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THE UNIVERSITY OF TEXAS AT AUSTIN

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December 19, 2012

ATTN: Document Control Desk,
U.S. Nuclear Regulatory Commission,
Washington, DC 20555-0001

Allan Jason Lising
Project Manager
Division of Policy and Rulemaking
Research and Test Reactors Licensing Branch

SUBJECT: Docket No. 50-602, Request for Renewal of Facility Operating License R-129

REF: UNIVERSITY OF TEXAS AT AUSTIN - REQUEST FOR ADDITIONAL INFORMATION REGARDING THE LICENSE RENEWAL REQUEST FOR THE NUCLEAR ENGINEERING TEACHING LABORATORY TRIGA MARK II NUCLEAR RESEARCH REACTOR (TAC NO. ME7694)

Sir:

Attached are *Report on Neutronic Analysis for the UT TRIGA Reactor* and *Historical UT TRIGA Core Data*, which address some of the items in the referenced Request for Additional Information, including:

RAI 7: The guidance in NUREG-1537 Section 4.5, "Nuclear Design," requests that the licensee provide a detailed description of analytical methods used in the nuclear design, including computer codes used to characterize technical parameters pertaining to the reactor. UT SAR Section 4.5 states that the "characteristics and operating parameters of this reactor have been calculated and extrapolated using experience and data obtained from existing TRIGA reactors as bench marks in evaluating the calculated data." Please provide comprehensive analysis of UT TRIGA behavior. Please describe the methods used for steady state neutronic (steady-state and kinetics) and thermal-hydraulic analysis and include comparisons with UT TRIGA measurements that demonstrate that those methods are appropriate to analyze the limits imposed by the UT TRIGA TS.

RESPONSE:

RAI 7 is partially addressed in Attachment 1, specifically the neutronic portion of the RAI, by Sections 2 (describing the SCALE computer code suite) and 4 (modeling the UT TRIGA reactor in SCALE) of the attachment. Thermo hydraulic modeling is not addressed in this response.

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Validation of methodology and modeling is provided in Section 6.

RAI 8: The guidance in NUREG-1537 Section 4.5.1, "Normal Operating Conditions," requests that the licensee define the limiting core configuration (LCC) which defines the highest power densities and temperatures achievable.

RAI 8.1: UT SAR Section 4.5.1 discusses an "operational core of 85 fuel elements, 3 fuel followed control rods, and one air followed control rod is to be arranged in 5 rings." The NRC staff notes that elsewhere in the UT SAR, references are made to core configurations of 90 (page 4-36), 116 (page 4-55), 109 and 114 (page 4-57), 81 (page 13-7), 83 (page 13-2), 85 (pages 13-26 and 13-29), and 100 (page 13-33) fuel elements. For the UT TRIGA licensed power of 1,100 kW, please identify the LCC. Please provide schematic drawings showing the location of fuel elements, control rods, and other components installed in the lettered-and-numbered lattice positions. For fuel elements provide a cross reference to fuel element serial numbers and their accumulated burnup. Please provide all technical parameters and conclusions supplied for normal operation, accident analysis, and dose estimates using the LCC.

RESPONSE:

The limiting core configuration (LCC) is identified in Section 5 of the Attachment 1.

Schematics showing the location of core components are provided in Figs. 1-6, and 15 of Attachment 1. The LCC analysis assumes fresh fuel, and does not assume specific elements or burnup for initial operation. Burnup is discussed for the 1992 core in Sections 1.3 and 6.7, and illustrated on Fig. 3.

Technical parameters for normal operation, accident analysis, and dose estimates are provided in Section 5 of Attachment 1. The limiting core configuration is established, and the parameters are extended from the minimum number of fresh fuel elements that comprise the limiting core configuration through a fully-fueled core with burnup at end of core life.

Fuel element serial numbers, in-core locations, and accumulated burnup values are provided in Attachment 2.

RAI 8.2: Please provide analyses that quantify the effects of fuel burnup, plutonium buildup, and the effect of fission products on the UT TRIGA LCC.

RESPONSE:

Effects of fuel burnup are discussed in Sections 5.2.1 and 5.2.3 of Attachment 1.

RAI 8.3: Please provide the technical parameters including analysis of "reactor kinetic behavior, basis reactor criticality, control rod worth, definition of the limiting core configuration (LCC), [etc.]" (NUREG-1537, Section 4.5.1). State whether the comparison of calculated and measured values demonstrates acceptable model development.

RESPONSE:

Please contact me by phone at 512-232-5373 or email whaley@mail.utexas.edu if you require additional information or there is a problem with this submittal.

Thank you,

P. M. Whaley
Associate Director
Nuclear Engineering Teaching Laboratory
The University of Texas at Austin

I declare under penalty of perjury that the foregoing is true and correct.
Executed on December 12, 2012

Steven R. Biegalski
NETL Director

ATT: 1. Report on Neutronic Analysis of the UT TRIGA Reactor
2. Historical UT TRIGA Core data

The limiting core configuration is identified in Section 5 of Attachment 1, with considerations of critical mass in 5.1.1 and 5.1.2, kinetics parameters in Section 5.21, control rod worths in 5.2.6.

Validation of methodology and modeling is provided in Section 6.

RAI 9: The guidance in NUREG-1537 Section 4.5, "Nuclear Design," requests that the licensee provide a detailed description of analytical methods used in the nuclear design, including computer codes used to characterize technical parameters pertaining to the reactor. UT SAR Section 4.5 states that the "characteristics and operating parameters of this reactor have been calculated and extrapolated using experience and data obtained from existing TRIGA reactors as bench marks in evaluating the calculated data." Please provide comprehensive analysis of UT TRIGA behavior. Please describe the methods used for steady state neutronic (steady-state and kinetics) and thermal-hydraulic analysis and include comparisons with UT TRIGA measurements that demonstrate that those methods are appropriate to analyze the limits imposed by the UT TRIGA TS.

RESPONSE:

RAI 7 is partially addressed by Attachment 1, specifically the neutronic portion of the RAI by Sections 2 (describing the SCALE computer code suite) and 4 (modeling the UT TRIGA reactor in SCALE) of the attachment. Thermal-hydraulic modeling is not addressed in this response.

Validation of methodology and modeling is provided in Section 6.

RAI 10: UT SAR Table 4.21, "Limiting Core reactivity," displays Reference and Current control rod worths. In Table 4.14, please explain the origin of the values listed under the "Reference" column. Given the difference between the "Reference" and "Current" values of excess reactivity and shutdown margin, which values are being used in the UT TRIGA TS.

RESPONSE:

The Table will be deleted, with reference values used in model validation (Section 6 of Attachment 1) and reactivity values for the LCC with burnup provided in Section 5.2.6.

RAI 12: UT SAR Section 4.5.4, Subsection B provides Figure 4.22 for the power within a fuel element. The NRC staff notes that the power distribution in the figure continues to the center of the fuel element indicating that this curve is not applicable to stainless steel-clad fuel that has a zirc rod in the center. Please confirm and revise accordingly.

RESPONSE:

Updated power distribution analysis is reported in in Section 5.2.2 of Attachment 1.

We respectfully request an additional 90 days to complete response to the remaining items.

ATTACHMENT 1:
REPORT ON NEUTRONIC ANALYSIS FOR THE UT TRIGA REACTOR

The UT TRIGA critical mass of the original UT TRIGA reactor core configuration is compared with the critical mass required in prototypical cores. The computer codes used in this analysis is described. The geometry of the UT TRIGA core is identified. The representation of the core geometry and materials in modeling within the program is described. The results of calculations using the model to characterize the UT TRIGA reactor are summarized. Finally, evidence demonstrating validity of the model in characterizing the UT TRIGA reactor is provided.

The UT TRIGA reactor core uses a triangular pitch, composing a hexagonal geometry as shown in Fig. 1. Core positions are indexed as rings (A through G), with index numbers increasing for sequential positions. Neutronic analysis was performed for the UT TRIGA reactor configured with three standard fuel follower control rods located in positions C01, C07, D06, and D14. Position A01 (central thimble) does not contain a standard fuel element. Positions G32 and G34 are reserved for a neutron source and an in-core pneumatic terminal. The A and B rings are within a removable assembly that allows insertion of large experiments; because of the associated reactivity deficit, removing the fuel in these positions severely limits potential operation. Fuel element positions E03, E04, F03, F04, F05, G04 and G05 can similarly be removed with full power operations possible. Two smaller removable "3-element" assemblies are located at the D17/E22/E23 and E11/F12/F14 positions.

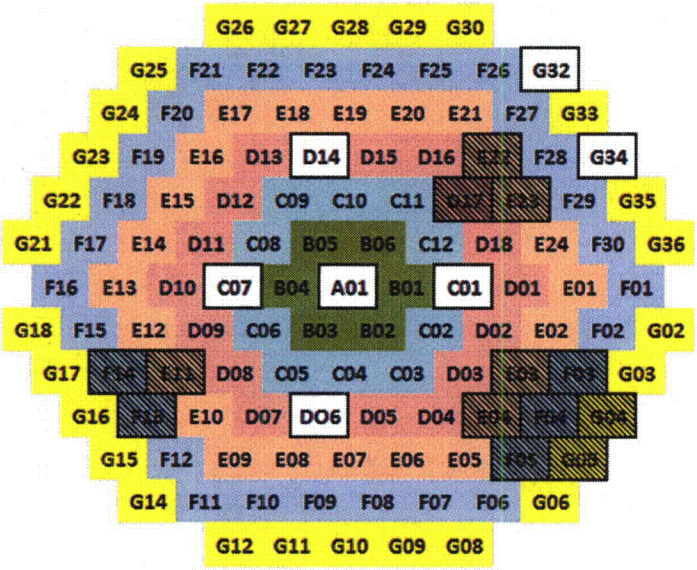

Figure 1, UT TRIGA Reactor Core

1.0 COMPARISON WITH HISTORICAL CRITICAL MASS DATA

1.1 CRITICAL MASS OF HISTORICAL GENERAL ATOMICS TRIGA REACTORS

The cylindrical GA TRIGA Mark I reactor achieved critical condition with 54 standard fuel elements (8.5% weight uranium, 20% enriched, 1.94 kg ^{235}U) and four water filled control rod positions. Core positions not occupied by fuel elements used graphite "dummy" rods for reflection; removing the graphite rods increased critical mass about 25%. The GA Advanced TRIGA Prototype reactor achieved critical

condition with three fuel follower control rods, 1 transient rod, and 75 standard fuel elements (2.7 kg 235U). The GA TRIGA Mark III reactor achieved critical condition with four fuel follower control rods (including a fuel element in the "A" ring) and 56 standard fuel elements (2.24 kg 235U)

1.2 CRITICAL MASS OF THE 1992 UT TRIGA REACTOR

Initial criticality for the UT TRIGA reactor at the Nuclear Engineering Teaching Laboratory (NETL) was accomplished on 02/13/1992. Criticality was attained with 3 fuel follower control rods (fully withdrawn, i.e., fuel fully inserted) and 56 fuel rods (including two instrumented fuel elements). Total mass of ^{235}U was 2.12 kg in the standard fuel elements and 94.46 g in the three fuel followers.

The UT TRIGA reactor was configured with both water voids and graphite rods in non-fueled positions. With the exception of fuel follower elements (i.e., fuel follower control rods), the 1992 University of Texas (UT) TRIGA reactor core was composed of fuel elements with power history at the previous UT TRIGA (located on main campus in Taylor Hall) and/or a General Atomics facility. These lightly-burned fuel elements decayed approximately 1 year prior to use at the current location. The new UT TRIGA core included three new fuel follower elements with fresh fuel. Critical fuel loading is displayed in Fig. 2, with labels:

- CT for the central thimble (water void)
- SFE for standard fuel rods
- TC for instrumented fuel elements
- GR for graphite rods
- S for the neutron source
- WV for water voids
- S2, S1 and RR for the fuel follower control rods (Shim 1 and 2, Regulating Rod)
- TR for the transient rod

Although the pitch is hexagonal, positions are considered to be in "rings." The B ring is shaded green in Fig. 2, C ring light blue-green, D ring rose, E ring light brown, F ring light blue, and the G ring yellow.

A direct comparison between critical masses of the initial 1992 UT TRIGA core and the historical GA TRIGA cores is complicated by (1) differences in reflection (graphite rod and water void configurations), (2) previous power history for the standard fuel elements, and (3) fundamental difference in core geometry. Nevertheless, the 2.2 kg ^{235}U in the UT TRIGA compares well with the 1.94 kg (approximately 2.4 kg water moderated) of fresh elements required in the original GA TRIGA reactor.

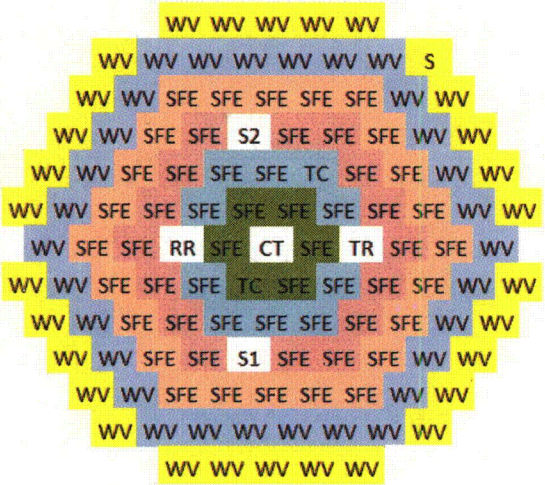

Figure 2, Initial UT TRIGA Core

1.3 OPERATIONAL LOADING OF THE 1992 UT TRIGA REACTOR

Fuel was loaded in the UT TRIA reactor to support operation at 1.1 MW on 03/16/1992. The core contained 84 standard and 3 fuel follower elements with 3.35 kg ^{235}U. Operational fuel loading is displayed in Fig. 3, using the same labeling as in Fig. 2.

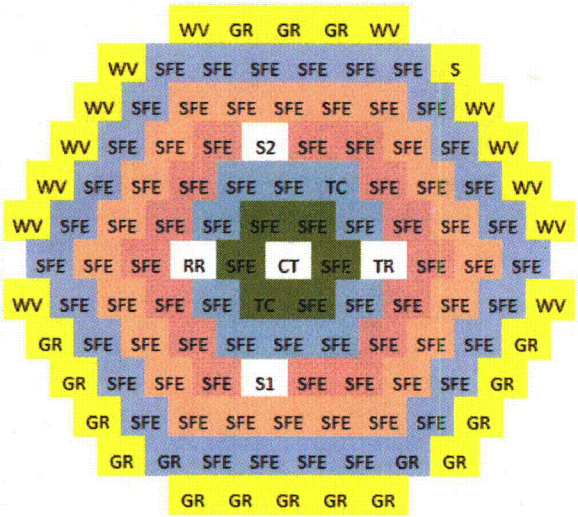

Figure 3, Operational 1992 Core

Power history for the standard fuel elements in this configuration included 7.91 MWD generated at Taylor Hall in 46 elements, and 41.07 MWD generated at General Atomics facilities in 56 elements (15 elements had burn from both facilities). With the exception of the fuel followers and standard fuel elements in positions C11 and D18, fuel in the B ring through the D ring (B01 through D17) had an average burn of 0.255 MWD per element, with a standard deviation of 0.037 MWD. With the exception of standard fuel elements in elements in core position F30, the standard fuel elements in the E and F rings and positions C11 and C18 had an average burnup of 0.710 MWD with a standard deviation of 0.026 MWD. The element in position F30 was an outlier, received from GA with almost a gram of ^{235}U depleted. Fuel elements in the inner ring with the lower average burnup are marked on Fig. 4 with a

light shade of violet, and higher burn elements with a darker shade of purple. (Water voids are blue, graphite element's gray.)

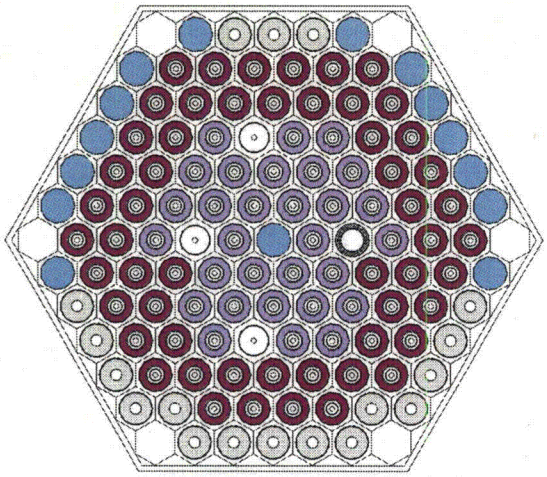

Figure 4, Loading of Previously Burned Elements

Control rod calibrations are accomplished under surveillance procedure SURV-6, and excess reactivity determinations are documented with SURV 3. The initial control rod reactivity worth calibration was completed on 03/31/1992, although excess reactivity was not determined until completion of initial testing in July when both SURV-3 and SURV-6 were performed. Surveillance data is provided in Table 1.

Table 1: 1992 UT TRIGA REACTIVITY ($), SURV-6 & SURV 3

PARAMETER	03/25/1992	07/23/1992
EXCESS	NA	6.38
REG ROD	4.59	4.08
TR ROD	3.34	3.26
SHIM 2	3.46	3.30
SHIM 1	3.32	3.17

2.0 NUCLEAR PHYSICS MODELING

Neutronic and gamma transport modeling for the UT TRIGA reactor is based on calculations with SCALE[1] 6.1. The SCALE code package is a comprehensive modeling and simulation suite for nuclear safety analysis and design developed under the sponsorship of the U.S. Nuclear Regulatory Commission (NRC) and maintained by Oak Ridge National Laboratory (ORNL). SCALE integrates a suite of programs and routines that pass, format, and process relevant information to support a variety of nuclear-system calculations based on standard nuclear data sets. SCALE control sequences select and integrate the programs and modules required for the type of calculation (and any selected options). For the UT TRIGA reactor, SCALE is used to calculate:

- Critical mass
- Core Radial Peaking Factor
- Physics parameters & Flux density

[1] ORNL/TM-2005/39 Version 6.1, Scale: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design, Oak Ridge National Laboratories

- Fuel Element Axial and Radial Peaking Factors
- Fission Product Poisons
- Transuranic buildup
- Fuel temperature reactivity deficit
- Water temperature reactivity deficit
- Control Rod worth
- Excess reactivity
- Shutdown Margin
- Effects of Experiments
- Effects of Burnup
- Accident source Terms

The SCALE Material Information Processor (MIP) is based on free-form engineering parameter input that specifies nuclear data to be selected from cross section libraries for use in control and functional modules. Physical modeling in SCALE is accomplished by the Generalized Geometry Package (SGGP); SGGP uses a specific set of surface geometries as the framework, with volumes developed by defining orthogonal boundaries for surface extension. Analysis of the UT TRIGA reactor is based on the T-6 (TRITON) depletion sequence, calculating information of interest based on burnup and fission product generation. The T-6 sequence is based on a 3-D Monte-Carlo transport code, KENO-VI, in conjunction with an isotope buildup and decay code, ORIGEN. Other than (1) a broad description of the code and (2) identification of subroutines and codes used in processing, questions regarding methodology of calculation should be referred to the SCALE manual[2].

2.1 SCALE 6.1, T-6 Depletion Sequence

The T-6 sequence is a specification in the TRITON computer code. TRITON is a multipurpose SCALE control module for transport, depletion, and sensitivity and uncertainty analysis with sequences to perform various calculations of interest. TRITON provides automated, problem-dependent cross-section processing followed by multigroup transport calculations. TRITON functions with the ORIGEN (Oak Ridge Isotope GENeration) depletion module to predict isotopic concentrations, source terms, and decay heat, as well as generating few-group homogenized cross sections for nodal core calculations. Isotopic concentrations are calculated by ORIGEN at burnup intervals specified in the input. The results of the ORIGEN calculations are used to modify materials specified in the input file for subsequent calculations.

A T-6 calculation involves three major steps. KENOVI, a 3-D Monte Carlo code, is used to determine region-specific multi-group fluxes, cross sections, and power generated by the material specified for depletion. From the information developed in the first step, COUPLE develops a "response function," a one-group cross section library. Using the one-group cross sections and the material power, ORIGEN performs depletion (fission and transmutation) calculations to determine material composition following the burnup interval. Material composition is transferred by KMART6 into KENOVI for the subsequent depletion calculation. The control modules and codes invoked by the T-6 depletion sequence include:

1. CRAWDAD creating a continuous energy library for use by CENTRM PMC.

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[2] *Scale: 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-785

2. BONAMI performing resonance self-shielding calculations for nuclides that have Bondarenko data associated with their cross sections.
3. WORKER creating an AMPX working format library from a master format library.
4. CENTRM using the pointwise continuous cross-section library and a cell description to create a pointwise continuous flux spectrum.
5. PMC using the pointwise continuous flux spectrum created in CENTRM, to collapse pointwise continuous cross sections into a set of multigroup cross sections.
6. CAJUN combining homogenized point cross-section libraries.
7. CHOPS computing pointwise flux disadvantage factors and creates homogenized point cross sections.
8. KENO-VI calculating k_{eff} of a 3-D system using the Monte Carlo method, and developing flux spectral flux distribution.
9. KMART6 performing the flux post-processing for the KENO/VI sequence
10. COUPLE generating a one-group cross-section library for each depletion material from region-averaged multigroup cross sections and multigroup fluxes
11. ORIGEN calculating (in a matrix exponential expansion model) time-dependent concentrations, activities, and radiation source terms for a large number of isotopes simultaneously generated or depleted by neutron transmutation, fission, and radioactive decay.
12. OPUS program (optional) producing a condensed output file and data formatted for plotting from output generated by the ORIGEN-S code that computes reactor fuel depletion, activation and fission- product buildup, and radioactive decay.

2.2 SCALE OPTIONS IN TRIGA CALCULATIONS

SCALE is a well-established tool for reactor analysis, with a set of default parameters to support commercial nuclear power reactors. Some default options create program conflicts in TRIGA analysis unless disengaged, and relaxing some of the defaults may improve processing time. Specification of parameters and the information desired for the UT TRIGA reactor leads to specific strategies.

2.2.1. Default and Override Parameters

Flux weighted, material averaged cross sections are calculated for the midpoint of burnup intervals. If there are large changes in materials over the burnup interval, the material averaged cross sections may not accurately represent the system cross section over the full range of the burnup interval. Because ORIGEN was originally designed to be used in power reactor systems operating at steady state power levels for very long periods of time, there is a default limit of 20 days on burnup intervals that ensure processed cross sections represent the material adequately. The burnup for a TIRGA reactor is generally not continuous or at a flux as high as a commercial reactor; therefore default may be overridden by substituting an alternate maximum number of days with the command "MAXDAYS=N."

Trace quantities of fission products may individually be low enough to not affect calculations, but in aggregate may have a significant effect. By default, TRITON automatically adds trace quantities of a set of nuclides to all fuel materials that have been determined to be important in the characterization of spent fuel. The nuclides are not explicitly listed in the material specification of the input file. The default set of added nuclides can be augmented to improve accuracy (with other defined groups of nuclides, up to about 380 total nuclides) using a command "ADDNUX=N" where N is a number that specifies sets of nuclides to be used. The default TRITON depletion sequences setting (N=2) adds 94 nuclides. Unfortunately, one of the nuclides in the augmented set is hydrogen. Since zirconium-hydrogen cross sections are explicit in the input file, hydrogen included automatically by ADDNUX

creates a conflict by using two different cross sections for a single nuclide. This conflict is alleviated by setting ADDNUX=0 and explicitly reproducing the augmented data set in the input, without hydrogen-1 as a fission product. For fresh fuel, the atom-density of all additional isotopes set to a minimum (1E-20) results in a reactivity effect of about $0.04.

The number of input lines required to support fission product inventory is excessive. It is convenient to specify the isotopes in a separate file, with the file read into the SCALE input. A shell (with a copy command) at the beginning of the input file copies the file into the temporary directory. The material file is read into material specification with a right bracket and the file name. An example of the use of the shell and copy command to read the text file NEWFUEL.TXT into the temporary directory with the name FILE1.TXT is provided below; the read statement that is inserted in the material specification section is shown in a separate section.

```
=shell
 copy "%RTNDIR%\NEWFUEL.TXT" "%TMPDIR%\FUEL1.TXT"
end
<FUEL1.TXT
```

The number of nuclides added to improve calculations increases processing time. In cases where parameters of interest are not related to burnup, the "INFDCUTOFF=X" specification compares the material cross section to the total system cross sections. If the system cross section is greater than the material cross section by the specified value, the material is not used in transport calculations. Removing isotopes that have little impact relative to system cross section improves processing time.

ORIGEN was originally developed for power level applications, and uses default mass normalization to a metric ton of heavy metal (i.e. uranium). While this is adequate for determining densities of materials such as fission products in large cores, it de-emphasizes fuel depletion in smaller cores. The "LENGTH=" option provides a means of normalizing to an arbitrary mass. This option was originally used with 2D models where scale is synonymous with length, and has been extended to 3D in the current version of SCALE. If the LENGTH is set to 1, normalization is the mass of the model based on fuel mass as calculated from volume and density.

2.2.2 Strategies

Strategies developed for the UT TRIGA SCALE model are considered in three sections. First, strategies for acquiring data used in calculations are discussed. Second, strategies related to how the geometry is developed are discussed. Finally, development of the material specifications is discussed.

A. Reporting and Calculations

OPUS is a SCALE report module for specific parameters. The key word "symnuc" defines nuclides of interest. Parameters used in TRIGA calculations include material density (GPERCM is the specification for mass density, ATOMS the specification atomic density in units of atoms per barn-cm), mass (grams), fractional material absorptions (ABSORB), radioactivity (CURIES), gamma spectrum (PHOTONS), and decay heat (WATTS). Multiple reports for different set of nuclides and parameters are generated using a "new case" delimiter within a single calculation. Application of the specifications is provided in Table 2A.

Table 2A: OPUS Specifications

PURPOSE	Title	symmnuc	units	time
^{235}U, ^{238}U mass burned	URANIUM MASS	u-235 u-238	grams	days
	MATERIAL SET 1	see Table 2.B	atoms	days
Fuel material specification	MATERIAL SET 2	see Table 2.C	atoms	days
	MATERIAL SET 3	see Table 2.D	atoms	days
Reactivity Effects	ABSORPTIONS	u-235 u-238 pu-238 pu-240 pu-241 pu-242 xe-131 xe-133 xe-135 sm-147 sm-150 sm-151 sm-152 sm-153		hours
Loss of Pool Water Accident Source term	DECAY HEAT	u-236 pu-238 pu-240 pu-241 pu-242	watts	days
Loss of Pool Water Accident Source term	GSPEC			
Fission Product Inventory MHA Source term	CURIES			

For calculations used to generate material specification, nuclides specified in the ADDNUX routine (Table 4A/B/C) are reported; a limit on the number of nuclides prompted splitting the total set into three separate specifications. In modeling cores through operation, burnup is simulated to approximate the average power history for the core. The results of calculations of isotope concentrations following simulated operation are used as material specification for calculations of burned cores. Subsequent calculations are used to benchmark against reference reactivity values, characterize reactivity effects from burnup, or characterize the reactivity effect of specific materials (i.e., comparing k_{eff} from cases where the nuclide exists and is deleted).

Table 2B, MATERIAL SET 1

u-235	u-238	u-232	u-233	u-234	u-236	u-237	h-2	h-3	b-10
b-11	n-14	n-15	o-16	o-17	f-19	p-31	s-32	i-127	i-129
i-130	i-131	i-135	w-182	w-183	w-184	w-186	y-89	y-90	y-91
zr-93	ag-107	ag-109	ag-111	al-27	am-243	as-75	au-197	ba-134	ba-135
ba-136	ba-137	ba-138	ba-140	be-9	bi-209	bk-249	br-79	br-81	cd-106
cd-108	cd-110	cd-111	cd-112	cd-113	cd-114	cd-115m	cd-116	ce-140	ce-141
ce-142	ce-143	ce-144	cf-249	cf-250	cf-251	cf-252	cm-241	cm-242	cm-243
cm-244	cm-245	cm-246	cm-247	cm-248	co-59	cs-133	cs-134	cs-135	cs-136
cs-137	dy-160	dy-161	dy-162	dy-163					

Table 2C, MATERIAL SET 2

dy-164	er-166	er-167	es-253	eu-151	eu-152	eu-153	eu-154	eu-155	eu-156
eu-157	gd-152	gd-154	gd-155	gd-156	gd-157	gd-158	gd-160	ge-72	ge-73
ge-74	ge-76	he-3	he-4	hf-174	hf-176	hf-177	hf-178	hf-179	hf-180
ho-165	in-113	in-115	kr-78	kr-80	kr-82	kr-83	kr-84	kr-85	kr-86
la-139	la-140	li-6	li-7	lu-175	lu-176	mn-55	mo-100	mo-92	mo-94
mo-95	mo-96	mo-97	mo-98	mo-99	na-23	nb-93	nb-94	nb-95	nd-142
nd-143	nd-144	nd-145	nd-146	nd-147	nd-148	nd-150	np-237	pa-231	pa-233
pd-102	pd-104	pd-105	pd-106	pd-107	pd-108	pd-110	pm-147	pm-148	pm-148m
pm-149	pm-151	pr-141	pr-142	pr-143	pu-236	pu-238	pu-239	pu-240	pu-241

Table 2D, MATERIAL SET 3

pu-242	pu-243	pu-244	rb-85	rb-86	rb-87	re-185	re-187	rh-103	rh-105
ru-100	ru-101	ru-102	ru-103	ru-104	ru-105	ru-106	ru-96	ru-98	ru-99
sb-121	sb-123	sb-124	sb-125	sb-126	se-74	se-76	se-77	se-78	se-80
se-82	sm-144	sm-147	sm-148	sm-149	sm-150	sm-151	sm-152	sm-153	sm-154
sn-112	sn-114	sn-115	sn-116	sn-117	sn-118	sn-119	sn-120	sn-122	sn-123
sn-124	sn-125	sn-126	sr-84	sr-86	sr-87	sr-88	sr-89	sr-90	ta-181
ta-182	tb-159	tb-160	te-122	te-123	te-124	te-125	te-126	te-127m	te-128
tc-99	te-120	te-129m	te-130	te-132	th-230	th-232	xe-124	xe-126	xe-128
xe-129	xe-130	xe-131	xe-132	xe-133	xe-134	xe-135	xe-136	zr-95	

Table 2E: Nuclides Identified in ADDNUX sets, Lacking Cross Section Data

ti-50	ti-49	ti-48	ti-47	ti-46	th-234	th-233	th-229	th-228	th-227
sn-113	si-30	si-29	si-28	se-79	sc-45	ra-226	ra-225	ra-224	ra-223
pu-246	pb-208	pb-207	pb-206	pb-204	pa-232	np-239	np-238	np-236	np-235
ni-64	ni-62	ni-61	ni-60	ni-59	ni-58	mg-26	mg-25	mg-24	la-138
ir-193	ir-191	hg-204	hg-202	hg-201	hg-200	hg-199	hg-198	hg-196	ge-70
gd-153	ga-71	ga-69	fe-58	fe-57	fe-56	fe-54	es-255	es-254	er-170
er-168	er-164	er-162	dy-158	dy-156	cu-65	cu-63	cr-54	cr-53	cr-52
cr-50	co-58	cm-250	cm-249	cl-37	cl-35	cf-254	cf-253	ce-139	ce-138
ce-136	ca-48	ca-46	ca-44	ca-43	ca-42	ca-40	bk-250	be-7	ba-133
ba-132	ba-130	as-74	ar-40	ar-38	ar-36	am-244	am-242	ac-227	ac-226
ac-225	u-241	u-240	u-239	k-41	k-40	k-39	s-36	s-34	s-33
ag-110m	am-244m	co-58m	ho-166m						

The second and third group includes (1) hydrogen, (2) stable zirconium isotopes generated as fission products, and (3) a set of isotopes with inadequate library data (Table 2D). The hydrogen and zirconium isotopes are included in the calculations directly by standard materials specifications h-zrh2 and zr-zrh2, and the isotopes that do not have library data were excluded from material specifications.

The fraction of neutrons absorbed in fission product poisons is used to evaluate the impact of transuranic isotopes and fission products on performance. The activity and gamma spectrum of specific isotopes and the decay heat following shutdown was used to generate accident source terms.

KENO output includes calculations of fission density and the fraction of fissions in fissile materials of geometry units. Duplicating fuel elements with unique identification numbers for specific core locations is used provided data to calculate radial core-peaking factors. Dividing fuel element geometries axially is used to provide data to calculate core axial core peaking factors. Dividing fuel into concentric cylinders is used to provide data to calculate power distribution axially within a fuel element.

As previously discussed, modeling zirconium hydride based fuel requires atom-density specification, and it is convenient to specify fuel in a separate file. The file is called from the SCALE input file. The isotopes in the zirconium hydride are not altered, and these components can either be included in the input file or in the external file; in both cases the material is represented identically, but processing material files for subsequent operations can be somewhat simplified by including z-zrh2 and h-zrh2 directly in the SCALE input file.

While the T-6 sequence is designed to support material changes during operation, a single very short interval at very low power provides initial k_{eff} and k_{eff} following a small perturbation. This provides two sequential criticality calculations under essentially the same conditions, and provides an indicator for anomalies or issues in calculation.

Material temperature and/or density can be varied within a single calculation. This option provides data for reactivity calculations to determine temperature coefficients for fuel and water. An example of specification to perform calculations with varying water temperature is provided below. Temperature changes are specified by material (3 in the example). Density variations are specified by material identification (3 in the example), the number of isotopes in the material for which temperature is changed (2 in the example) and identification of the isotopes (1001 and 8016). Density is specified as the fraction of material theoretical density at the midpoint of the burnup interval.

```
READ TIMETABLE
TEMPERATURE 3
 0          293.15
 1.00E-09   293.15
 2.00E-09   297.15
 3.00E-09   301.15
 4.00E-09   305.15
 5.00E-09   309.15
 6.00E-09   313.15
 7.00E-09   317.15
 8.00E-09   321.15
 9.00E-09   325.15 END
```

```
DENSITY 3  2 1001  8016
 0          1
 1.00E-09   1
 2.00E-09   0.99909077
 3.00E-09   0.998025861
 4.00E-09   0.996816392
 5.00E-09   0.995471982
 6.00E-09   0.994000844
 7.00E-09   0.992410292
 8.00E-09   0.990706538
 9.00E-09   0.98889499 END
END TIMETABLE
```

B. Geometry

The geometry units used in UT TRIGA analysis are shown in Table 3. The table includes units used in all calculations ("Base Units"), and special-purpose units for specific calculations ("Optional Units"). A more complete description and use of geometry units is provided later.

Geometry specifications in SCALE allow "units" to be placed within an "array." With the core specified as an array bounded by the core barrel, locations within the core are filled with "units" containing fuel elements, "dummy" graphite rods, water, or control rods.

Table 3: Geometry Units

Base Units		Optional Units	
Unit #	Description	Unit #	Description
1	Fuel element	8	Fuel Follower control Rod
2	Graphite element	200	Fuel axially/radially divided
3	Water void	201-206	Grid Plate Position B01-B06
4	Fuel Follower control Rod	13	Fuel element, alternate burnup
5	Transient Rod	14	Fuel element, alternate burnup
6	Grid plate		
7	Reflector and core		

Variations in fuel specifications were used. Identifying individual elements (Units 201-206) allows determining local, element-specific flux and power. Segmenting fuel element geometry (Unit 200) allows determining flux profile in individual elements. As previously indicated, fuel in the 1992 core included 4 different burnup values; the single outlier is not modeled separately. Burnup values uses include new fuel in the fuel follower control rods (Unit 1), standard fuel elements located in the inner rings (Unit 13), and standard fuel elements located principally in the outer rings (Unit 14).

The three fuel follower control rods are identical, and can be represented by a single unit description (Unit 4). Control rod positions are manipulated by translation along the Z axis. Duplicating the fuel follower control rod description with a different unit geometry number (Unit 8) allows a single control rod to be manipulated independently of other control rods in calculations.

C. Material Specifications

SCALE material specification uses standard (pre-defined) material identification or a combination of standard materials in a mixture, with mixture identification starting "wtp." Material identification is followed by an integer label, used in geometry units to specify the presence of the material. The next value in a material specification establishes density. For standard materials, a density-value greater than 0 is a multiplier applied to theoretical density. For "wtp" material, the density-value is the mass density. A density value of 0 indicates the next value in the specification is atom-density, with units of atoms per barn cm^{-1}. For "wtp" materials, density is followed by a value indicating the number of standard materials in the mixture, then the standard material identification and density multiplier for each isotope. The last two values are the temperature and an "end" statement to indicate completion.

Standard compositions exist for all materials used in calculations for the UT TRIGA except for air and fuel. Air was specified as "wtpair." Fuel was specified by isotope atom density to support an implementation of fission products analogous to ADDNUX routines (with hydrogen removed).

Table 4: Sample Material Specification

Material Description	Standard Material	ID	Fraction TD or ρ	No. Iso.	Iso. Frac.	atom-B/cm	TEMP	END
Zr in ZrH_2	zr-zrh2	1	0			0.034448473	300	END
H in ZrH_2	h-zrh2	1	0			0.055117557	300	END
Read file FUEL1.TXT	<FUEL1.TXT							
Zirconium	Zirconium	2	1				300	END
Water	h2o	3	1				300	END
Stainless Steel 304	ss304	4	1				300	END
Graphite	Graphite	5	1				300	END
Aluminum	aluminum	6	1				300	END
Aluminum in RSR	aluminum	7	0.944				300	END
	Wtptair	8	0.00123	2			300	END
Air	7014				80		300	END
	8016				20		300	END
			1				300	END
B4C Poison	b4c	9	0.984				300	END
Molybdenum	Mo	10	1				300	END

Materials used in UT TRIGA analysis are shown in Table 4, along with the required specification by column (where applicable). "Standard Material" indicates (except for the "read file" line) a pre-

defined label identified in the in SCALE standard material library. "ID" is the input file unique identification. "Fraction TD or ρ" is the fraction of thetorical density identified in the in SCALE standard material library or (for "wtp") density in g cm^{-3}. "No. Iso." and "Iso. Frac." is the number of isotopes used in a "wtp" specification and the fraction of each isotope in the material. If "Fraction TD or ρ" has a zero-value, the material concentration is atom density normalized to a barn. The TEMP and END specifications are self-explanatory.

3.0 UT CORE AND FUEL GEOMETRY

Principle dimensions used in modeling the UT TRIGA core are taken from schematic drawings prepared by General Atomics. Major dimensions of fuel, core shroud, core barrel, and the reflector are incorporated in modeling. Although the models include beam ports and the rotary specimen rack geometry, modeling of these features is not discussed in this section. Fuel end fittings are modeled as conical, stainless steel structures at the ends of the element (not including the fins or end-pins). End fittings for graphite "dummy" elements are not modeled.

3.1 CORE BARREL (REFLECTOR AND GRID PLATES)

The radial reflector is annular, with a cylindrical outer surface (radius of 81.994 cm) and the inner surface conforming to the inner shroud. For convenience the reflector is assumed to extend the full height of the inner shroud; however, the reflector actually ends about 3 in. above the lower grid plate (T2W210J111), at an elevation corresponding to about ½ of the fuel element's lower axial reflector (about 1 ⅓ in. below the fuel). The reflector annulus (core barrel) and the lower grid plate are composed of aluminum alloy 6061. They are shaped by two hexagonal prisms, one rotated 30° as illustrated in Fig. 5. One of the hexagonal shapes is 55.625 cm, the other 52.637 cm (GA Technologies Inc. drawing T2W210J111). The lower grid plate is 1.25 in, thick (NETL BGP001), with the upper surface 33.249 cm, and the lower surface 36.424 cm below the center of the fuel.

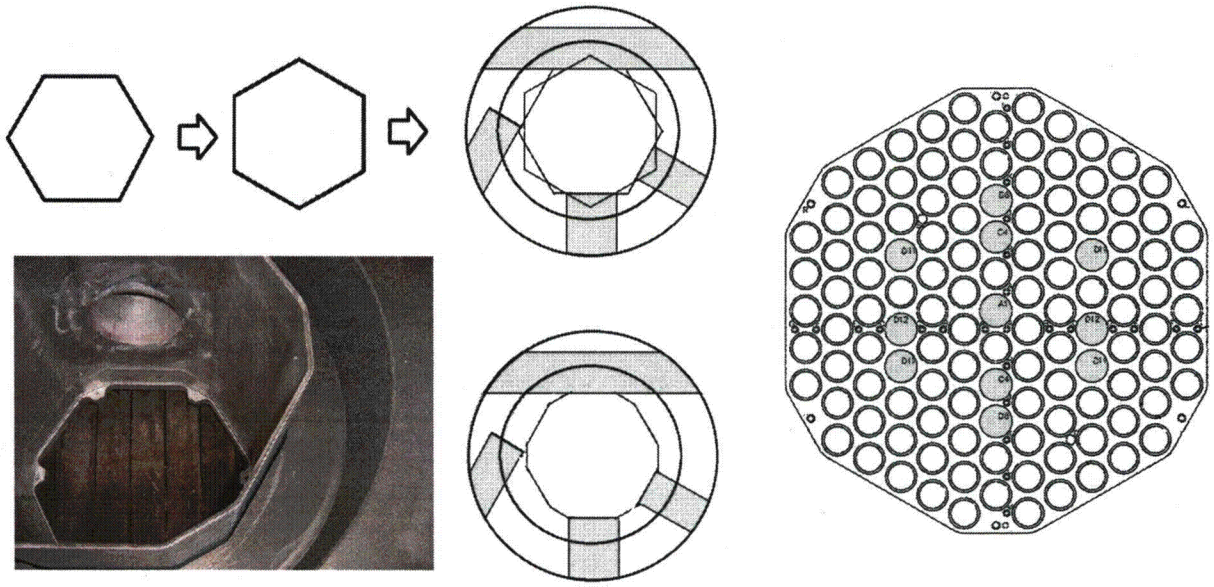

Figure 5, Core Shroud and Lower Gird plate

The annulus at the upper part of the inner reflector surface and the upper grid plate are cylindrical. The upper reflector has a space for a rotary specimen rack, displacing the reflector to an outer radius of

35.715 cm (Rotary Rack Assy. Mark I & II, TO6S14E115). Center to center distance for the fuel element positions (pitch) in the grid plates is 1.714 in. (4.355 cm, T2W210E108 - Top Grid Plate, NETL BGP0001 - Bottom Grid Plate). Upper grid plate penetrations are nominally 1.505 in. (3.823 cm) in diameter. The bottom grid plate has a set of positions with the same diameter as the top, shaded in Fig. 6, and the remaining positions 1.250 in. in diameter. In addition to the variations in lower grid plate penetration diameters, the upper grid plate has clearance for fuel elements, graphite elements, control rods, control rod guide tubes, and experiments. To simplify modeling, all grid plate penetrations are assumed at the same diameter as the component inserted in the position.

The upper grid plate is 5/8 in. thick (Top Grid plate, NETL TGP0001), with the lower elevation and the upper elevation referenced to the center of the fuel 42.799 cm and 41.224 respectively. Dimensions of penetrations in the upper grid plate are provided in Table 5.

Table 5: Upper Grid Plate Dimensions

Label		
A	Description	In.
B	Fuel Position	1.505
C	6/7 Element Facility diameter	5.140
D	3 Element Facility Y displacement	4.285
E	3 Element Facility X displacement	3.464
F	3 Element Facility Y displacement	5.999
G	3 Element Facility X displacement	3.464
H	Grid Plate Diameter	21.75
I	6/7 Element, Diameter 1	4.002
J	Grid plate thickness	0.62
K	6/7 Element Facility, Diameter 2	4.175
L	3 Element Cutout	2.370
M	Alignment pin hole	5/6

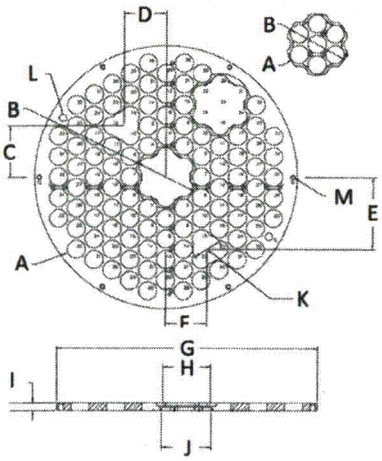

Figure 6, Upper Grid plate

Although the core is not actually symmetric about the fuel, the reference point selected for physics models is the center of the active fuel. The distance from the center of the fuel to the top of the lower grid plate is 13.09 in. (33.249 cm) above the top of the lower grid plate (based on dimensions from GA drawings[3]). Modeling of standard fuel elements as based on these dimensions is provided in Table 5.

3.2 TRIGA FUEL ASSEMBLIES

TRIGA fuel is composed of stainless steel cladding enclosing three 5 in. cylinders of uranium in zirconium hydride. Standard TRIGA fuel is identified[4] as

> *8.5 to 12 wt % uranium (20% enriched) as a fine metallic dispersion in a zirconium hydride matrix. The H/Zr ratio is nominally 1.6 (in the face centered cubic delta phase).*

[3] TS13S210B217, TS13S210B229, TS13S210C212, TS13S210C214, TS13S210C218, TS13S210C226, TS13S210C227, TS13S210D210, and TS13S210D213, derived by GA

[4] GA Project No. 4314, The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel, M. T. Simnad (Feb. 1980) and NUREG 1282, Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors, Docket No. 50-163 (Aug 1987)

Fuel used at the UT TRIGA is nominally 8.5 wt%, 38 grams ^{235}U. Density is calculated from fuel mass and dimensions. At nominal enrichment of 20% with fuel volume 389.5 cm^3, fuel has 2235.3 g U-$ZrH_{1.6}$, with density calculated at 5.74 g/cm^3; at enrichment of 19.5%, fuel mass is calculated to be 2292.6 g U-$ZrH_{1.6}$, with density 5.89 g/cm^3. Axial graphite reflectors are installed above and below the fuel, with a protective molybdenum disk between the lower reflector and the fuel. A gap above the upper axial reflector permits thermal expansion and provides space for outgassing of fission products and hydrogen. End fittings are welded to the cladding above the gas gap and below the lower axial reflector; the end fittings are machined to extend into the cladding.

Table 6: TRIGA Standard Fuel Element (SFE) Dimensions

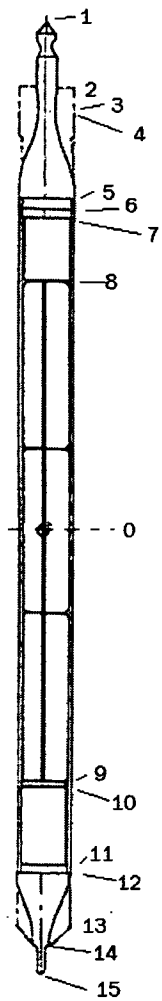

Figure 7, SFE

	NODE	z_2 in.	cm
1	Top of upper pin	15.50	39.370
2	Top of upper fins	13.94	35.408
3	Top of reinforcment	13.06	33.172
4	Bottom of reinforcment Top of conical shape	12.81	32.537
5	Bottom of conical shape Top of cladding Weldment plug top	10.81	27.457
6	Weldment plug bottom Top of gap	10.56	26.822
7	Bottom of gap Top of axial reflector	10.060	25.552
8	Bottom of axial reflector Top of fuel	7.50	19.050
9	Bottom of fuel Top of Moly disk	-7.5	-19.050
10	Bottom of moly disk Top of axial reflector	-7.531	-19.129
11	Bottom of axial reflector Top of weldment plug	-11.251	-28.578
12	Bottom of welment plug Bottom of cladding Top of conical end-shape Top of lower fins	-11.501	-29.213
13	Top of lower grid plate Start fin taper	-13.09	-33.249
14	Bottom of conical shape Top of lower pin	-13.695	-34.785
15	Bottom of lower pin	-14.441	-36.680

Although the end fittings are integral subassemblies, they are modeled for convenience as a plug located within the cladding and a separate conical shape above (below) the upper (lower) part of the cladding. The fins and the upper and lower pins (Fig. 7, 8) extending beyond the conical sections are neglected in modeling; the low mass of the fins and the distance (of the fins and pins) from active fuel should result in negligible impact on core physics performance.

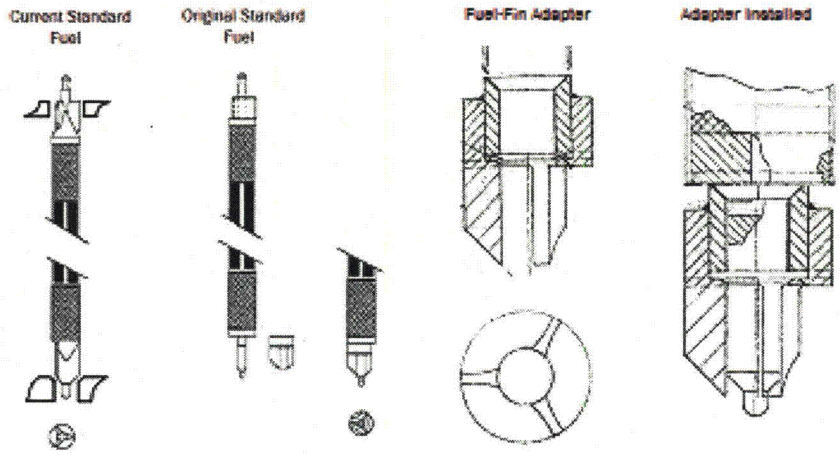

Figure 8, Standard Fuel Element Details

Early TRIGA reactor fuel was supported in the core on a pin at the lower end of the fuel element that rested in a depression in the lower grid plate; later TRIGA fuel is supported by three fins in contact with the lower grid plate. An adapter (Fig. 9) is required to use older fuel in current TRIGA grid plates. The adapter mimics the geometry of the fins structure, and the older fuel with the adapter is not differentiated from the newer fuel in modeling.

Figure 9, Adapter

There are 9 grid plate locations designed to accommodate control rods and the central thimble, with holes at the same diameter in the upper and lower grid plates. Fuel can be used in these locations using an adapter (a hollow cylinder) secured to the aluminum "safety plate" below the lower grid plate. The top of the adapter mimics the top of the lower gird plate, providing support surface for fuel elements fins.

Holes in the surface of the adapter permit cooing flow through the bottom of the fuel element (Fig. 9). These adapters are designed to make the geometry of the lower grid plate in these positions compatible with the geometry of the other penetrations, and are not modeled. A photograph of the area under the lower gird plate is provided in Fig. 10.

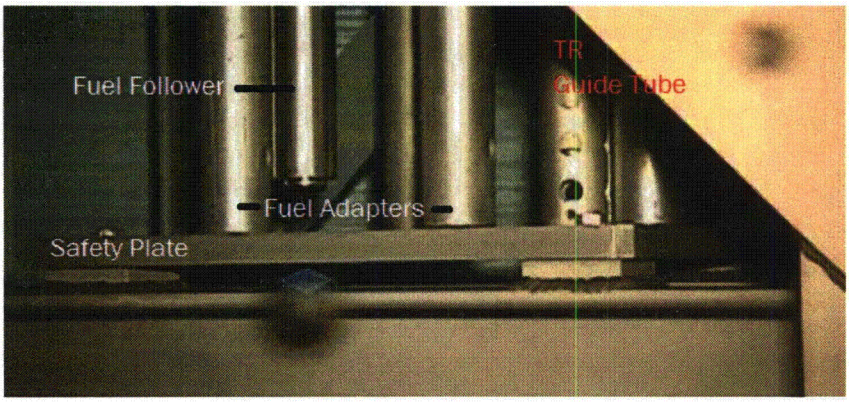

Figure 10, Area Below the Lower Grid Plate

Table 6: Standard Fuel Element Modeling

Figure 11, SFE Model

	Component	Upper NODE	Lower NODE	Δz in.	Δz cm	r_1 cm	r_2 cm
A	Upper conical shape	3	5	2.25	5.715	0	1.8771
B	Upper weldment Plug	5	6	0.25	0.635	1.8263	na
C	Air gap	6	7	0.5	1.27	1.8263	na
D	Upper aixial relfector	7	8	2.56	6.502	1.8263	na
E	Fuel	8	9	15	38.1	1.8263	na
F	Cladding	5	12	22.32	56.570	1.8263	1.8771
G	Moly disk	9	10	0.031	0.079	1.8263	na
H	Lower axial reflector	10	11	3.72	9.449	1.8263	na
I	Lower weldment plug	11	12	0.25	0.635	1.8263	na
J	Lower conical shape	12	14	2.194	5.572	0	1.8771

Dimensions of UT TRIGA reactor fuel assemblies for computational models are provided in Table 5 and Fig. 7, based on dimensions from GA drawings identified in private communication with General Atomics. A simplified model for calculations is provided in Table 6 and Fig. 11.

3.3 CONTROL RODS

Two types of control rods are used in the UT TRIGA: fuel follower control rods (FFCR) and a transient control rod. The poison sections of control rods are composed of B_4C, heat pressed to a density[5] greater than 2.48 g cm^3 (theoretical density of B_4C is 2.52 g cm^3). Fuel follower control rods incorporate a section of TRIGA fuel below the control element so that as boron poison is removed from the core, fuel is added to the core. The transient control rod uses a guide tube and has an air filled follower. Control rod components are positioned by z-axis translation to simulate control rod motion.

3.3.1 FUEL FOLLOWER CONTROL RODS.

Standard fuel element follower control rods are stainless steel tubes with welded end fittings approximately 45 in. (114 cm) long by 1.35 in. (3.429 cm) in diameter (1991 UT SAR). Component dimensions are taken from GA drawing TOS250D225 (Control Rod – Fuel Follower).

Table 7: FFCR Component Thickness (Z Axis)

Component	in.	cm
Upper End Fitting	1.5	3.81
Upper Air Void	3.5	8.89
Magneform Plug	0.5	1.27
Poison/Air Gap	0.12	0.3048
Poison	15	38.1
Magneform Plug	0.5	1.27
Fuel/Air Gap	0.25	0.635
Fuel	15	38.1

[5] GA Drawing TOS250B226, Poison

Magneform Plug	1	2.54
Lower Air Void	5.375	13.6525
Lower End Fitting	0.5	1.27

The upper 6.5 in. (16.51 cm) of the fuel follower control rod is an air void, separated and secured from the poison section by a 0.5 in. (1.27 cm) plug secured with a magneform weld. There is a small (0.12 in, 0.305 cm) air gap at the top of the poison section, between the poison and the plug. The poison is 15 in. (38.1 cm) of B_4C. 1.187 in. (3.01498 cm) in diameter. The poision section is separated from the fuel follower section by another 0.5 in. plug and magneform weld. The top of the fuel follower section is a 0.25 in (0.635 cm) air gap, above the fueled. The fuel rests on a double thickness 1 in. (2.54 cm) plug and magneform weld, followed by a 6.5 (16.51) in. air void. The bottom air void has an aluminum insert with wall thickness 0.35 in. (0.089 cm). For convenience, the reference point for the control rod in Table 8 aligns the fuel section with the center of the fuel in standard fuel elements when the rod is withdrawn.

Table 8: Fuel Follower Control Rod Geometry

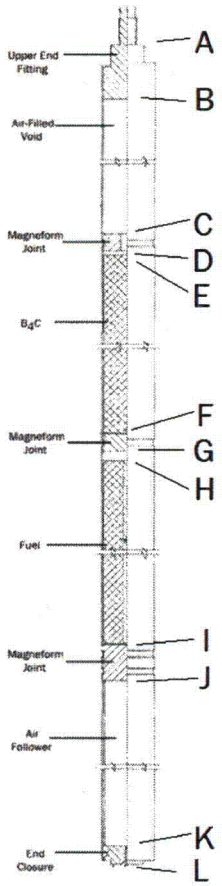

Figure 11, FFCR

KEY	DESCRIPTION	ELEVATION in	ELEVATION cm
A	Top upper end	13.87	35.230
B	Bottom upper end Top of top void	12.37	31.420
C	Bottom of top void Top of top plug	8.87	22.530
D	Bottom of mag plug Top of poison gap	8.37	21.260
E	Bottom of poison gap Top of poison	8.25	20.955
F	Bottom of poison Top of lower poison mag plug	-6.75	-17.145
G	Bottom of lower mag plug top of fuel gap	-7.25	-18.415
H	bottom of fuel gap top of fuel	-7.5	-19.05
H	bottom of fuel top of lower mag plug	-22.5	-57.15
J	bottom of lower mag plug top of lower air follower	-23.5	-59.69
K	bottom of air follower top of lower end fitting	-28.875	-73.343
L	bottom of lower end fitting	-29.375	-74.613

3.3.2 TRANSIENT CONTROL ROD.

The transient control rod is less complex than the fuel follower control rod. Transient rod cladding is a 1.25 in. (3.175 cm) diameter aluminum tube with wall thickness 0.028 in. (0.071 cm)[6]. The transient rod operates within an aluminum guide tube (1.490 in. or 3.7846 cm in diameter machined from 1.5 in./3.81 cm aluminum tubing with a wall thickness 0.065 in. or 0.1651 cm[7]; therefore inner diameter is 3.480 cm) secured to the upper grid plate and the safety plate (below the lower grid plate). The guide tube is perforated by ½ in. holes at 90° rotations on 1 in. centers at the top and bottom of the core barrel. The guide tube extends 5 feet above the safety plate, below the lower grid plate. Perforations and extensions above the upper- and below the lower-grid plate are not modeled.

The poison section of the transient rod is 15 in (38.1 cm)[5]. A double (1 in., 2.54 cm) plug with a magneform weld secures the poison section at the top and bottom. The air follower above the poison section is in an assembly 5.94 in., which includes the upper end fitting and the upper magneform weld. The air follower under the plug is 20.88 in. (53.02 cm) long, terminating in a bottom end-fitting.

Table 9: Transient Rod Dimensions

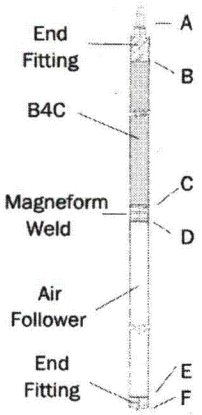

Figure 12, Transient Rod

KEY	DESCRIPTION	ELEVATION[8] in	ELEVATION[8] cm
A	Top of end fitting	9	22.860
B	Bottom of end fitting Top of poison section	7.5	19.050
C	Bottom of poison section Top of magneform weld	-7.5	-19.050
D	Bottom of magneform weld Top of air follower	-8.5	-21.59
E	Bottom of air follower Top of end fitting	-27.75	-70.485
F	Bottom of end fitting	-28.25	-71.755

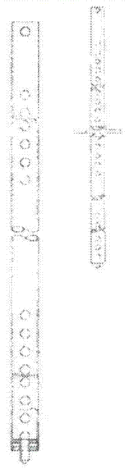

Figure 13, Guide Tube

4.0 MODELING

As previously described, SCALE uses a standard description of geometry based on extruding planer surfaces from a base to an upper extension. A material is associated with each of these volumes. Details of the geometry used to model the UT TRIGA are followed by a description of the materials.

4.1 GEOMETRY UNITS

Component geometries are used to develop "units" within the SCALE SGGP module. The model is bounded by a "global unit" composed of the reflector and core. The reflector portion includes the radial graphite reflector, the reflector canister, beam port tubes, rotary specimen rack, grid plates, and water. Each unit has one assembly identified as the boundary, a volume that encompasses all geometries in the unit.

[6] T0S252D191 – Transient Rod Assembly (ADJ)
[7] T135210D150, Guide Tube – Transient Rod
[8] T0S252D191 & 1991 UT SAR

The core portion is an array statement defining a set of hexagonal prisms bounded by the inner shroud. The hexagonal prisms are filled with appropriate geometry units that incorporate components inserted in core spaces. Components that fill the core array include cells with grid plate penetrations, water, and (where appropriate) fuel elements, graphite elements, water voids, fuel follower control rods, or the transient rod. One unit is composed of two solid aluminum plates at the thickness of the grid plates separated by water. Fig 14. illustrates three hexagonal positions filled with fuel. Fig. 15 is an example of a core configured with 71 standard fuel elements and 3 fuel followers with non-fuel positions occupied by graphite rods, and with non-fuel positions occupied by water filled channels (or water voids).

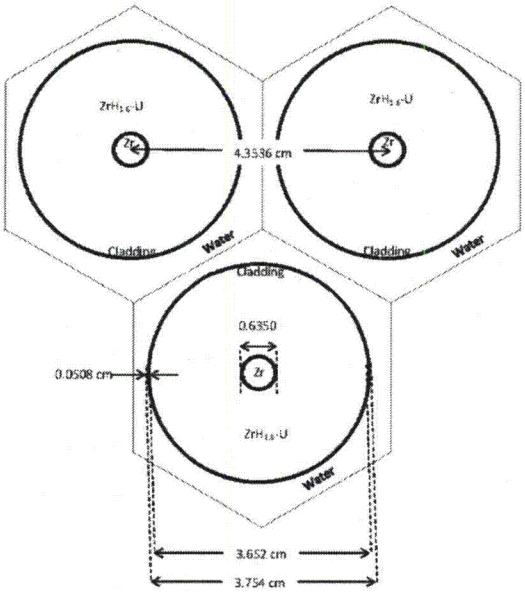

Figure 14, Dimensions of Contiguous Unit Cells with Fuel

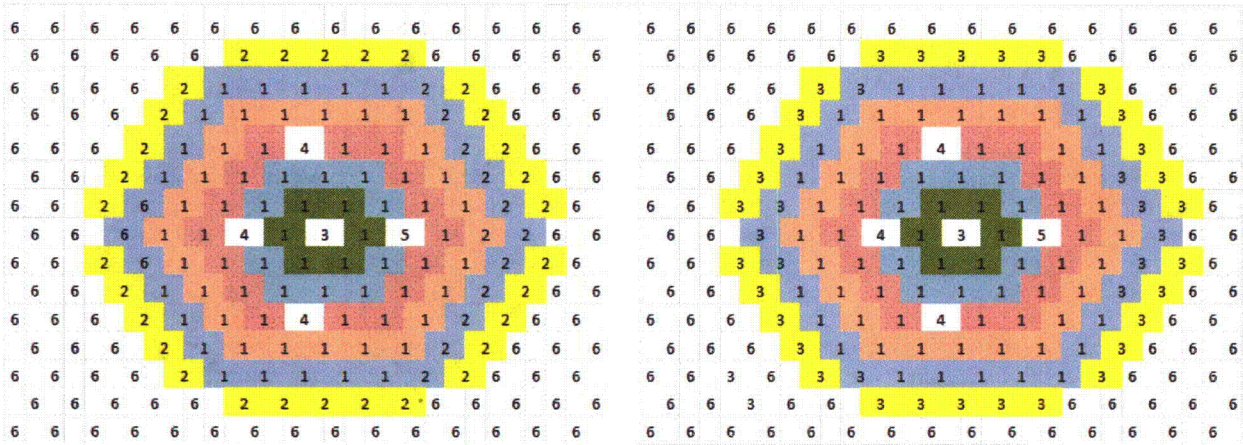

Figure 15, Core Array with Fuel (1) Graphite (2), Water void (3), Control Rods (4 & 5) & Grid Plates (6)

4.1.1 Fuel Geometry Unit

The SCALE model description for a unit to be inserted in a grid plate position that contains TRIGA fuel is provided in Table 10. As noted in Table 3, the fuel unit may be labeled or segmented to support specific calculations. When it is desired to determine power in a specific element (or set of elements, such as the B ring), duplicate units with unique labels are used in the core array. If materials are different

between elements, e.g. generated by different power histories, the unit can be duplicated with a different fuel material. When fission distribution within an element is significant, the fuel cylinder is segmented either radially or axially, as appropriate. The boundary for unit 1 is hexprism 30.

Table 10: unit 1: com="unit 1: fuel channel" *(all units cm)*

Description	Geometry	ID	Radius or Pitch	Top	Radius	Bottom
Full Channel	hexprism	30	2.177	32.309		-36.424
Grid Plate	hexprism	40	2.177	32.309		30.734
Grid Plate H_2O	cylinder	41	1.8771	32.309		30.734
End Fitting ½	cone	10	0.635	32.537	1.8771	27.457
Cladding	cylinder	11	1.8771	27.457		-29.213
End Fitting ½	cylinder	12	1.8263	27.457		26.822
Gas Gap	cylinder	13	1.8263	26.822		25.552
Axial Reflector	cylinder	14	1.8263	25.552		19.05
Fuel	cylinder	15	1.8263	19.05		-19.05
Zirc Filler	cylinder	16	0.285	19.05		-19.05
Moly Disk	cylinder	17	1.8263	-19.05		-19.129
Axial reflector	cylinder	18	1.8263	-19.129		-28.578
End Fitting ½	cylinder	19	1.8263	-28.578		-29.213
End Fitting ½	cone	20	1.8771	-29.213	0.635	-34.775
Grid Plate	hexprism	50	2.177	-33.249		-36.424
Grid Plate H_2O	cylinder	51	1.8771	-33.249		-36.424

4.1.2 Graphite Dummy Rod Geometry Unit

The SCALE model description for a unit to be inserted in a grid plate position that contains a dummy (graphite) rod is provided in Table 11. The boundary for unit 2 is hexprism 20.

Table 11: unit 2 com="unit 2: graphite rod "*(all units cm)*

Description	Geometry+	ID	Radius or Pitch	top	bottom
Channel	Hex-prism	20	2.177	32.309	-36.424
Grid Plate H_2O	Cylinder	25	1.873	32.309	-36.424
Grid plate	Hex-prism	30	2.177	32.309	30.734
Upper Pin	Cylinder	40	0.635	32.537	27.457
Cladding	Cylinder	50	1.873	27.457	-29.213
Upper Plug	Cylinder	51	1.822	27.457	26.822
Graphite	Cylinder	52	1.822	26.822	-28.578
Lower Plug	Cylinder	53	1.822	-28.578	-29.213
Lower Pin	Cylinder	60	0.635	-29.213	-34.775
Grid Plate	Hex-prism	70	2.177	-33.249	-36.424

4.1.3 Water Void Geometry Unit

The SCALE model description for a unit to be inserted in a grid plate position that does not contain a rod (dummy or fuel) is provided in Table 12; this is considered a water-void. The boundary for unit 3 is hexprism 30.

Table 12: unit 3 com="unit 3: Water Channel" *(all units cm)*

Description	Geometry	ID	Radius or Pitch	top	bottom
Channel	Hex-prism	30	2.177	32.309	-36.424
Grid plate	Hex-prism	40	2.177	32.309	30.734
Grid plate H_2O	Cylinder	41	1.911	32.309	30.734
Grid Plate	Hex-prism	10	2.177	32.309	-36.424
Grid Plate	Hex-prism	50	2.177	-33.249	-36.424
Grid plate H_2O	Cylinder	51	1.922	-33.249	-36.424

4.1.4 Fuel Follower Control Rod Geometry Unit

The SCALE model description for a unit to be inserted in a grid plate position that contains a standard, fuel follower control rod is provided in Table 13. Components in the table shaded and bounded by the dashed line are modeled with an axis translation within the geometry unit to simulate withdrawal and insertion of the control rod by specifying a position on the vertical axis. Where motion is required for a single control rod, the geometry unit is duplicated with a different unit number with the alternate unit number placed to represent the single rod. The boundary for unit 1 is hexprism 60.

Table 13: Unit 4, Standard Control Rod[9] *(all units cm)*

Description	Geometry	ID	Radius	Top	Bottom
Channel	hexprism	60	2.177	32.309	-36.424
Grid plate	hexprism	50	2.177	32.309	30.734
Cladding	cylinder	30	1.715	35.23	-74.613
Upper Void	cylinder	10	1.6637	35.23	22.53
Magneform Plug	cylinder	11	1.6637	22.53	21.26
Gap	cylinder	12	1.6637	21.26	-17.145
B4C	cylinder	14	1.651	20.955	-17.145
Magneform Plug	cylinder	15	1.6637	-17.145	-18.415
Gap	cylinder	16	1.6637	-18.415	-19.05
Fuel	cylinder	17	1.6637	-19.05	-57.15
Zirc Filler	cylinder	18	0.285	-19.05	-57.15
Magneform Plug	cylinder	19	1.6637	-57.15	-59.69
Lower void	cylinder	20	1.6637	-59.69	-73.343
End fitting	cylinder	25	1.6637	-73.343	-74.613
Grid Plate	hexprism	40	2.177	-33.249	-36.424

4.1.5 Geometry Unit for Transient Control Rod

The SCALE model description for a unit to be inserted in a grid plate position that contains the transient control rod is provided in Table 14. Movement is simulated similar to the fuel follower control rod. The boundary for unit 5 is hexprism 30.

[9] Items in the boxed are specified as "origin x=0 y=0 z=0" for rod fully out, origin x=0 y=0 z=38.1" for rod fully out

Table 14: Unit 5, Transient Control Rod[8] *(all units cm)*

Description	Geometry	ID	Radius	Top	Bottom
Channel	hexprism	30	2.177	32.309	-36.424
Outer Guide Tube	cylinder	60	1.911	32.309	-36.424
Inner Guide Tube	cylinder	65	1.829	32.309	-36.424
Grid plate	hexprism	40	2.177	32.309	30.734
Cladding	cylinder	20	1.59	32.309	-76.248
End Fitting	cylinder	10	1.519	32.309	29.151
Air void	cylinder	11	1.519	29.151	20.828
Magneform Weld	cylinder	12	1.519	20.828	19.05
Radial Gap	cylinder	13	1.519	19.05	-19.05
B_4C	cylinder	14	1.3	19.05	-19.05
Magenform Weld	cylinder	15	1.519	-19.05	-20.32
Air void	cylinder	16	1.519	-20.32	-64.61
End Fitting	cylinder	17	1.519	-64.61	-66
Grid plate	hexprism	50	2.177	-33.249	-36.424

4.1.6 Geometry Unit for Solid Grid Plate

The core grid plate does not have penetrations at the periphery of the core. Since the core is represented by an array of hexagonal spaces, the spaces at the core periphery are solid aluminum plates separated by water. The SCALE model description for these grid plate positions is provided in Table 15. The boundary for unit 1 is hexprism 30.

Table 15: Unit 6, Grid Plate and Water *(all units cm)*

Description	Geometry	ID	Radius	Top	Bottom
Grid Plate	hexprism	30	2.177	32.309	-36.424
Water	hexprism	40	2.177	32.309	30.734
Grid Plate	hexprism	50	2.177	-33.249	-36.424

4.1.7 Geometry Unit for Reflector and Core

The reflector and core includes an aluminum "shroud" composed of two nested hexagonal geometries, the rotary specimen rack, the graphite reflector, and beam ports. The beam ports require translation not shown. The SCALE model description for the reflector and core is provided in Table 16. The boundary for unit 7 is cylinder 40.

Table 16: Global Unit Unit 7, Reflector and Core[10] *(all units cm)*

Description	Geometry	ID	Radius	Top	Bottom
Core Barrel Hex	rhexprism	10	25.161	2.309	-36.424
Core Barrel Hex	rhexprism	11	25.796	2.309	-36.424
Rotary Specimen Rack	cylinder	20	30.083	32.309	6.509
Rotary Specimen Rack	cylinder	21	30.718	32.309	6.509

[10] Core barrel hexagonal prisms are rotated by "rotate a1=0 a2=0 a3=60"; beam ports are translated by (50) origin x=0 y=35.255 z=-6.985, (51) origin x=-33 y=10 z=-6.985 rotate a1=0 a2=0 a3=-30, (52) origin x=0 y=0 z=-6.985, and (53) origin x=0 y=0 z=-6.985 rotate a1=0 a2=0 a3=60

Table 16: Global Unit Unit 7, Reflector and Core[10] *(all units cm)*

Description	Geometry	ID	Radius	Top	Bottom
Rotary Specimen Rack	cylinder	22	31.115	32.309	6.509
Rotary Specimen Rack	cylinder	23	35.401	32.309	6.509
Rotary Specimen Rack	cylinder	24	36.671	32.309	6.509
Outer Reflector Surface	cylinder	40	59.69	32.309	-36.424
Beam Port	xcylinder	50	7.62	90	-90
Beam Port	ycylinder	51	7.62	-10	-90
Beam Port	ycylinder	52	7.62	0	-90
Beam Port	ycylinder	53	7.62	0	-90

4.2 MATERIALS

ORNL/TM-2005/39, Section M.8, Standard Composition Library, provides material composition data for all materials used in modeling with SCALE except for air and the zirconium hydride fuel. Nominal values of density from the Standard Composition Library are used except in the case of the control rod poison sections and aluminum in the rotary specimen rack that have complex structures (approximated as 94.4% theoretical density). As previously note, General Atomics specifications on boron carbide used in the control rods is at least 98% of theoretical density listed in ORNL/TM-2005/39, Table M8.2.4. Small volumes of air are modeled in gas-gaps for the fuel, void spaces in control rods, and beam port tubes, assumed to be 80% nitrogen and 20% oxygen at 0.00123 g/cm^3 with only the major stable isotopes.

4.2.1 Material Specification in Geometry Unit 1 (Fuel)

Fission product and transuranic isotopes in a fuel element are functions of initial loading and power history. Benchmarking to actual conditions requires some level of rigor in defining materials representative of initial conditions. In this case, calculations are performed to determine isotope concentrations at burnup intervals. Geometry unit 1 (Table 17) is designated to use fresh fuel, and is reproduced under different unit labels (13 or 14) where a material represents fuel composition at burnup intervals as calculated by simulated operation at a specified power for a specified time.

Table 17: Material Specification in Geometry Unit 1 (Fueled Channel)

Description	Material	ID		Geometry	
				Included	Excluded
Grid Plate	Aluminum	7	1	40	51
Grid Plate Water	Water	3	1	41	10
End Fitting 1/2	SS304	4	1	10	--
Cladding	SS304	4	1	11	12-19
End Fitting 1/2	SS304	4	1	12	--
Gas Gap	Air	9	1	13	--
axial reflector	Graphite	5	1	14	--
FUEL	Fuel	1[11]	1	15	16
Zirc Filler	Zr	2	1	16	--

[11] Alternately 13 or 14, as required

Table 17: Material Specification in Geometry Unit 1 (Fueled Channel)

Description	Material	ID		Geometry	
				Included	Excluded
Moly disk	Molybdenum	12	1	17	--
axial reflector	Graphite	5	1	18	--
End Fitting 1/2	SS304	4	1	19	--
End Fitting 1/2	SS304	4	1	20	--
Grid Plate	Aluminum	7	1	50	51,20
Grid Plate Water	Water	3	1	51	20
Remainder	Water	3	1	30	Everything else

4.2.2 Material Specification in Geometry Unit 2 (Graphite Rod Channel)

Graphite "dummy" rods are used in core positions, with material specification provided in Table 18.

Table 18: Material Specification in Geometry Unit 2 (Graphite Rod)

Description	Material	ID		Geometry	
				Included	Excluded
Channel water	Water	3	1	20	25,30,70
Upper Grid Plate	Aluminum	7	1	30	25
Grid plate water	Water	3	1	25	40,50,60
Upper Pin	Aluminum	7	1	40	--
Cladding	Aluminum	7	1	50	51-53
Upper plug	Aluminum	7	1	51	--
Graphite	Graphite	6	1	52	--
Lower Plug	Aluminum	7	1	53	--
Pin	Aluminum	7	1	60	--
Lower Grid Plate	Aluminum	7	1	70	25

4.2.3 Material Specification in Geometry Unit 3 (Water Void Channel)

Grid plate positions may be left vacant. Therefore this unit contains upper and lower gird plates and water. The unit is referred to as a "water-void," described in Table 19.

Table 19: Material Specification in Geometry Unit 3 (Water Void Channel)

Description	Material	ID		Geometry	
				Included	Excluded
Upper Grid Plate	Aluminum	7	1	40	41
Upper Grid Plate Hole	Water	3	1	41	--
Water	Water	3	1	30	40,41,50,51
Lower Grid Plate	Aluminum	7	1	50	51
Lower Grid Plate Hole	Water	3	1	51	--

4.2.4 Material Specification in Geometry Unit 4 (Fuel Follower Control Rod)

The material specification for fuel follower control rods is provided in Table 20. The fuel follower control rods in the 1992 core were new, requiring material specification as fresh fuel.

Table 20: Material Specification in Geometry Unit 4 (Fuel Follower Control Rod)

Description	Material	ID		Geometry	
				Included	Excluded
Channel Water	Water	3	1	60	50,40,30
Upper Grid Plate	Aluminum	7	1	50	30
Cladding	SS304	4	1	30	10,11,12,15,16,17,19,20
Upper Void	Air	9	1	10	--
Magneform Weld	SS304	4	1	11	--
Gas Gap	Air	9	1	12	14
Poison	B_4C	10	1	14	--
Magneform Weld	SS304	4	1	15	--
Gas Gap	Air	9	1	16	--
FUEL	Fuel	1	1	17	18
Zirc Filler	Zirconium	2	1	18	--
Magneform Weld	SS304	4	1	19	--
Lower Void	Air	9	1	20	--
Lower Grid Plate	Aluminum	7	1	40	30

4.2.5 Material Specification in Geometry Unit 5 (Transient Rod)

The material specification for the transient control rod is provided in Table 21.

Table 21: Material Specification in Geometry Unit 5 (Transient Rod)

Description	Material	ID		Geometry	
				Included	Excluded
Channel Water	Water	3	1	30	40,50,60
Guide Tube	Aluminum	7	1	60	65
Water in Guide Tube	Water	3	1	65	20
Grid Plate Water	Aluminum	7	1	40	60
Cladding	Aluminum	7	1	20	10,11,12,13,15,16,17
End Fitting	Aluminum	7	1	10	--
Upper Void	Air	7	1	11	--
Magneform Weld	Aluminum	7	1	12	--
Air Surrounding Poison	Air	9	1	13	14
Poison	B_4C	11	1	14	--
Magneform Weld	Aluminum	7	1	15	--
Lower Void	Air	9	1	16	--
Lower End Fitting	Aluminum	7	1	17	--
Lower Grid Plate	Aluminum	7	1	50	60

4.2.6 Material Specification in Geometry Unit 6 (Grid Plate and Water)

The material specification for array locations where the grid plate does not have penetrations (i.e., areas at the periphery of the core barrel) rods is provided in Table 22.

Table 22: Material Specification in Geometry Unit 6 (Grid Plate and Water)

Description	Material	ID		Geometry	
				Included	Excluded
Upper Grid Plate	Aluminum	7	1	40	--
Lower Grid Plate	Water	3	1	30	40,50
Lower Grid Plate Hole	Aluminum	7	1	50	--

4.2.7 Material Specification in Geometry Unit 7 (Core and Reflector)

Outer surfaces of the reflector bound the model. The material specification for the core and reflector (including the rotary specimen rack and the beam ports) is provided in Table 23.

Table 23: Material Specification in Geometry Unit 7 (Core and Reflector)

Description	Material	ID		Geometry	
				Included	Excluded
Core Barrel	Aluminum	7	1	11	10
Reflector	Graphite	6	1	40	24,11,50,51,52,53
Beam Port	Air	9	1	50	40,11
Beam Port	Air	9	1	51	40,11
Beam Port	Air	9	1	52	40,11
Beam Port	Air	9	1	53	40,11
RSR	Air	9	1	20	11
RSR Walls	Aluminum	7	1	21	20
RSR Walls	Aluminum	7	1	22	21
RSR Tubes	Aluminum 2	8	1	23	22
RSR Walls	Aluminum	7	1	24	23

5.0 LIMITING CORE CONFIGURATION

NUREG 1537 requires a Limiting Core Configuration (LCC) be identified as the configuration that has a fuel element with the highest power density. The highest power density occurs in the fuel element where the ratio of the power in an element to the average power across all elements (power per element) is largest.

NUREG 1537 requires information relative to the operational core. Fuel loading in excess of the limiting core configuration supports compensates for fuel burnup and the negative reactivity that may occur with experiment insertion. Operational core loading is limited by the maximum excess reactivity permitted in the Technical Specifications.

5.1 Determining the Limiting Core Configuration

The highest average power per fuel element will occur when the number of fuel elements is smallest. The use of a reflector reduces critical mass by reducing neutron leakage. With less leakage, the power generated in fuel elements at the core periphery will be higher and generally reduce the peak-to-average power ratio. However, fewer fuel elements are required if non-fuel spaces are filled with graphite rods; the use of graphite rods will result in higher average power (per element). Calculations were therefore performed to determine the minimum number of fuel elements required to achieve criticality using graphite reflector rods and also using water voids.

The highest power density will occur when the reactor is operating at full power. Therefore, SCALE T-6 calculations were performed with the number of fuel elements varying from 49 to 117 assuming fresh fuel at varying fuel temperatures (300 K, 350 K, 450 K, 550 K and 600 K) simulating fuel temperature range from zero-power critical to full power operation.

The highest power density will occur in an element closest to the center of the core. Therefore the B-ring elements were individually labeled to provide measurement of fissions in each element.

The average power for each fuel element is determined based on the minimum number of fuel rods required to support full power operation for graphite and for water void core configurations. The ratio of power generated in each B ring element to the average power per element is determined for the two cases. The power in each B ring element is determined as the product of the average power and the peaking factor for that B ring element.

Therefore, calculations using the base units indicated in Table 3 and alternate units 201-206 were performed to determine the fuel element producing the maximum power assuming:

- The number of fuel elements was varied from 49 elements to 117 elements (Table 24)
- Two sets of core configurations with gird plate positions that are not occupied by fuel or control rods specified as:
  - all water voids
  - all graphite rods
- Fuel temperatures varying from room temperature to 600 K
- B ring elements individually specified to determine element-specific power

Table 24: Tested Core Configurations *(Vol in cm^3)*

CORE	No. SFE	VOL SFE, No B Ring	VOL SFE, B Ring	ALL SFE	VOL FFCR	ALL FUEL
UNIT		1	201-206		4	
49	46	15580.16	2337.02	17917.19	964.74	18881.93
53	50	17138.18	2337.02	19475.20	964.74	20439.94
57	54	18696.19	2337.02	21033.22	964.74	21997.96
59	56	19475.20	2337.02	21812.23	964.74	22776.97
65	62	21812.23	2337.02	24149.25	964.74	25113.99
73	70	24928.26	2337.02	27265.28	964.74	28230.02
74	71	25317.76	2337.02	27654.79	964.74	28619.53

Table 24: Tested Core Configurations *(Vol in cm³)*

CORE	No. SFE	VOL SFE, No B Ring	VOL SFE, B Ring	ALL SFE	VOL FFCR	ALL FUEL
78	75	26875.78	2337.02	29212.80	964.74	30177.54
83	80	28823.30	2337.02	31160.32	964.74	32125.07
87	84	30381.32	2337.02	32718.34	964.74	33683.08
89	86	31160.32	2337.02	33497.35	964.74	34462.09
92	89	32328.84	2337.02	34665.86	964.74	35630.60
97	94	34276.36	2337.02	36613.38	964.74	37578.12
97	94	34276.36	2337.02	36613.38	964.74	37578.12
100	97	35444.87	2337.02	37781.89	964.74	38746.63
106	103	37781.89	2337.02	40118.92	964.74	41083.66
117	114	42066.44	2337.02	44403.46	964.74	45368.20

The number of fuel elements in Table 24 includes standard fuel elements (SFE) and fuel followers. Three fuel follower control rods at 321.58 cm^3 are included, and six individual B ring elements at 389.50 cm^3 each. Excess reactivity was derived from k_{eff} values calculated in KENO ("Transport k"). The number of fissions in regions within geometry units was taken from the KENO "Fission Densities" table.

5.1.1 Zero Power Critical Mass

The result of the calculations for graphite and water-void configured cores is shown in Fig. 17 for room temperature (300 K) and nominal full power operation (600 K). Minimum fuel loading with graphite rods at 300 K is 60 elements (57 standard fuel elements and 3 fuel followers). Associated mass is 2.27 kg of ^{235}U (slightly more than the 1.94 kg ^{235}U in the GA TRIGA Mark reactor, slightly less than the 2.7 kg ^{235}U in the GA Advanced TRIGA Prototype reactor, and comparable to 2.24 kg ^{235}U of the GA TRIGA Mark III critical condition, op cit). The water-void configuration requires about 74 fuel element to achieve criticality at room temperature, 23% more fuel elements comparable to the 25% increase noted at GA.

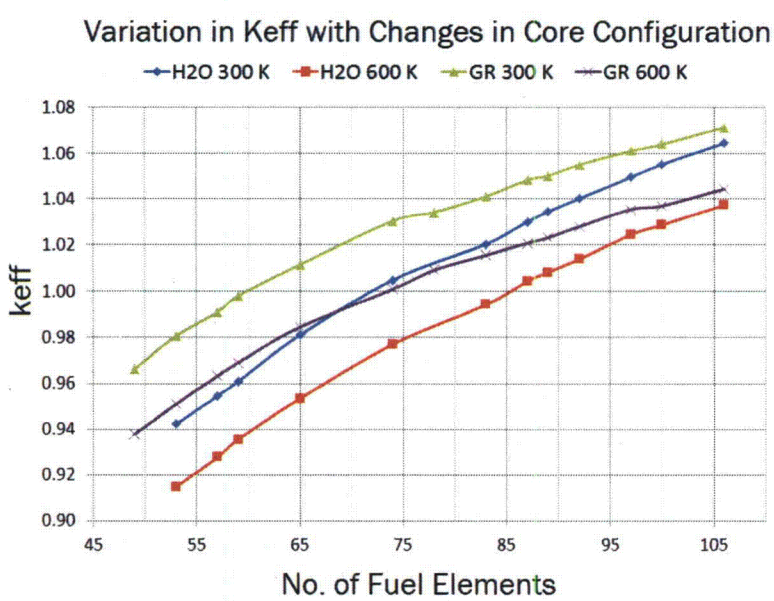

Figure 17: Excess Reactivity and Fuel Loading

5.1.2 Operational Critical Mass

Compensation for elevated temperatures associated with power operation requires additional fuel to be loaded. Reactor operation at 600 K requires about 74 fuel elements with graphite rods in non-fuel spaces and K about 86 fuel elements in the water-void configured core. Table 25 shows the bounding range of possible core configurations considered (using fresh fuel) with excess reactivity high enough to support full power operations, and low enough to meet maximum requirements. Bounding values are provided for context with shading. Therefore analysis for the limiting core configuration considers the range of graphite-rod reflected cores greater with than 74 elements, and water-void configurations with greater than 86 fuel element elements.

Table 25: Excess Reactivity ($, All Rods Out) for Potential Core Configurations

TEMP °K	Number of Fuel Elements								
	73	74	78	83	87	89	92	97	100
	Graphite Rod Configuration								
300	$3.27	$4.11	$4.87	$5.69	$6.54	$6.85	$7.41		
350	$3.05	$3.38	$4.44	$5.32	$5.98	$6.44	$7.18	Excess	
450	$1.72	$2.32	$3.38	$4.12	$4.99	$5.63	$5.92	Reactivity	
550	$0.31	$1.04	$2.08	$2.75	$3.76	$4.04	$4.71	Too High	
600	-$0.36	$0.11	$1.27	$2.17	$2.88	$3.21	$3.85		
	Water Void Configuration								
300				$2.93	$4.19	$4.77	$5.48	$6.83	$7.44
350		Excess		$2.37	$4.04	$4.39	$4.95	$6.39	$7.00
450		Reactivity		$1.37	$2.81	$3.09	$4.11	$5.34	$5.89
550		Too Low		-$0.04	$1.23	$1.86	$2.73	$4.09	$4.43
600				-$0.86	$0.60	$1.15	$1.93	$3.38	$3.95

If the core configured with all graphite rods is loaded to 87 fresh fuel elements, excess reactivity at ambient temperature exceeds the limit. If the core configured with all water voids in non-fueled positions is loaded to 100 fresh fuel elements, excess reactivity at ambient temperature exceeds the limit. Introducing water voids in the graphite-configured core will reduce the core excess reactivity, and more fuel can be added as the core transitions from all graphite to all water voids. Depletion of fuel and the introduction of fission products associated with fission will reduce core excess reactivity and more fuel can be added as burnup increases. Therefore the operational core with fresh fuel considers graphite cores loaded with less than 89 fuel elements and water void configured cores with less than 97 fuel elements.

5.1.3 Fuel Element with Highest Power Density

Elements in the B ring are exposed the highest neutron flux, and therefore generate the most power. The ratio of the power in a fuel element to the average power per element in the core is the "Peaking Factor." The product of the average power and the peaking factor is used to calculate the power produced in a fuel element.

5.1.4 B RING PEAKING FACTORS

KENO transport calculations provide fission density in each region of each geometry unit that contains fissionable material. Average fission density for the core is calculated as the sum of the total fissions for

all regions (the product of the fission density in a region and the volume of the region) normalized to the total volume of all regions with fissionable materials. The peaking factor of individually labeled units (i.e., all B ring positions) is calculated as the ratio of the fission density in the labeled unit to the (volume averaged) total fission density

The maximum B ring peaking factor for each core configuration (that can support full power operation) at 300 K, 350 K, 450 K, 550 K, and 600 K are tabulated in Table 24. The peaking factors for the graphite configuration generally decreased slightly (on the order of 1%) as temperature increased from 300 K to 600 K. In all cases, the maximum peaking factor occurred in the B01 position. Cores configured with graphite elements have a more uniform power distribution and lower maximum peaking factors. The peaking factors in the graphite configurations decreased slightly (order of 1%) as temperature increased; this effect was not observed in the water-void core.

Table 26: Maximum B Ring Peaking Factors

No.	GRAPHITE CONFIGURATIONS					WATER VOID CONGIFURATION				
Elements	300	350	450	550	600	300	350	450	550	600
74	1.53	1.53	1.54	1.55	1.57					
78	1.55	1.54	1.55	1.56	1.58					
83	1.53	1.53	1.53	1.54	1.56					
87	1.54	1.54	1.54	1.56	1.57	1.60	1.60	1.60	1.62	1.63
89	1.52	1.52	1.53	1.54	1.56	1.60	1.59	1.60	1.61	1.63
92						1.59	1.59	1.60	1.61	1.63
97						1.63	1.63	1.64	1.65	1.67

5.1.5 B RING ELEMENT POWER

Average power per fuel element is the rated power distributed over all fuel elements. The maximum B ring fuel element power (Table 27) is product of the peaking factor from Table 25 and the average power for each core configuration. It is clear that the core configured with graphite has a higher power for equivalent reactivity values as fewer fuel elements are required with the graphite reflection. The maximum power occurs with 74 elements (73 elements will allow operation at less than full power and the data is provided only for context).

Table 27: Maximum B Ring Power (kW)

No.	GRAPHITE CONFIGURATIONS Max Power, B Ring Element					WATER VOID CONGIFURATION Max Power, B Ring Element				
Elements	300	350	450	550	600	300	350	450	550	600
74	22.8	22.8	22.9	23.1	23.3					
78	21.8	21.8	21.8	22.0	22.2					
83	20.2	20.2	20.3	20.4	20.7	21.3	21.3	21.4	21.5	21.7
87	19.4	19.4	19.5	19.7	19.8	20.2	20.2	20.3	20.4	20.6
89	18.8	18.9	19.0	19.2	19.4	19.7	19.7	19.8	20.0	20.1
92						19.1	19.1	19.1	19.3	19.5
97						18.5	18.5	18.6	18.7	18.9

5.1.6 LIMITING CORE CONFIGURATION

The B ring fuel element with the maximum power across all cores that that will support full power operation generates is 23.2 kW at 600 KW, 74 elements in the graphite core configuration.

5.2 Nuclear Characteristics of the Limiting Core Configuration

KENO transport and ORIGEN buildup and decay calculations provide information representative of or useful in determining physics and operational parameters. Control rod positions are adjusted to achieve the desired conditions (critical or rods fully withdrawn, as applicable). The beginning of core life simulation assumes the limiting core configuration with unirradiated fuel then. The end of core life is simulation assumes a fully fueled core (with water void positions reserved for the neutron source and the pneumatic tube), operated at full power until criticality cannot be maintained at full power. Space is reserved in the core lattice for the neutron source and the pneumatic irradiation facility.

5.2.1 Physics Parameters and Flux Density

Physics parameters and flux density are determined from simulation of the critical condition. All of the base units identified in Table 3 were used, with control rods in a banked position (equal Unit 4 and Unit 5 translations along the z axis). A series of calculations was performed for the LCC from 1 mW to full power at fuel temperature of 300 K with the control rods positioned at 18 cm withdrawn. Additional simulations were performed at full power operation, first with the same control rod position at temperature of 600 K and then with the control rod position adjusted to compensate for the elevated fuel temperature, approximately critical at control rod position 26.5 cm withdrawn. Table 28 values for "Neutron lifetime," "generation time," "mean free path" are taken from the KENO summary report. Neutron flux from transport calculations at each power level is taken from the (energy) "group" report row totals, tabulated in Table 29.

Table 28: (Critical) Nuclear Physics Parameters

Parameter	300 K 18cm		600 K 18 cm	600 K26 cm
	1E-9 MW	1.1 MW	1.1MW	1.1 MW
Neutron Lifetime	14.32 μs	14.32 μs	14.77 μs	14.62 μs
Neutron Generation time	53.92 μs	53.92 μs	54.83 μs	54.45 μs
Mean free path	1.011 cm	1.011 cm	1.003 cm	0.9964 cm
nu bar	2.439	2.439	2.439	2.4385

During reactor operation, ^{235}U is burned and fission products are generated, affecting reactor characteristics. Operation of a fully fueled core load was simulated as representative of changes to the operational core configuration over core life. At the end of core life with a k_{eff} of 1.005, neutron lifetime is 11.25 μs, neutron generation time 52.48 μs, and system mean free path 1.017 cm.

\

Table 29: Flux Density & Fission, Absorption & Leakage Fractions and Power

POWER	Temp.	Rod Position	keff	thermal flux	total flux	fissions	absorptions	leakage
1.00E-06	300 K	18	1.0037	1.07E+04	3.86E+04	1.00449	0.89233	0.107918
0.01	300 K	18	1.0037	1.07E+08	3.86E+08	1.00449	0.89233	0.107918
0.1	300 K	18	1.0037	1.07E+09	3.86E+09	1.00449	0.89233	0.107918
1	300 K	18	1.0037	1.07E+10	3.86E+10	1.00449	0.89233	0.107918
10	300 K	18	1.0037	1.07E+11	3.86E+11	1.00449	0.89233	0.107918
100	300 K	18	1.0037	1.07E+12	3.86E+12	1.00449	0.89233	0.107918
1000	300 K	18	1.0037	1.07E+13	3.86E+13	1.00449	0.89233	0.107918
1100	300 K	18	1.0037	1.18E+13	4.24E+13	1.00449	0.89233	0.107918
1100	600 K	18	0.9676	1.35E+13	4.53E+13	0.96865	0.890989	0.108998
1100	600 K	26	0.9982	1.35E+13	4.48E+13	0.99950	0.893075	0.107205

The flux density for the LCC core with fuel temperatures of 300 K is shown to be $1.19X10^{14}$ n/cm^2-s in the thermal range, and $4.30X10^{14}$ n/cm^2-s total, varying as $1.08X10^{7}$ n/cm^2-s per watt in the thermal range, and $3.91X10^{7}$ n/cm^2-s per watt above thermal energy range. These values agree well with calculations reported by General Atomics[12], where 2-D, 24 group calculations indicate average flux values for an 80 element 1 MW TRIGA to have $1.1X10^{7}$ n/cm^2-s per Watt from 0 to 1 eV and $2.46X10^{7}$ n/cm^2-s from 1 eV to 10 eV. At the end of core life as described, 1.1 MW power in the operational core requires a thermal neutron flux of $9.16X10^{12}$ n/cm^2-s, and total flux of $3.03X10^{13}$ n/cm^2-s.

5.2.2 Element Peaking Factors

As previously described, data from SCALE calculations to determine critical mass was used to evaluate maximum core peaking factors to support identification of the limiting core configuration, using the base geometry units as identified above. However, power within a fuel element is spatially distributed. Therefore it was necessary to use the base geometry units of Table 3 and the optional unit 200, segmented first axially then radially. First, the unit 200 was segmented into 20 axial sections (Table 30), then 21 radial segments (Table 31). The results of calculations using all base geometry units in Table 3 and alternate geometry unit 200 in position B01 are provided in Table 30 and Fig. 18 for axial distribution and Table 31 and Fig. 19 for radial distribution.

[12] GA-4361, Calculated Fluxes and Cross Sections for TRIGA Reactors, G. B. West (August 14, 1963)

Table 30: B01 Axial Peaking Factor

NODE	300 K	600 K
1	5.63E-01	5.64E-01
2	6.48E-01	6.40E-01
3	7.62E-01	7.24E-01
4	8.59E-01	8.86E-01
5	9.33E-01	9.80E-01
6	1.02E+00	1.01E+00
7	1.09E+00	1.12E+00
8	1.12E+00	1.12E+00
9	1.20E+00	1.16E+00
10	1.19E+00	1.20E+00
11	1.18E+00	1.18E+00
12	1.20E+00	1.17E+00
13	1.14E+00	1.16E+00
14	1.15E+00	1.13E+00
15	1.06E+00	1.07E+00
16	1.01E+00	9.66E-01
17	8.63E-01	9.09E-01
18	7.85E-01	7.60E-01
19	6.84E-01	6.65E-01
20	6.11E-01	6.04E-01

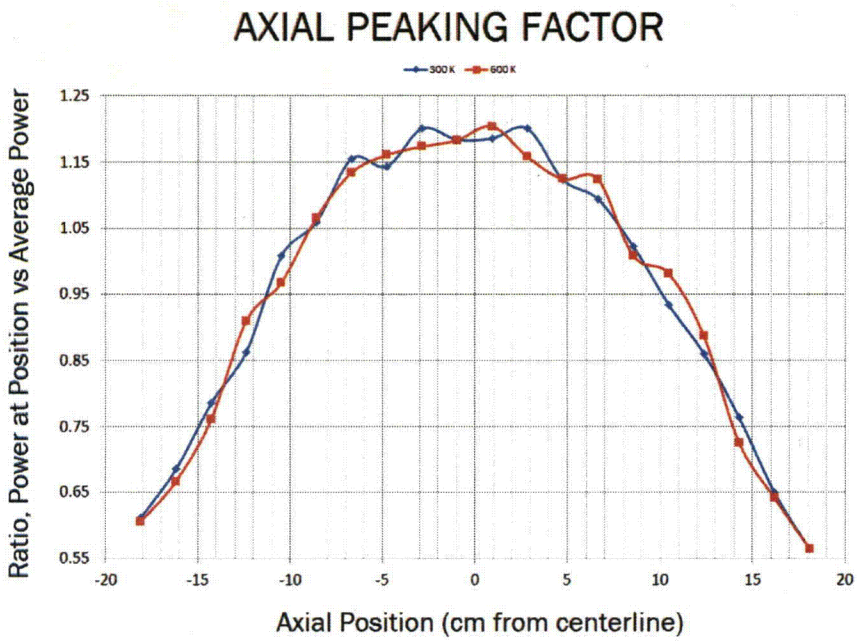

Figure 18, Axial Peaking Factor

Table 31: Fuel Element B01 Radial Peaking Factor

Radial Position			Temperature	
Outer	Inner	Ave.	300	600
1.826	1.753	1.790	1.972	2.088
1.753	1.680	1.716	1.853	1.918
1.680	1.606	1.643	1.718	1.732
1.606	1.533	1.569	1.583	1.599
1.533	1.459	1.496	1.481	1.484
1.459	1.386	1.423	1.380	1.365
1.386	1.313	1.349	1.284	1.271
1.313	1.239	1.276	1.183	1.168
1.239	1.166	1.202	1.096	1.067
1.166	1.092	1.129	1.022	0.989
1.092	1.019	1.056	0.933	0.910
1.019	0.946	0.982	0.849	0.828
0.946	0.872	0.909	0.779	0.763
0.872	0.799	0.835	0.713	0.701
0.799	0.725	0.762	0.649	0.634
0.725	0.652	0.689	0.574	0.570
0.652	0.579	0.615	0.513	0.503
0.579	0.505	0.542	0.441	0.437
0.505	0.432	0.468	0.387	0.383
0.432	0.358	0.395	0.324	0.323
0.358	0.285	0.322	0.267	0.266

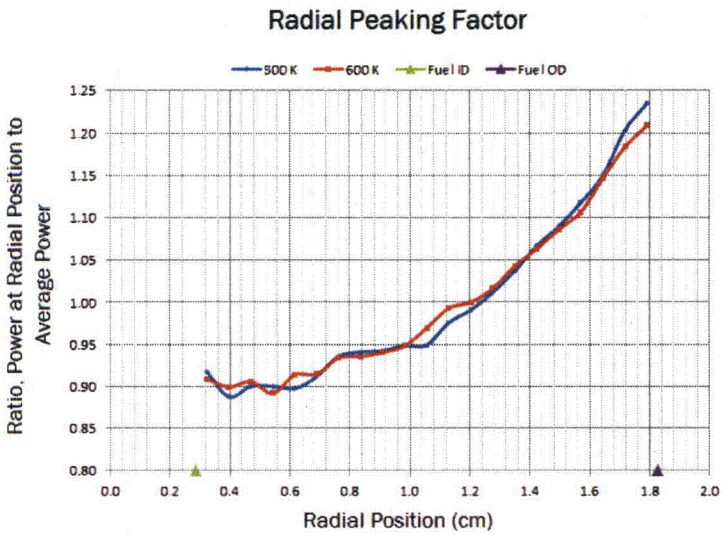

Figure 19, Element Radial Peaking factor

5.2.3 Burnup Effects

Calculations were performed to determine the mass of uranium isotopes (^{235}U and ^{238}U) in a single fuel element (Fig. 20) for two conditions. The uranium burnup for the initial LCC core was determined, and the mass for a core with all locations loaded with fuel to predict the operational core at end of core life. Excess reactivity was calculated from transport k_{eff}, and is shown in Fig. 21.

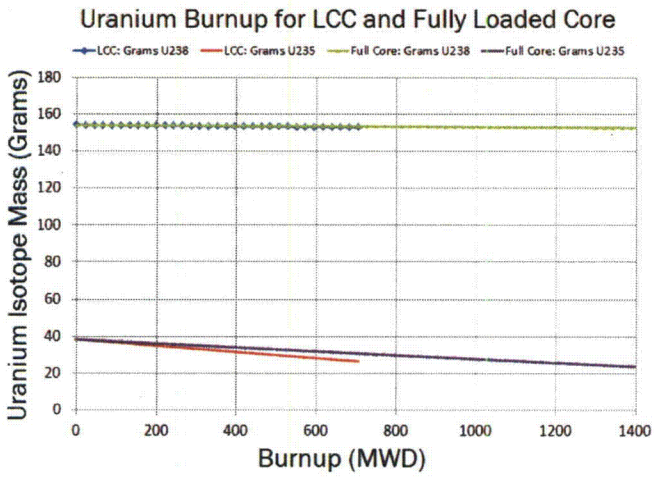

Figure 20: Uranium Burnup

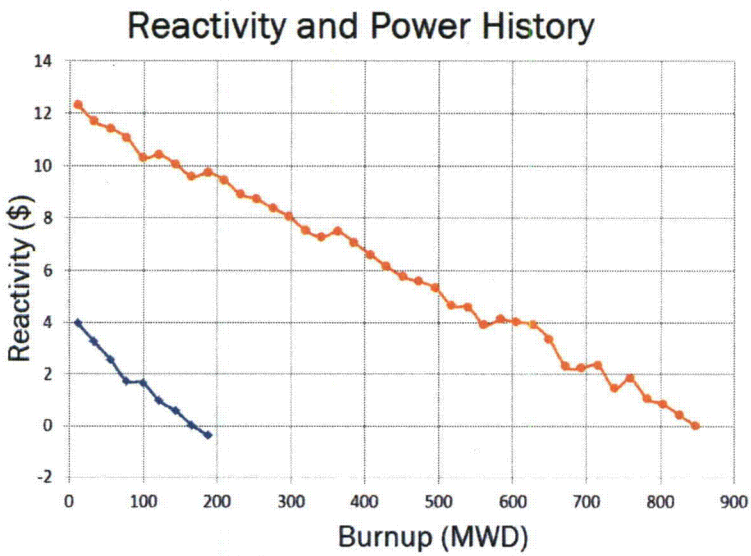

Figure 21: Excess reactivity and Burnup

The change in excess reactivity following startup from a clean core to equilibrium full power operation at 300 K was simulated using the base geometry units in Table 3. Excess reactivity was determined from KENO Transport K values. Full power equilibrium excess reactivity decrease attributed principally to fission product poisons is shown in Fig. 22 to result in a reactivity deficit of approximately $3.5. Excess reactivity following shutdown from the operation was simulated, with the results provided in Fig. 23.

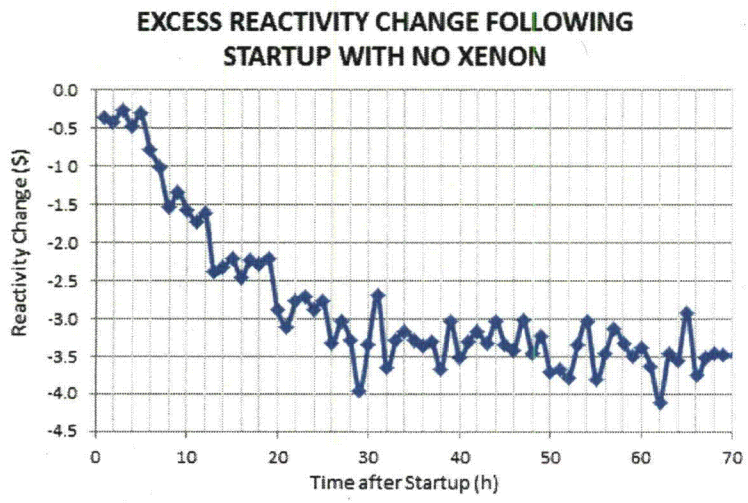

Figure 22, Excess Reactivity from Clean Core

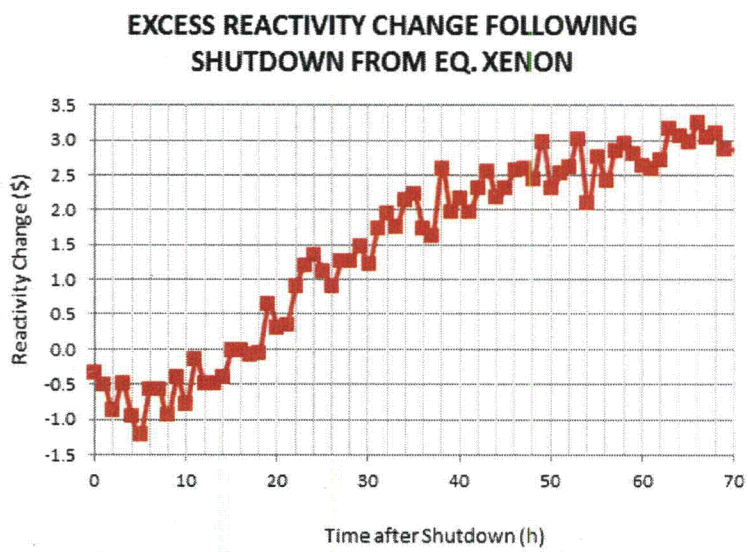

Figure 23: Excess Reactivity Following Shutdown

Simulation of steady state reactor operations at 1.1 MW from initial operation of the LCC to the operational end of core life was performed to predict the behavior of fission product poisons and transuranic isotopes on system performance over time. Absorptions are used as a proxy for reactivity, but ORIGEN calculations are infinite medium, neglecting leakage. All base geometry units in Table 3 were used, with OPUS reports for neutron absorption for selected fission product poisons, uranium and transuranic isotope. Near the beginning of core life, 85.8% of all absorptions occur in the isotopes of interest, as listed in Table 32. A large fraction (on the order of 80-85%) of absorptions in these isotopes occur in 5 isotopes, U235, U238, Pu239, Xe135, and Sm151. The change in absorption for these isotopes over long term, steady state, full power operation is shown in Fig. 24.

Table 32: Isotope Absorption-Fractions

ISOTOPE	0 MWD	66 MWD	198 MWD	264 MWD	396 MWD	462 MWD	550 MWD
U235	8.02E-01	7.67E-01	7.46E-01	7.36E-01	7.14E-01	7.03E-01	6.88E-01
U238	5.93E-02	5.84E-02	6.04E-02	6.15E-02	6.38E-02	6.50E-02	6.68E-02
Pu238	2.46E-17	1.50E-08	3.66E-07	8.60E-07	2.95E-06	4.76E-06	8.25E-06
Pu239	6.79E-17	3.77E-03	1.15E-02	1.53E-02	2.27E-02	2.63E-02	3.09E-02
Pu240	5.46E-17	3.17E-05	2.97E-04	5.29E-04	1.19E-03	1.61E-03	2.28E-03
Pu241	7.54E-17	8.19E-07	2.35E-05	5.55E-05	1.85E-04	2.91E-04	4.86E-04
Pu242	7.03E-18	3.51E-10	3.24E-08	1.05E-07	5.55E-07	1.05E-06	2.16E-06
Xe131	8.24E-18	1.19E-04	4.25E-04	5.86E-04	9.21E-04	1.10E-03	1.34E-03
Xe133	9.14E-18	4.79E-05	4.96E-05	5.05E-05	5.24E-05	5.34E-05	5.49E-05
Xe135	1.49E-13	2.41E-02	2.43E-02	2.43E-02	2.44E-02	2.44E-02	2.45E-02
Sm147	6.33E-18	1.14E-06	1.41E-05	2.60E-05	6.06E-05	8.33E-05	1.19E-04
Sm150	5.97E-18	2.75E-05	1.12E-04	1.57E-04	2.53E-04	3.03E-04	3.74E-04
Sm151	5.75E-16	1.15E-03	2.33E-03	2.60E-03	2.89E-03	2.96E-03	3.02E-03
Sm152	2.30E-17	5.08E-05	2.09E-04	3.06E-04	5.19E-04	6.33E-04	7.92E-04
Sm153	4.27E-17	2.06E-06	2.50E-06	2.76E-06	3.37E-06	3.72E-06	4.21E-06
TOTAL	86.1%	85.5%	84.6%	84.1%	83.1%	82.6%	81.9%

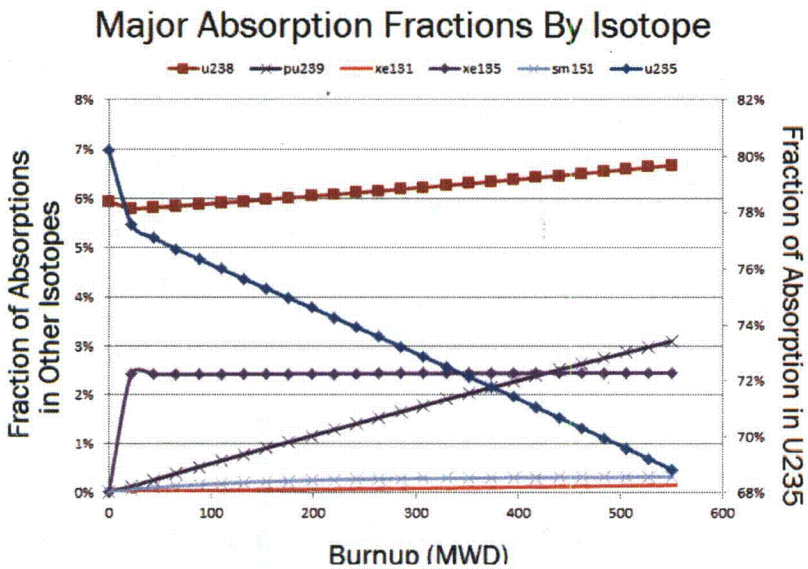

Figure 24, Neutron Absorption in Major Isotopes

5.2.4 Fuel Temperature Reactivity Coefficient and Excess Reactivity

As previously described, data from SCALE calculations using base units of Table 3 was performed to determine critical mass across a range of temperatures from 300 K to 600 K. The transport k_{eff} in each configuration and temperature variation was used to calculate net core reactivity, as reported in Table 33. The values for the LCC were plotted in Fig. 25 (along with other core configurations near the LCC). The response for all data was remarkably similar, providing confidence that the temperature coefficient of reactivity can reliably be determined by the slope of the linear function relating temperature to reactivity change. The value of $-0.0127/Δ°K is approximately 11% lower than the value reported in the

original UT TRIGA SAR for a critical configuration of 64 elements. A similar fit for the 65 element core results in agreement with the original UT SAR to within about 6%.

Table 33: Excess Reactivity at Fuel temperature By Core Configuration:

Fuel Temp (K)	Number of Fuel Elements										
	65	73	74	78	83	87	89	92	97	100	106
	Graphite Configuration										
300	$1.61	$3.24	$4.19	$4.70	$5.64	$6.58	$6.83	$7.46	$8.24	$8.61	$9.55
300	$1.62	$3.29	$4.04	$5.04	$5.73	$6.50	$6.87	$7.36	$8.39	$8.71	$9.47
350	$1.16	$3.05	$3.38	$4.44	$5.32	$5.98	$6.44	$7.18	$8.02	$8.31	$8.97
450	-$0.03	$1.72	$2.32	$3.38	$4.12	$4.99	$5.63	$5.92	$6.80	$7.20	$8.09
550	-$1.65	$0.31	$1.04	$2.08	$2.75	$3.76	$4.04	$4.71	$5.45	$5.98	$6.71
600	-$2.29	-$0.36	$0.11	$1.27	$2.17	$2.88	$3.21	$3.85	$4.86	$5.07	$6.06
	Water Void Configuration										
300	-$2.81	-$0.21	$0.68	$1.94	$2.83	$4.13	$4.72	$5.49	$6.74	$7.47	$8.67
300	-$2.77	-$0.24	$0.77	$1.90	$3.03	$4.24	$4.82	$5.47	$6.92	$7.41	$8.64
350	-$3.53	-$0.83	$0.29	$1.34	$2.37	$4.04	$4.39	$4.95	$6.39	$7.00	$8.23
450	-$4.65	-$2.34	-$0.98	$0.30	$1.37	$2.81	$3.09	$4.11	$5.34	$5.89	$7.15
550	-$6.17	-$3.60	-$2.56	-$1.15	-$0.04	$1.23	$1.86	$2.73	$4.09	$4.43	$5.88
600	-$6.98	-$4.46	-$3.41	-$1.81	-$0.86	$0.60	$1.15	$1.93	$3.38	$3.95	$5.11

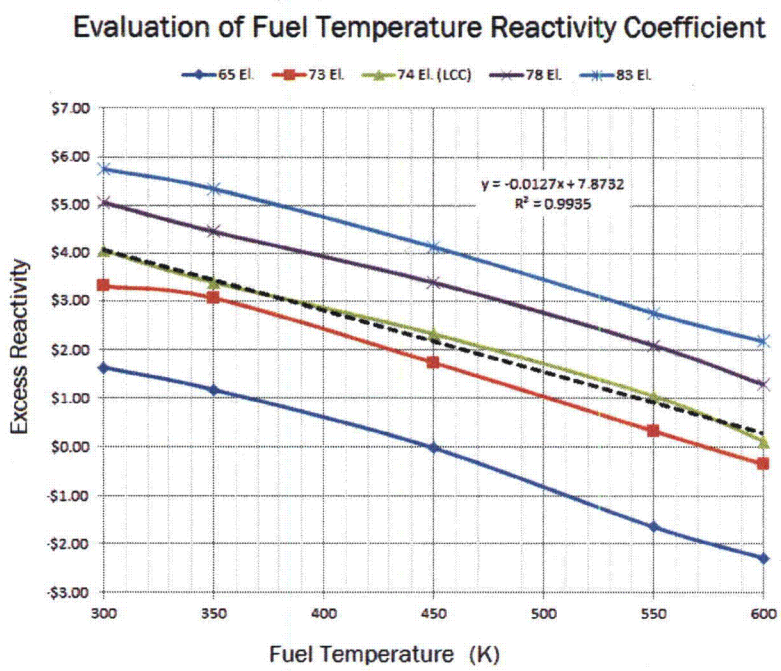

Figure 25: Excess Reactivity and Fuel Temperature

5.2.5 Excess reactivity & Shutdown Margin

The base units in Table 3 and the fuel follower option Unit 8) were used in the Limiting Core Condition (i.e., 74 element core) and a fully filled core lattice (i.e., 117 fuel elements installed, water voids in space

reserved for source and pneumatic transit facility). For the LCC core, calculations were performed using material files generated following simulations of from 50 MWD, 100 MWD, 200 MWD and 300 MWD power histories. Calculations were performed to determine excess reactivity with the following conditions:

1) all control rods positioned at 38.1 cm;

2) all control rods positioned at 0 cm;

3) all control rods except the transient rod at 38.1 cm and the transient rod at 0 cm;

4) all control rods except shim rod 1 at 38.1 cm and shim rod 1 at 0 cm;

5) all control rods except shim rod 2 at 38.1 cm and shim rod 2 at 0 cm;

Excess reactivity was determined directly from transport k_{eff} for the "all rods out" condition. The worth imposed by control rod insertion was calculated for the remaining conditions using the formula (where insertion is understood to cause negative reactivity worth):

$$\$_{CR} = \frac{k_{eff,ARO} - k_{eff,CRin}}{k_{eff,ARO} \cdot k_{eff,CRin}} \cdot \frac{1}{0.007}$$

The limiting shutdown margin (LSDM) is calculated as the shutdown margin with the most reactive control rod fully withdrawn. In all cases, the regulating rod was determined to be the most reactive control rod. The limiting shutdown margin was calculated in three separate ways. The label "LSDM 1" indicates a value based on k_{eff} from transport calculations configured with transient rod, shim 1 and shim 2 fully inserted, and the regulating rod fully withdrawn. This directly represents the limiting control rod configuration and is therefore has the most confidence. Values labeled "SDM" were calculated by assuming the reactivity worth of all rods out, decreased by the reactivity worth of inserting the transient rod, shim 1, and shim 2. Since control rod worths are interdependent, this is less representative of the core configuration but closely matches experimental determination of the limiting shutdown margin. The results are provided in Table 36.

Table 36: LCC Control Rod and Excess Reactivity ($) and Burnup

MWD	8.3	10	15	20	30	100
ARO	6.94	6.71	6.81	6.32	6.05	4.05
ARI	16.90	16.72	17.05	17.05	17.05	17.20
RR	4.36	4.10	4.57	4.37	4.37	4.20
TR	3.05	3.31	3.28	3.37	3.37	3.05
SH1	2.97	2.63	3.08	3.10	3.10	2.68
SH2	2.80	2.64	3.27	3.12	3.12	10.95
ROD-SUM	13.18	12.68	14.21	13.95	13.95	20.87
LSDM	3.82	3.75	4.21	4.67	4.67	6.90
SDM	1.87	1.88	2.83	3.27	3.53	12.62
ARO-ARI	9.96	10.01	10.24	10.73	11.00	13.15

5.2.6 Burnup effects

Over time, the limiting core configuration will require increasing the number of fuel elements to compensate for burnup until all locations reserved for fuel are filled. Therefore, burnup calculations are based on a full core load. A simulation of steady state reactor operations at 1.1 MW from initial operation to end of core life provides data on flux density, changes in uranium mass, and the effects of significant isotopes generated during operation over core life. Thermal and total neutron flux in each material is calculated by KENO; the average flux values for the ZrH fuel material shown in Fig. 26. The total ^{235}U and ^{238}U mass at each burnup interval is calculated by ORIGEN, and reported by OPUS as indicated in Fig. 27. Excess reactivity derived from transport calculations over core life are provided in Fig. 28. Neutron absorption in ^{235}U and ^{238}U (as a fraction of total absorptions) is shown in Fig. 29. The absorption fractions for other significant isotopes generated during operations are shown in Fig. 30.

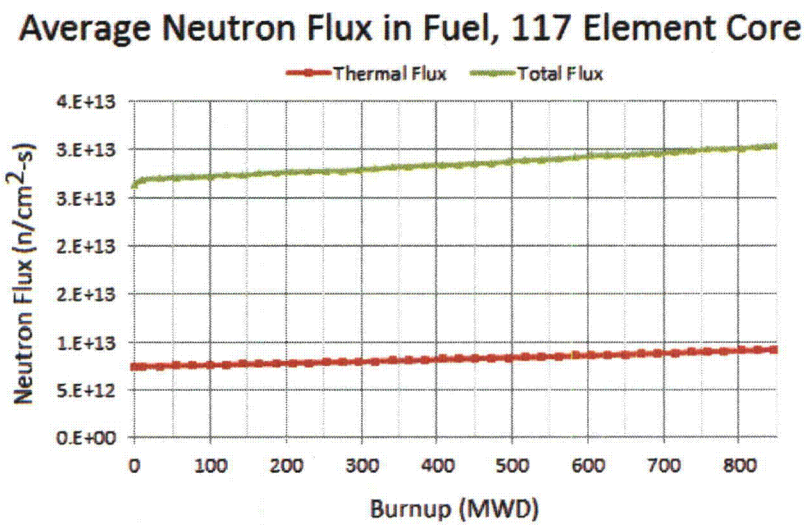

Figure 26: Neutron Flux in Fuel of a 117 Element Core as a Function of Burnup

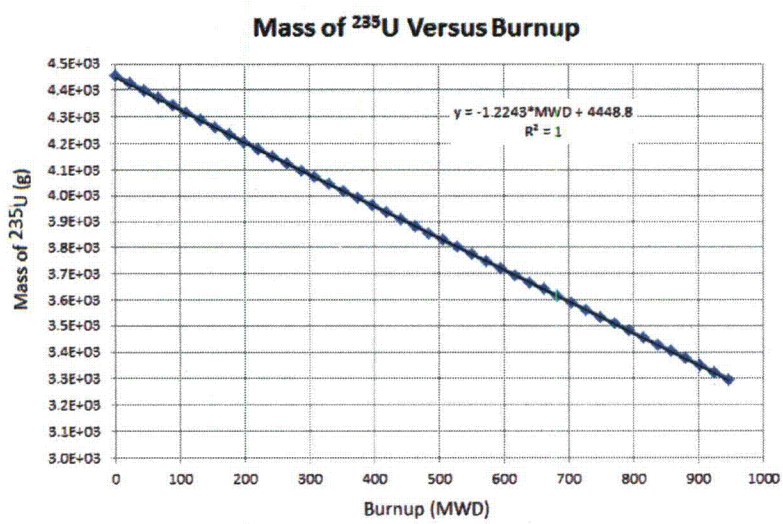

Figure 27: Uranium (235 and 238) Mass in a 117 Element Core as a Function of Burnup

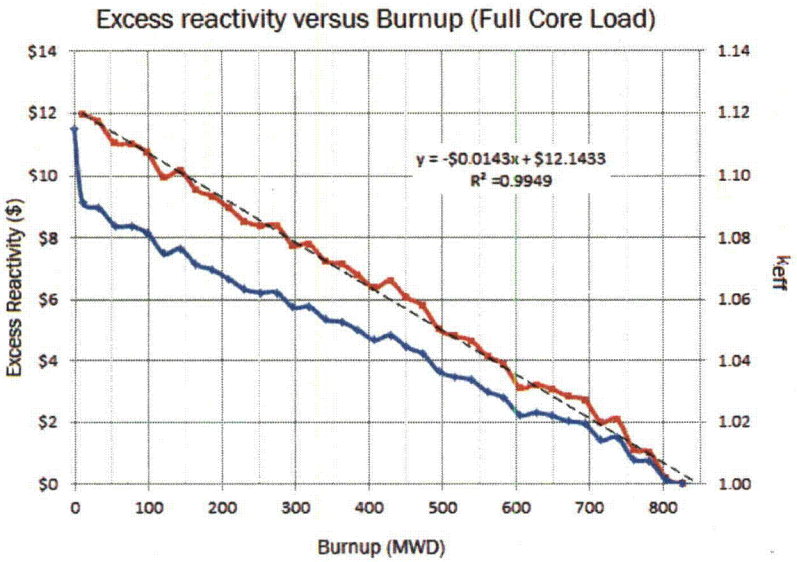

Figure 28: Excess Reactivity in a 117 Element Core as a Function of Burnup

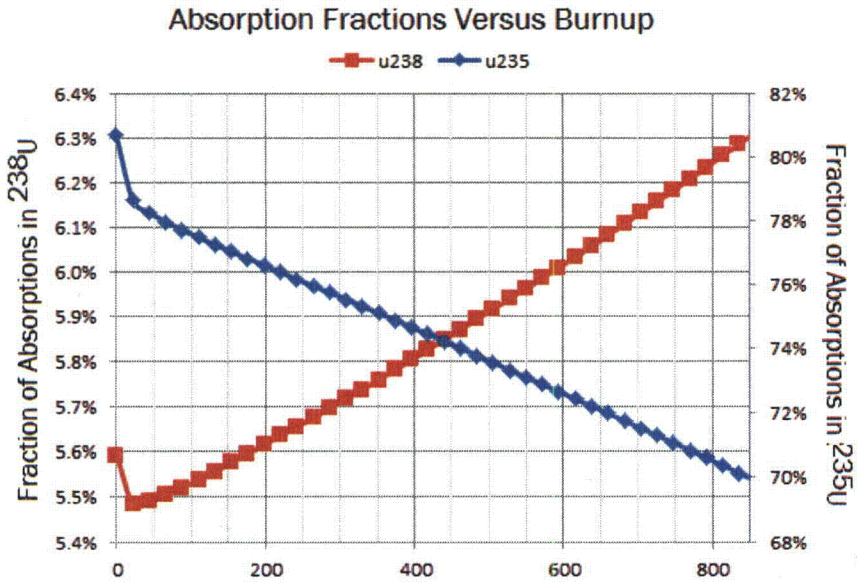

Figure 29: Fraction of Neutrons Absorbed in ^{235}U and ^{238}U (117 Element Core) vs. Burnup

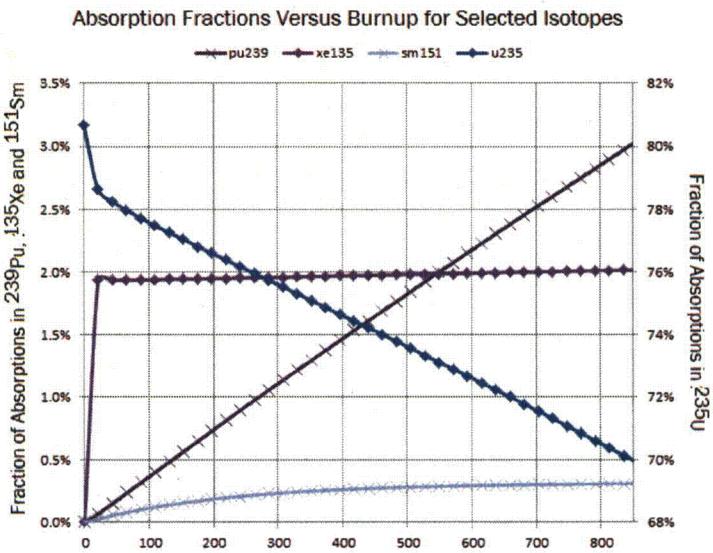

Figure 30: Fraction of Neutrons Absorbed in Selected Isotopes (117 Element Core) vs. Burnup

5.2.7 Experiment effects

As shown in Table 36, actual shutdown margin based on k_{eff} as determined with the most reactive rod fully withdrawn with fresh fuel is shown to be $3.82. However, the method of calculating excess reactivity consistently underestimates excess reactivity by a factor of two. Therefore the limit on experiment worth provides assurance that excess reactivity limits with the most reactive shutdown margin fully with draw is met.

5.2.8 Accident Source terms

A simulation was performed with a fully loaded core (all core spaces filled with fresh standard fuel elements) operated at 1.1 MW until k_{excess} was less than unity. The simulation then decayed the core for 20 minutes, simulating the amount of time after shutdown required to remove a fuel element from the core. Activity of the major halogens and iodine isotopes was calculated for strategic time intervals, and is provide in Table 37. Total decay heat was calculated for time intervals after shutdown (Fig. 30).

Table 37: Fission product Inventory, Maximum Single Fuel Element

TIME	20 M	50 M	7.5 H	11.5 H	12 H	1 D	7 D	30 D	180 D	365 D
br82	1.82E-1	1.81E-1	1.80E-1	1.57E-1	1.45E-1	1.14E-1	6.75E-3	1.32E-7	1.64E-11	1.63E-11
br83	6.37E1	6.17E1	5.61E1	7.75	2.44	7.61E-2	5.80E-9	5.80E-9	5.80E-9	5.80E-9
br84m	2.25	2.17E-1	6.67E-3	2.05E-10	2.05E-10	2.05E-10	2.05E-10	2.05E-10	2.05E-10	2.05E-10
br84	1.12E2	7.93E1	4.12E1	4.44E-3	2.35E-5	1.02E-8	1.02E-8	1.02E-8	1.02E-8	1.02E-8
br85	1.61E2	1.51	1.12E-3	1.46E-8	1.47E-8	1.47E-8	1.47E-8	1.47E-8	1.47E-8	1.47E-8
br86	1.98E2	5.78E-5	1.80E-8	1.80E-8	1.80E-8	1.80E-8	1.80E-8	1.80E-8	1.80E-8	1.80E-8
br87	2.39E2	6.58E-5	2.17E-8	2.17E-8	2.17E-8	2.18E-8	2.18E-8	2.18E-8	2.18E-8	2.18E-8
i131	3.56E2	3.56E2	3.55E2	3.48E2	3.43E2	3.30E2	1.98E2	2.72E1	6.38E-5	3.24E-8
i132	5.34E2	5.34E2	5.32E2	5.04E2	4.86E2	4.37E2	1.19E2	8.23E-1	4.82E-8	4.82E-8
i133	8.05E2	8.02E2	7.94E2	6.37E2	5.57E2	3.74E2	3.08	1.05E-7	7.37E-8	7.37E-8
i134	9.40E2	8.85E2	7.44E2	7.12	3.28E-1	2.62E-5	8.57E-8	8.57E-8	8.57E-8	8.57E-8
i135	7.60E2	7.33E2	6.96E2	3.33E2	2.18E2	6.15E1	1.56E-5	6.92E-8	6.92E-8	6.92E-8

Table 37: Fission product Inventory, Maximum Single Fuel Element

TIME	20 M	50 M	7.5 H	11.5 H	12 H	1 D	7 D	30 D	180 D	365 D
i136	3.15E2	1.46E-2	3.31E-8	2.87E-8	2.87E-8	2.87E-8	2.87E-8	2.87E-8	2.87E-8	2.87E-8
kr83m	6.28E1	6.27E1	6.19E1	1.94E1	7.30	2.86E-1	2.67E-7	2.23E-7	7.07E-8	2.04E-8
kr85m	1.55E2	1.49E2	1.38E2	4.69E1	2.52E1	3.94	1.50E-8	1.42E-8	1.42E-8	1.42E-8
kr85	9.59	9.59	9.59	9.59	9.59	9.59	9.58	9.54	9.29	9.00
kr87	3.00E2	2.52E2	1.92E2	4.26	4.79E-1	6.91E-4	2.73E-8	2.73E-8	2.73E-8	2.73E-8
kr88	4.06E2	3.74E2	3.31E2	6.01E1	2.26E1	1.21	3.70E-8	3.70E-8	3.70E-8	3.70E-8
kr89	5.18E2	6.03	7.96E-3	4.72E-8	4.72E-8	4.72E-8	4.72E-8	4.72E-8	4.72E-8	4.73E-8
xe131m	4.25	4.25	4.25	4.25	4.25	4.24	3.87	1.61	3.49E-4	7.23E-9
xe133m	8.93	8.93	8.93	8.79	8.66	8.09	1.57	1.10E-3	8.16E-10	8.16E-10
xe135	3.55E2	3.65E2	3.78E2	4.19E2	3.80E2	2.20E2	6.79E-3	7.26E-8	7.26E-8	7.26E-8
xe137	7.38E2	1.97E1	8.28E-2	6.71E-8	6.71E-8	6.72E-8	6.72E-8	6.72E-8	6.72E-8	6.72E-8
xe138	7.50E2	2.77E2	6.28E1	1.37E-7	6.82E-8	6.83E-8	6.83E-8	6.83E-8	6.83E-8	6.83E-8

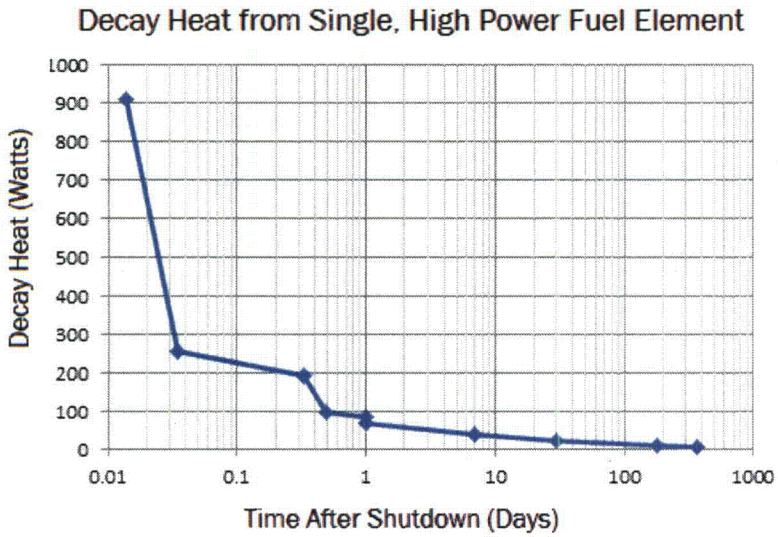

Figure 30, Decay Heat Following Steady State Full Power, End of Life Operations

6.0 Model Validation

Validity of the SCALE model application to the UT TRIGA reactor is verified in two ways. First, the prediction of mass required to support operations is compared to the 1992 operational core loading. Second, data from model calculations is used to evaluate core excess reactivity and the reactivity worth of individual control rods with values compared to measured data for the 1992 operational core.

6.1 Model Comparison with Historical Reactors

As previously shown (Fig. 7), the SCALE model predicts criticality with fresh fuel and graphite moderation to be 60 elements, and 86 for water moderation. A fully operational core was assembled which included 84 lightly burned standard fuel elements and three fresh fuel followers augmented with 18 graphite rods, leaving 13 water voids in the G ring. According to the figure, SCALE predicts excess reactivity (for 87 fuel elements, 84 standard and 3 followers) of approximately $1.00 with fresh fuel and

a water-void configured core, $7.00 with a full graphite-configured core. The lightly burned fuel with a measured excess reactivity of approximately $6.4 is within the predicted range for fresh fuel.

6.2 Reactivity Values

Data from calculations using the SCALE model of the UT TRIGA reactor was used to determine excess reactivity and the total reactivity worth of each control rod. Excess reactivity was calculated based on k_{eff} with the control rods in a fully withdrawn position ($k_{eff,ARO}$). The worth of an individual control rod ($\$_{CR}$) was based on the previous calculation and k_{eff} calculation with the control rods in a fully inserted position ($k_{eff,CRin}$) as follows:

$$\$_{CR} = \frac{k_{eff,ARO} - k_{eff,CRin}}{k_{eff,ARO} \cdot k_{eff,CRin}} \cdot \frac{1}{0.007}$$

Reactivity values calculated from data generated using the SCALE model for control rod worth and excess reactivity measurements are compared to measurements accomplished in 1992. The 1992 core ^{235}U mass and enrichment of the 1992 are used to adjust material composition in the SCALE model to reflect conditions representative of the fuel loaded in the 1992 core. All of the standard fuel elements used in 1992 had prior power history, lightly burned at the previous UT TRIGA reactor at Taylor Hall or at General Atomics facilities. Some of the elements had a power history at GA facilities followed by operation at the UT Taylor Hall facility. All of the fuel elements decayed approximately one year following removal prior to installation at the current UT TRIGA located at the NETL.

Although records indicate significant variation for fuel burnup, 30 of the inner ring elements had power history of approximately 0.243 MWD per element from operation at the Taylor Hall reactor on the UT campus. The average burnup had a standard deviation of 0.035 with a maximum of 0.304 MWD and a minimum of 0.144 MWD at Taylor Hall; a single element in this group had 0.657 MWD burnup from a GA facility (C11 position). Fifty-four of the outer elements were documented to have burn of approximately 0.676 MWD per element at GA facilities, with one exception, a single element with 30% higher burnup. The average burnup of this group had a standard deviation of 0.025 MWD with a maximum value of 0.752 MWD and a minimum of 0.619 MWD.

As indicated in Fig. 3, two of the elements with the higher average burnup value are located in a ring with the inner set of elements with lower burnup for the remaining standard fuel elements; to simplify the description, all elements with the lowest average power history will be referred to as "inner elements" and elements with the higher power history will be referred to as "outer elements."

A series of calculations was accomplished to determine isotopic concentrations of fuel at various burnup intervals. The power history for the previous UT TRIGA is complex. The UT records did not indicate variation expected from operation in various rings of the Taylor Hall circular TRIGA core, i.e. based on different core-radial peaking factors. Facility records report all of the Taylor Hall fuel to have initially contained 38 grams of ^{235}U; the lack of variation in assay values is unusual. UT does not have records relating the operation of the individual fuel elements to specific GA reactors, which do not all have the same rated power. The UT records do not contain information about the power level for the burnup of the GA fuel. Assumptions required to accurately model fuel composition based on prior power history are therefore extremely challenging.

Lacking information that would fully characterize the power history of the lightly burned full in the 1992 UT TRIGA core, all previous operation is assumed to have occurred at 250 kW. Simulations were made assuming 250 kW fresh TRIGA fuel in the 1992 core configuration, operated to target burnup values. The fresh fuel composition was based on the average value of ^{235}U mass and enrichment from special nuclear material inventory records. Target burnup value was based on total power generation for all of the lightly burned elements. Initial attempts to model the lightly burned fuel composition resulted in calculated excess reactivity much greater than excess reactivity measured in 1992.

Since the power history for 30 elements in the inner rings was approximately 1/3 of the power history for the outer elements and fuel followers were fresh fuel in high reactivity worth positions, three different material specifications were developed to simulate the core composition more accurately. One material specification was used to define fresh fuel, one to simulate material with average burnup for the inner elements, and one to simulate material with average burnup for the outer elements. While the results showed excess reactivity much closer to the measured 1992 values, excess reactivity calculations still greatly exceeded measured values.

Based on uncertainty in characterization of power history, previously developed material specifications were used in a set of calculations with the outer element burn three times the value of the inner element burn. With inner element material specification based on an average burnup of 0.69 MWD per element, the reactivity values calculated from the model show general agreement with measured values.

Reactivity values calculated from SCALE data and two sets of reactivity measurements from 1992 (immediately following core loading on 3/25/1992, and at the concluding of physics testing on 7/23/1992) are provided in Table 38. The column $\Delta\$_{ref}$ is the difference between the measured 1992 value and the value calculated from SCALE data. The $\Delta\$_{ref}/\$$ column is the ratio of the difference to the reference 1992 value (expressed as a per cent).

Table 38: 1992 UT TRIGA REACTIVITY ($) SCALE, SURV-6 & SURV 3

PARAMETER	SCALE	03/25/1992	$\Delta\$_{ref}$	$\Delta\$_{ref}/\$$	07/23/1992	$\Delta\$_{ref}$	$\Delta\$_{ref}/\$$
EXCESS	$6.76	NA	NA	NA	$6.38	$0.38	0.2%
REG ROD	$4.25	$4.59	$0.34	-7.5%	$4.08	$0.17	4.2%
TR ROD	$3.01	$3.34	$0.53	-15.1%	$3.26	-$0.26	-7.8%
SHIM 2	$3.26	$3.46	$0.20	-5.6%	$3.30	-$0.04	-1.2%
SHIM 1	$3.02	$3.32	$0.30	-8.9%	$3.17	-$0.15	-4.8%
SUM RODS	$13.53	$14.90	$1.37	9.2%	$13.82	-$0.28	0.04

6.3 Conclusion

Core loading for the 1992 fuel using a fraction of non-fuel spaces using graphite rods is consistent with calculations of loading to support full power operation. Flux density calculated for the LCC (74 element, graphite rod configured) core agrees with historical calculations performed by General Atomics. There is good agreement between measured and calculated reactivity worth values when nominal material specification is adjusted to bring excess reactivity into agreement. Therefore, results of calculations using the SCALE UT TRIGA model provide confidence that the model is capable of adequately predicting reactor performance.

ATTACHMENT 2: UT TRIGA Historical Core Data

Table 1 provides information about the UT TRGIA core in 1992. Initial critical was achieved with standard fuel elements in positions F06 and F11, but these elements were removed to reduce excess reactivity on 3/19/1992 prior to completion of physics testing. In the TYPE column: SFE indicates standard fuel element, IFE indicates instrumented fuel element, and FFCR indicates fuel follower. The initial ^{235}U mass (g) is provided as INIT. 235, and the mass in the element at initial criticality is 1992 235. Most of the fuel had prior power history, represented as GA BRN for fuel lightly irradiated at GA reactors and TH BRN for power history at the Taylor Hall TRIGA reactor located on the UT main campus.

Table 1: UT TRIGA 1992 Core Inventory (Physics Testing)

POS	ID	TYPE	INIT. 235	1992 235	GA BRN	TH BRN	POS	ID	TYPE	INIT. 235	1992 235	GA BURN	TH BURN
B01	5921	SFE	38.00	37.76		0.233	E10	2912	SFE	38.95	38.24	0.676	
B02	5922	SFE	38.00	37.76		0.225	E11	2913	SFE	40.11	39.38	0.695	
B03	5981	IFE	38.00	37.71		0.277	E12	2915	SFE	39.04	38.33	0.676	
B04	6143	SFE	38.00	37.85		0.144	E13	2918	SFE	38.68	37.97	0.676	
B05	6886	SFE	38.00	37.76		0.233	E14	2925	SFE	36.76	36.07	0.638	0.014
B06	6889	SFE	38.00	37.76		0.233	E15	2927	SFE	40.23	39.50	0.695	
C01							E16	2928	SFE	37.68	36.99	0.657	
C02	5916	SFE	38.00	37.78		0.207	E17	2929	SFE	38.35	37.65	0.667	
C03	5917	SFE	38.00	37.80		0.190	E18	2930	SFE	39.78	39.05	0.695	
C04	6924	SFE	38.00	37.76		0.233	E19	2932	SFE	37.76	37.07	0.657	
C05	6926	SFE	38.00	37.76		0.233	E20	2935	SFE	40.03	39.30	0.695	
C06	6927	SFE	38.00	37.76		0.233	E21	2938	SFE	40.11	39.38	0.695	
C07	10148	FFCR	32.26	32.26			E22	2939	SFE	37.40	36.72	0.648	
C08	6928	SFE	38.00	37.76		0.233	E23	2940	SFE	39.41	38.69	0.686	
C09	6929	SFE	38.00	37.76		0.233	E24	2941	SFE	36.78	36.09	0.638	0.014
C10	6930	SFE	38.00	37.76		0.233	F01	2944	SFE	36.58	35.89	0.638	0.014
C11	5283	IFE	38.00	37.31	0.657		F02	2946	SFE	38.79	38.08	0.676	
C12	6932	SFE	38.00	37.76		0.233	F03	2947	SFE	39.37	38.65	0.686	
D01	5844	SFE	38.00	37.68		0.304	F04	2948	SFE	40.57	39.83	0.705	
D02	5845	SFE	38.00	37.68		0.304	F05	2950	SFE	38.34	37.64	0.667	
D03	5846	SFE	38.00	37.68		0.304	F06		GR				
D04	5902	SFE	38.00	37.69		0.296	F07	2952	SFE	42.43	41.66	0.733	
D05	5903	SFE	38.00	37.69		0.296	F08	2954	SFE	40.04	39.31	0.695	
D06	10146	FFCR	31.57	31.57			F09	2955	SFE	38.40	37.68	0.667	0.014
D07	5904	SFE	38.00	37.70		0.283	F10	2957	SFE	40.17	39.44	0.695	
D08	5912	SFE	38.00	37.76		0.233	F11		GR				
D09	5913	SFE	38.00	37.76		0.233	F12	2959	SFE	37.38	36.70	0.648	
D10	5914	SFE	38.00	37.76		0.233	F13	2960	SFE	39.59	38.85	0.686	0.014
D11	5915	SFE	38.00	37.76		0.233	F14	2962	SFE	37.64	36.95	0.657	
D12	5918	SFE	38.00	37.76		0.233	F15	2964	SFE	37.22	36.54	0.648	
D13	5919	SFE	38.00	37.76		0.233	F16	2965	SFE	36.34	35.66	0.629	0.014
D14	10147	FFCR	31.63	31.63			F17	2968	SFE	40.02	39.29	0.695	
D15	5920	SFE	38.00	37.76		0.225	F18	2969	SFE	39.11	38.40	0.676	

Table 1: UT TRIGA 1992 Core Inventory (Physics Testing)

POS	ID	TYPE	INIT. 235	1992 235	GA BRN	TH BRN	POS	ID	TYPE	INIT. 235	1992 235	GA BURN	TH BURN
D16	6142	SFE	38.00	37.71		0.277	F19	2970	SFE	40.23	39.50	0.695	
D17	6923	SFE	38.00	37.76		0.233	F20	2971	SFE	39.58	38.86	0.686	
D18	6925	SFE	38.00	37.76		0.233	F21	2974	SFE	38.04	37.35	0.657	
E01	2899	SFE	36.17	35.49	0.629	0.014	F22	2975	SFE	38.50	37.78	0.667	0.014
E02	2902	SFE	39.04	38.33	0.676		F23	2976	SFE	40.41	39.67	0.705	
E03	2903	SFE	40.05	39.30	0.695	0.014	F24	2977	SFE	37.82	37.13	0.657	
E04	2904	SFE	42.72	41.94	0.743		F25	2979	SFE	36.91	36.22	0.638	0.014
E05	2905	SFE	38.15	37.43	0.667	0.014	F26	2983	SFE	36.72	36.03	0.638	0.014
E06	2906	SFE	37.72	37.03	0.657		F27	2894	SFE	42.92	42.87	0.051	
E07	2908	SFE	40.30	39.56	0.705		F28	2985	SFE	35.74	35.07	0.619	0.014
E08	2910	SFE	37.27	36.59	0.648		F29	5198	SFE	40.00	35.07	4.695	
E09	2911	SFE	38.73	38.00	0.676	0.014	F30	3513	SFE	39.00	38.01	0.752	0.190

The core inventory as indicated in Table 2 in 2012 is taken from special nuclear material records, with the power history indicated as MWD.

Table 2: Core Inventory, 2012

POS	ID	TYPE	MWD	POS	ID	TYPE	MWD	POS	ID	TYPE	MWD
B01	2985	SFE	3.295	E05	6886	SFE	2.894	F21	2971	SFE	3.347
B02	3384	SFE	2.130	E06	5912	SFE	2.894	F22	2969	SFE	3.337
B03	10878	IFE	2.297	E07	5846	SFE	2.965	F23	6926	SFE	2.894
B04	3013	SFE	3.532	E08	5903	SFE	2.957	F24	3513	SFE	3.603
B05	2899	SFE	3.304	E09	5917	SFE	2.851	F25	10811	SFE	1.854
B06	10708	IFE	2.490	E10	6929	SFE	2.894	F26	2960	SFE	3.361
C01		TR		E11		3EL		F27	2947	SFE	3.347
C02	2965	SFE	3.304	E12	6925	SFE	2.894	F28	2911	SFE	3.352
C03	2984	SFE	3.290	E13	5844	SFE	2.965	F29	5922	SFE	2.886
C04	2944	SFE	3.314	E14	6923	SFE	2.894	F30	10814	SFE	1.854
C05	2931	SFE	2.837	E15	5919	SFE	2.894	G02	10704	SFE	1.475
C06	2983	SFE	3.314	E16	5921	SFE	2.894	G03	2908	SFE	3.366
C07	10148	FFCR	2.661	E17	6927	SFE	2.894	G04	3700	SFE	2.633
C08	2980	SFE	2.924	E18	5902	SFE	0.296	G05	6931	SFE	0.698
C09	2925	SFE	3.314	E19	5904	SFE	2.944	G06	5920	SFE	2.886
C10	2941	SFE	3.314	E20	6930	SFE	2.894	G08	10701	SFE	2.213
C11	2979	SFE	3.314	E21	6889	SFE	2.894	G09	2957	SFE	3.356
C12	2964	SFE	3.309	E22	5914	SFE	2.894	G10	2938	SFE	3.189
D01	2910	SFE	3.309	E23	6142	SFE	2.938	G11	2927	SFE	3.356
D02	2959	SFE	3.309	E24	6928	SFE	2.894	G12	10702	SFE	2.308
D03	2906	SFE	3.318	F01	10817	SFE	1.854	G14	2970	SFE	3.356
D04	2992	SFE	3.742	F02	5911	SFE	2.439	G15	2976	SFE	3.366
D05	2962	SFE	3.318	F03	3496	SFE	2.681	G16	2952	SFE	2.413
D06	10146	FFCR	2.661	F04	3504	SFE	2.671	G17	10815	SFE	1.854
D07	2928	SFE	3.318	F05	3703	SFE	2.681	G18	2904	SFE	3.404
D08	2939	SFE	3.309	F06	10816	SFE	1.854	G20	2968	SFE	3.356

Table 2: Core Inventory, 2012

POS	ID	TYPE	MWD	POS	ID	TYPE	MWD	POS	ID	TYPE	MWD
D09	5918	SFE	2.894	F07	2915	SFE	3.337	G21	2903	SFE	3.518
D10	2977	SFE	3.318	F08	2946	SFE	3.337	G22	2935	SFE	3.356
D11	2974	SFE	3.318	F09	6924	SFE	2.894	G23	2930	SFE	3.356
D12	2905	SFE	3.342	F10	10812	SFE	1.854	G24	2951	SFE	3.213
D13	2943	SFE	2.880	F11	2958	SFE	3.038	G26	10699	SFE	1.475
D14	10147	FFCR	2.661	F12	5913	SFE	2.894	G27	2948	SFE	3.366
D15	2950	SFE	3.328	F13		3EL		G28	2913	SFE	3.356
D16	2929	SFE	3.328	F14		3EL		G29	2954	SFE	3.356
D17	2955	SFE	3.342	F15	2902	SFE	3.337	G30	10700	SFE	2.427
D18	2975	SFE	3.342	F16	10813	SFE	1.854	G32		SRC	
E01	5845	SFE	2.965	F17	2912	SFE	3.337	G33	2918	SFE	3.337
E02	6932	SFE	2.894	F18	6143	SFE	2.805	G34		PNT	
E03	2932	SFE	3.318	F19	5916	SFE	2.868	G35	10810	SFE	1.854
E04	5915	SFE	2.894	F20	2940	SFE	3.347	G36	10703	SFE	1.475