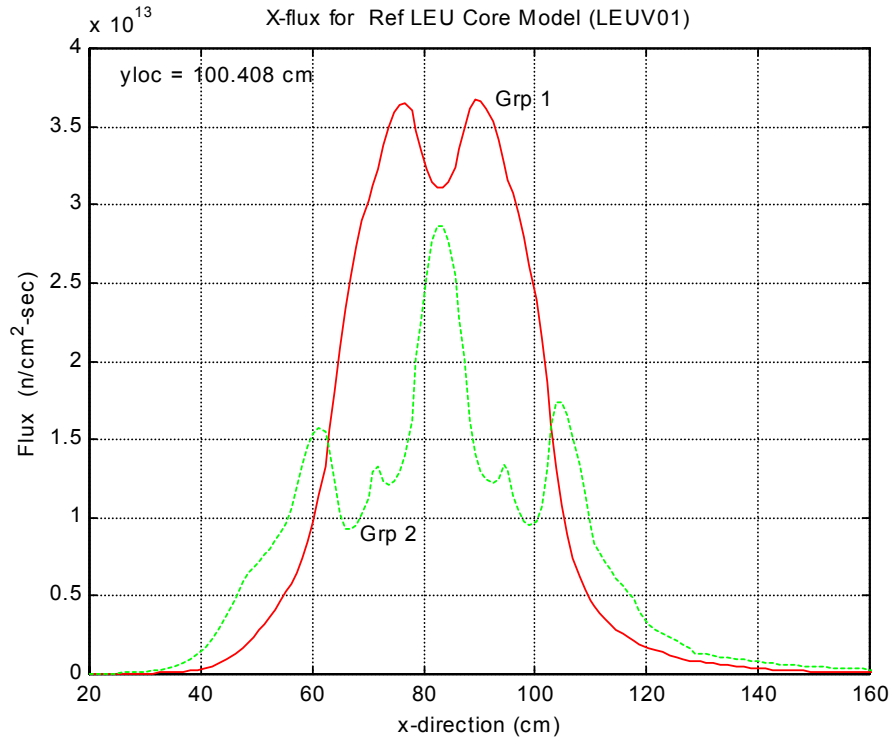


Fig. 8 X-directed flux profiles through the flux trap (LEUV01 model linear scale).

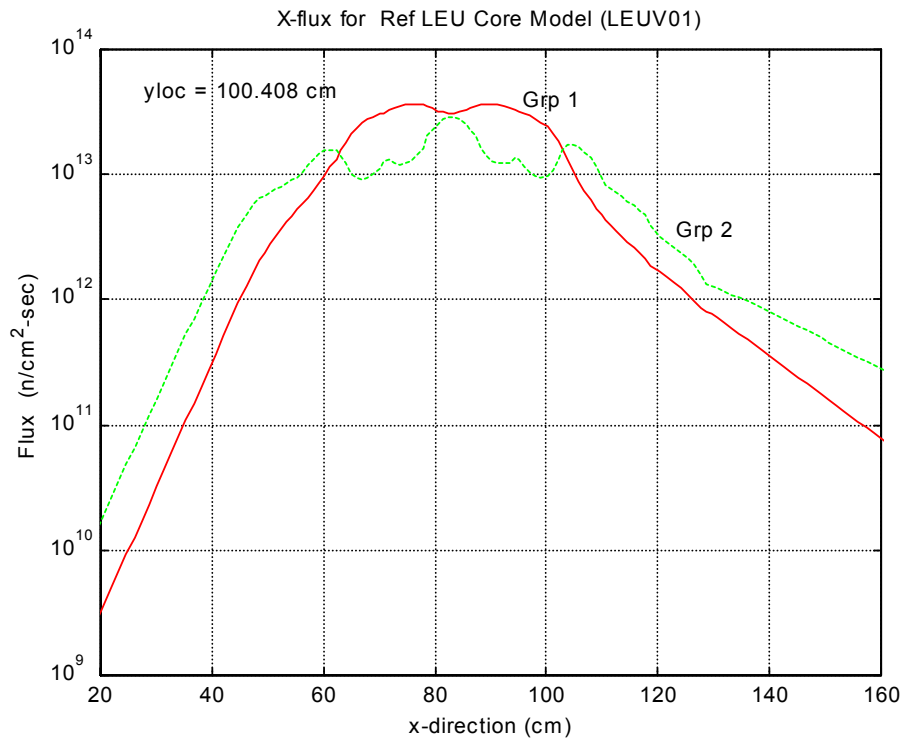


Fig. 9 X-directed flux profiles through the flux trap (LEUV01 model log scale).

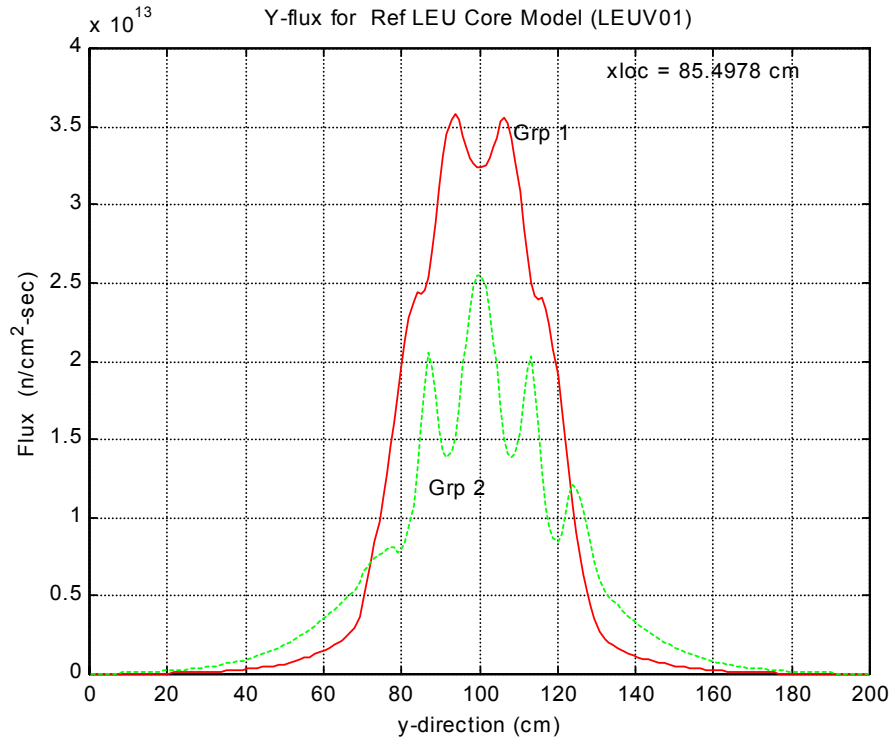


Fig. 10 Y-directed flux profiles through the flux trap for the LEUV01 model.

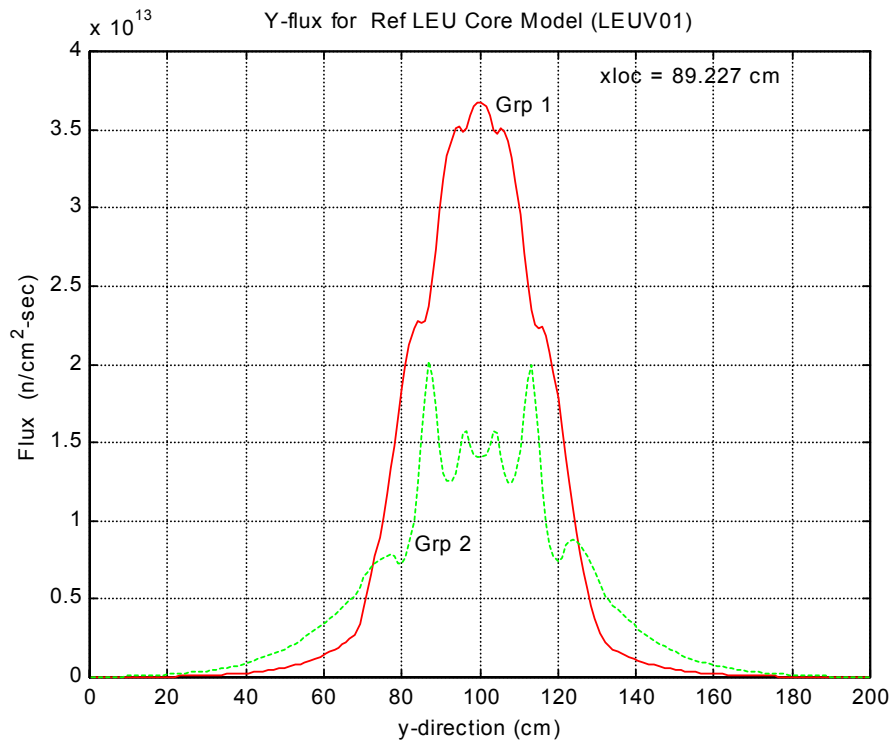


Fig. 11 Y-directed flux profiles through grid location #6 for the LEUV01 model.

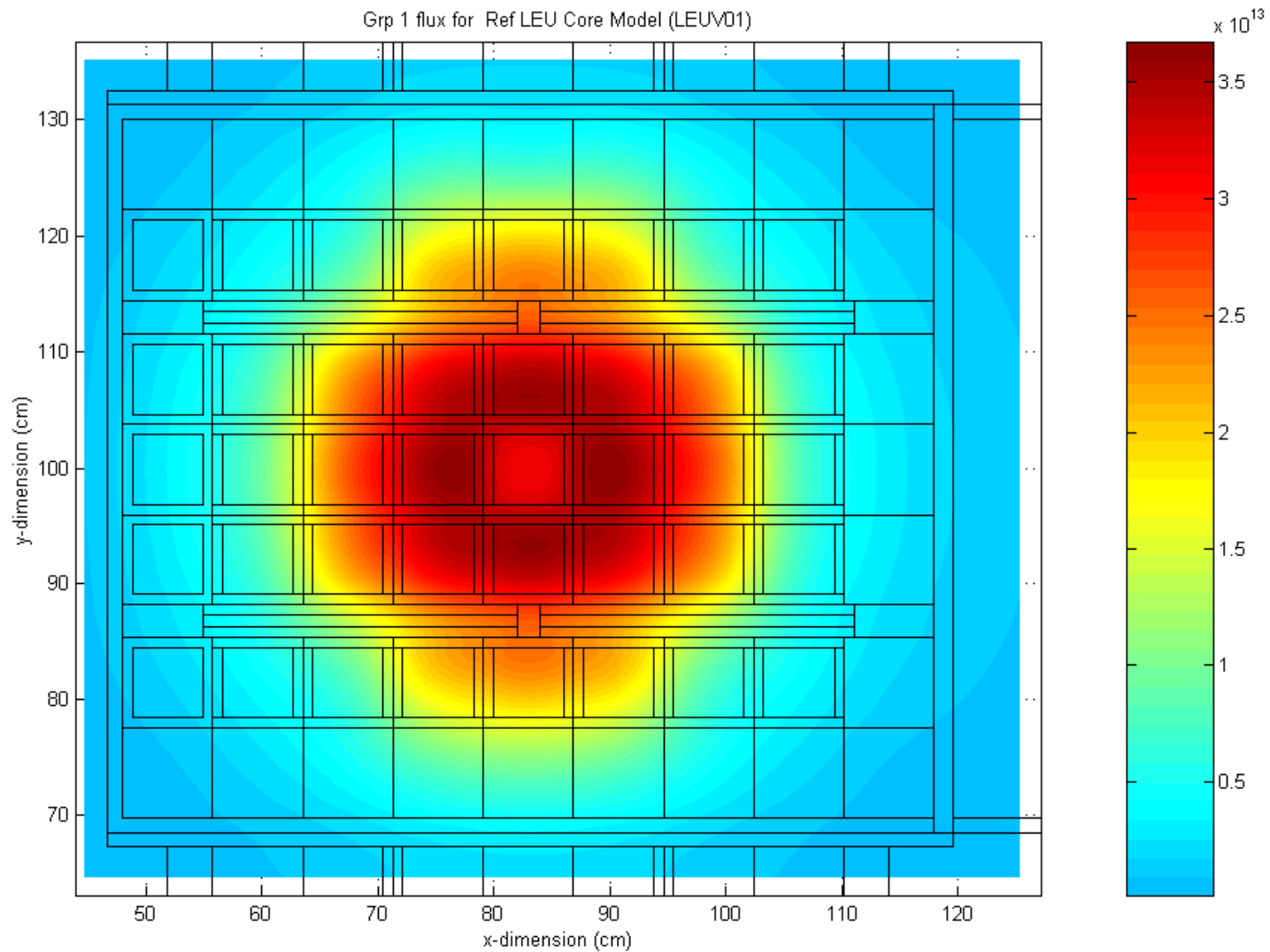


Fig. 12 Gray-scale/color representation of the two-dimensional fast flux distribution in the LEUV01 model.

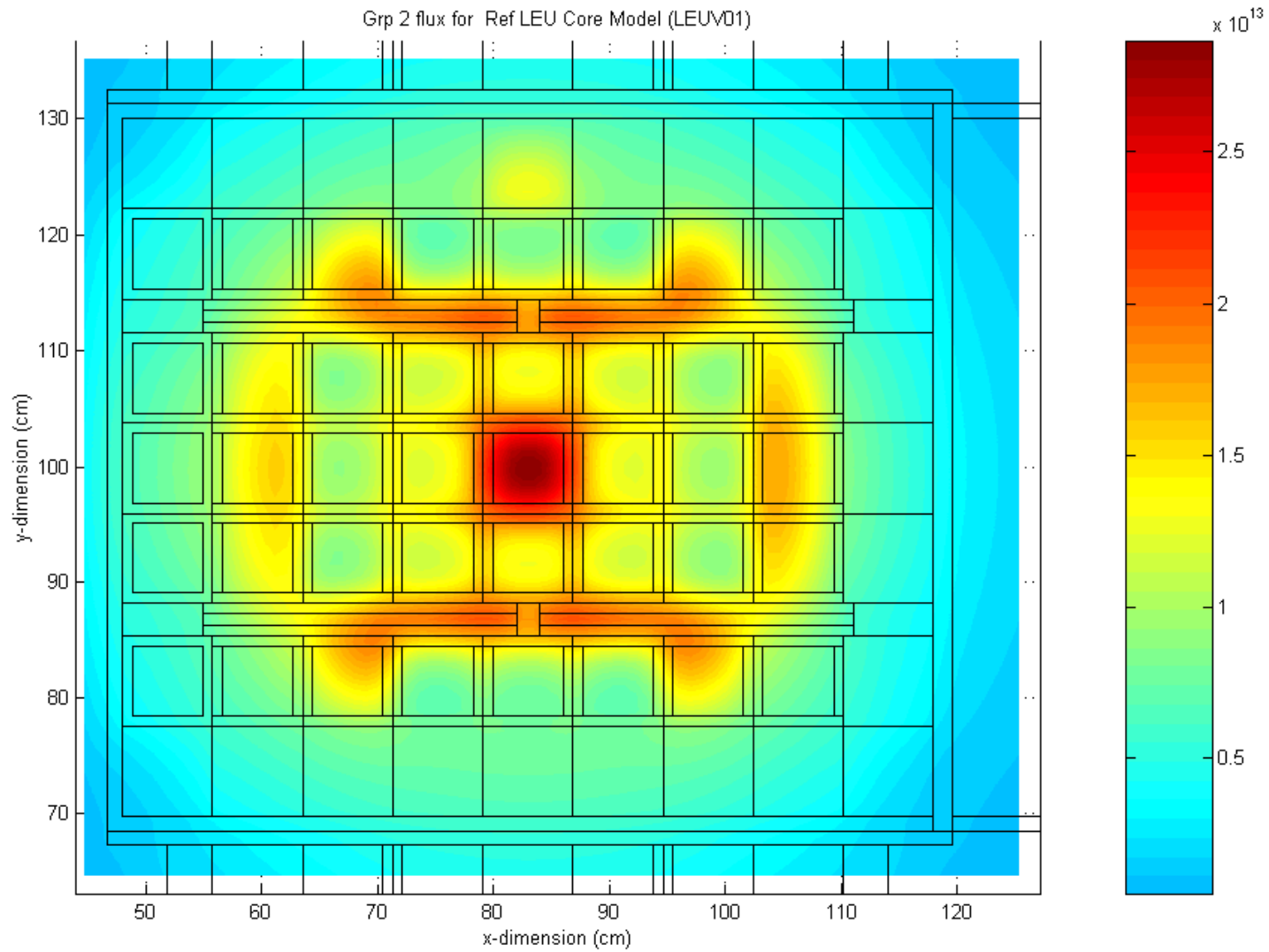


Fig. 13 Gray-scale/color representation of the two-dimensional thermal flux distribution in the LEUV01 model.

Summary

This report documents the general procedure used to obtain broad-group cross sections for the LEU-fueled UMass-Lowell Research Reactor (UMLRR). It also overviews, in some detail, the individual assembly models and the 1-D core model used in the actual generation of the 2-group VENTURE library and the coupled neutron-gamma 67-group library for DORT computations. Finally, this report also describes the development of a 2-D XY computational model for the reference LEU core configuration and it shows that the VENTURE results are as expected -- thus providing preliminary verification of the whole cross section generation process.

It is expected that the data and models developed here will serve as a good foundation for future analytical analyses of the LEU-fueled UMLRR. The LEU fuel should be on site later this year and it is hoped that the data and models developed as part of this work will be used with additional studies to help support the startup and operation of the new configuration. The preliminary results generated here should also help characterize the overall behavior of the new core and help establish analytical models for support of various experimental uses of the facility. More work is clearly needed in this area, but the present study represents a first step in the right direction.

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