ANALYSIS OF THE NEUTRONIC BEHAVIOR

OF THE

NUCLEAR ENGINEERING TEACHING LABORATORY REACTOR

AT

THE UNIVERSITY OF TEXAS

Submitted By:

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1. Introduction

This report contains the results of investigation into the neutronic behavior of the Nuclear Engineering Teaching Laboratory reactor (NETL) at the University of Texas Austin. The objectives of this study were to: 1) create a model of the NETL to study the neutronic characteristics, and 2) demonstrate acceptable reactor performance and safety margins for the NETL core under normal conditions.

2. Summary and Conclusions of Principal Safety Considerations

The conclusion of this investigation is that the MCNP model does an acceptable job of predicting behavior of the NETL core. As such, the results suggest that the current NETL core can be safely operated within the parameters set forth in the technical specifications. Discussion and specifics of the analysis are located in the following sections. The final sections of this analysis provide suggestions for a limiting core configuration.

3. Reactor Fuel

The fuel utilized in the NETL is standard TRIGA® fuel manufactured by General Atomics. The use of low-enriched uranium/zirconium hydride fuels in TRIGA® reactors has been previously addressed in NUREG-1282 [1]. This document reviews the characteristics such as size, shape, material composition, dissociation pressure, hydrogen migration, hydrogen retention, density, thermal conductivity, volumetric specific heat, chemical reactivity, irradiation effects, prompt-temperature coefficient of reactivity and fission product retention. The conclusion of NUREG-1282 is that TRIGA® fuel, including the fuel utilized in the NETL, is acceptable for use in reactors designed for such fuel.

The design of standard stainless steel clad fuel utilized in the NETL is shown in Figure 1. Stainless steel clad elements used at NETL all have fuel alloy length of 38.1 cm. The characteristics of standard fuel elements are shown in Table 1.

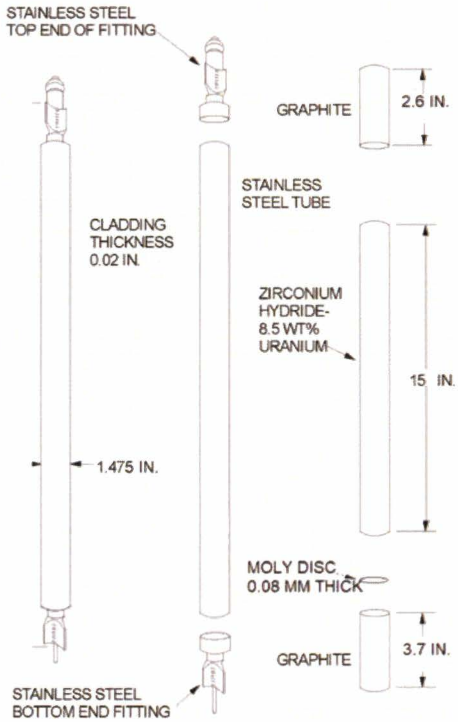

Figure 1 – TRIGA® Stainless Steel Clad Fuel Element Design used in the NETL Core

Table 1 – Characteristics of Stainless Steel Clad Fuel Elements

Uranium content [mass %]	8.5
BOL ^{235}U enrichment [mass % U]	19.75
Original uranium mass [gm]	37
Zirconium rod diameter [in]	0.25
Fuel meat outer diameter [in]	1.435
Cladding outer diameter [in]	1.475
Cladding material	Type 304 SS
Cladding thickness [in]	0.020
Fuel meat length [in]	15
Graphite slug outer diameter [in]	1.43
Upper graphite slug length [in]	2.6
Lower graphite slug length [in]	3.7
Molybdenum disc thickness [mm]	0.8

4. Reactor Core

The NETL core is a seven-ringed hexagonal grid array (labeled A through G) with 121 positions mostly composed of stainless-steel-clad standard TRIGA® fuel elements. The current core configuration contains 113 fuel elements (including three fuel-followed control rods, i.e. FFCRs). The core also contains an air-followed transient rod in C-1, a central thimble in A-1, several non-fueled locations that allow for a larger irradiation facility (in positions E-11, F-13 and F-14), a startup source in G-32, and a pneumatic transfer (Rabbit) irradiation facility in G-34, and an empty position G-26. The reactor is controlled by three electromagnetic control rods (Shim I, located in D-6; Shim II, located in D-14; and Regulating, located in C-7) and a pneumatic air-followed control rod (Transient, located in C-1), which utilize borated graphite (B_4C) as a neutron poison. Fuel temperature is measured by an instrumented fuel element (IFE) located in B-3. The current core configuration is shown in Figure 2.

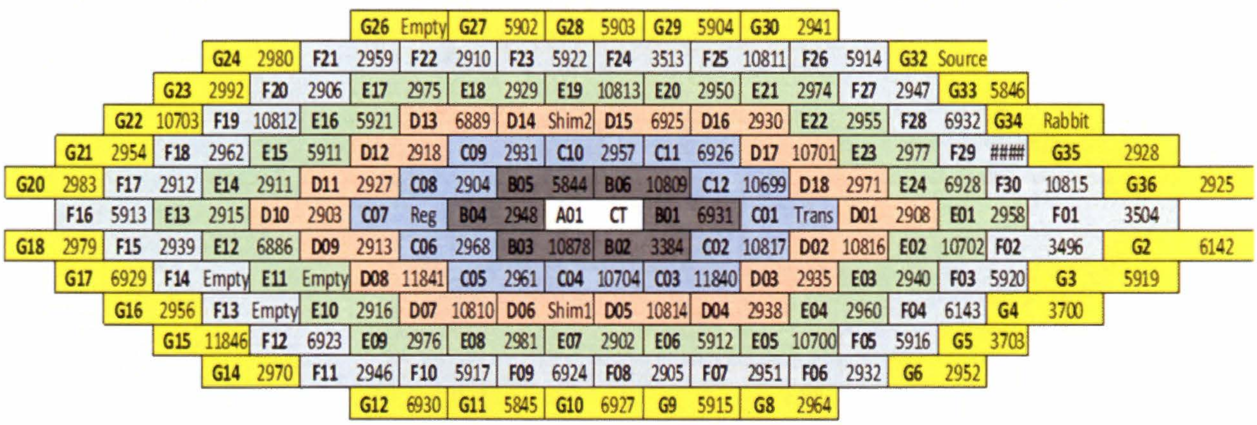

Figure 2 – Schematic Illustration of the NETL Showing the Current Core Configuration

Detailed neutronic analyses of the NETL core were undertaken using MCNP6.2 [2]. MCNP6.2 is a general purpose Monte Carlo transport code which permits detailed neutronic calculations of complex 3-dimensional systems. It is well suited to explicitly handle the material and geometric heterogeneities present in the NETL core. The original input deck for the NETL model was developed at UT Austin and modified by Oregon State University. Facility drawings provided by the manufacturer at the time of construction of the facility were used to define the geometry of the core and surrounding structures. The geometry of the stainless steel clad fuel elements and control rods were based upon the manufacturing drawings. Representative cross-sectional views of the MCNP model (of the initial core loading) are shown in Figure 3.

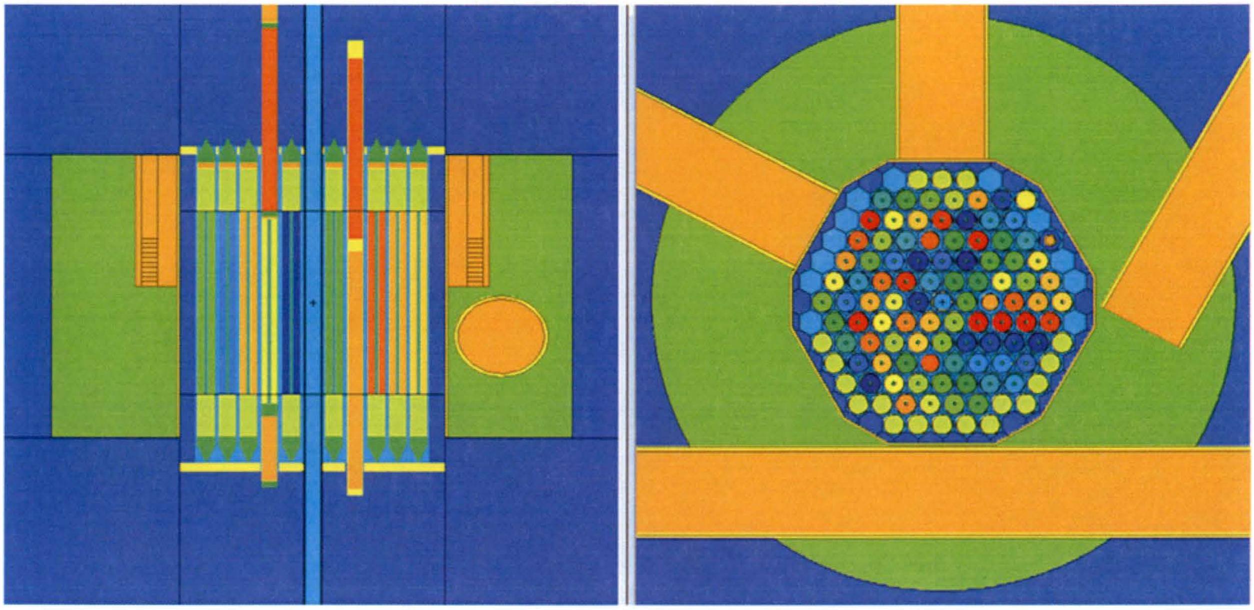

Figure 3 – Horizontal and Vertical Cross-sections of the NETL MCNP Model at BOL

The NETL reactor initially achieved criticality in March of 1992, however all of the fuel (except for the fresh FFCRs) was previously used at other facilities. Most of it came from a previous reactor on campus at Taylor Hall, but there were other sources as well. This made the beginning-of-life (BOL) fuel isotopic determination difficult. UT Austin performed a SCALE analysis to burn the fuel in conjunction with the given burnup records. The SCALE outputs were used to create BOL fuel isotopics for the MCNP runs.

5. Model Bias

Using critical rod height data from the first few months of NETL operation, a series of MCNP analyses based upon various critical rod heights were performed to determine the bias of the model. This bias represents such things as differences in material properties that are difficult to determine or unknown (i.e., exact composition of individual fuel meats and trace elements contained therein) or applicability of cross section data sets used to model the reactor (i.e., interpolation between temperatures). As a result, the validation of the model was based upon the ability of the code to accurately predict criticality as compared with measurements made on the reactor in early 1992.

A criticality calculation was performed using cold clean critical core configuration information from 3/23/1992. The k-effective of this configuration was 0.99393 ± 0.00013, or -$0.87 ± $0.04. Eighty different critical core configurations were then analyzed to determine how they bounded around the bias of this initial critical configuration. Figure 4 shows these 80 configurations with respect to the bias run. All of these kcode calculations utilized 500,000 neutrons per cycle for 200 total cycles (175 active cycles).

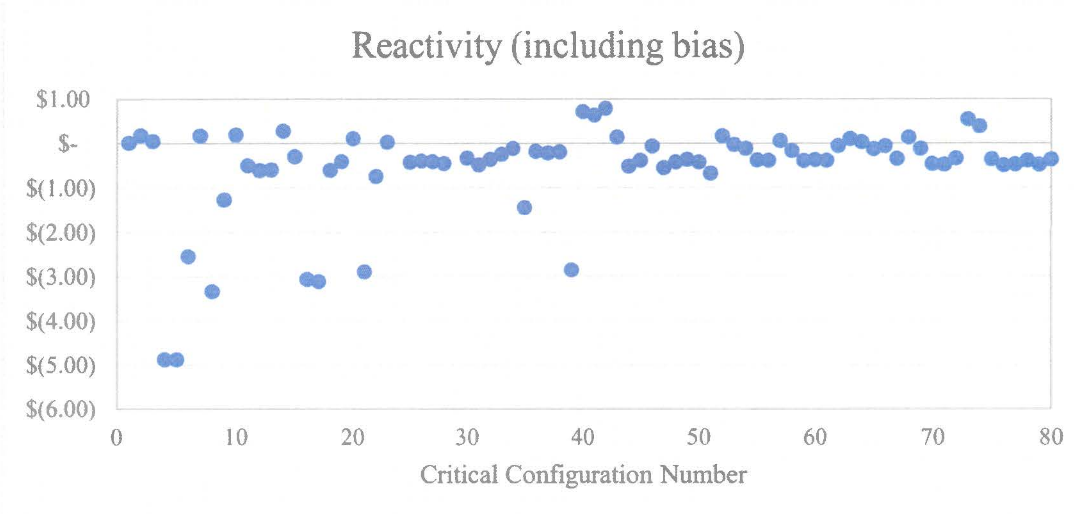

Figure 4 – Reactivity (including bias) of 80 Different BOL Critical Core Configurations

There appears to be significant deviation in the first 40 configurations. Note that most of these configurations are at low power but some are at high power. Most of the configurations with significant deviation are the high power runs, which would indicate that either the model is inaccurate or there is evidence of another problem. If the first 44 runs are ignored (if runs after 5/5/92 are observed), the data looks more accurate (see Figure 5), with an average of -$0.23.

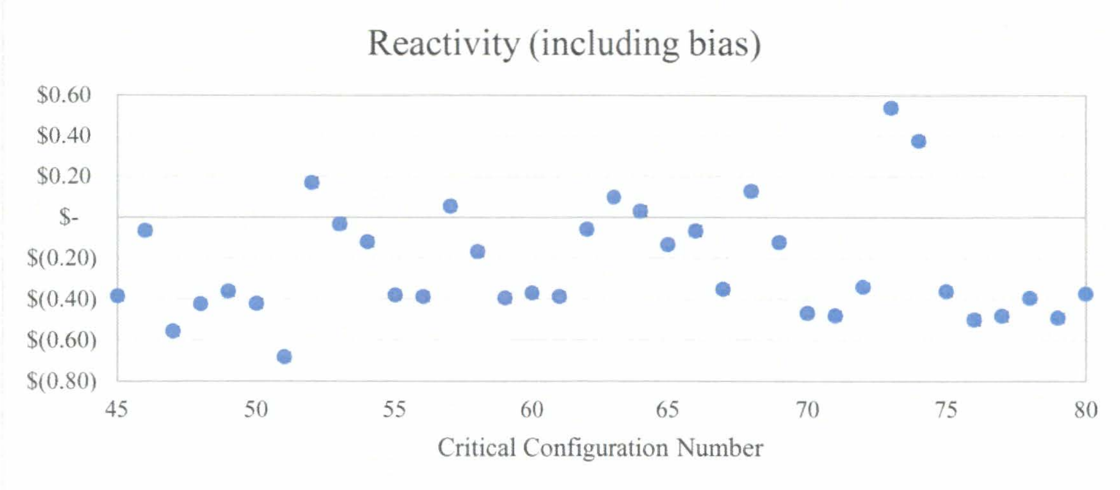

Figure 5 – Reactivity (including bias) of 36 Different BOL Critical Core Configurations

Note that these latter 36 configurations include some full power operations (cases #70-72, 76, 78 and 80). There is only one outlier over ±$0.60 (case #51), which would indicate that there were inconsistencies between high power operations during the first few months of operation. Other evidence, such as lower-than-expected fuel temperatures at these supposed high-power levels, would also indicate that something was inconsistent during the first few months of operation.

Thus the model bias that will be used for this study is -$1.10 (the -$0.23 bias plus -$0.87 bias). This bias represents such things as differences in material properties that are difficult to determine or unknown (i.e., lack of manufacturer mass spectroscopy data on the exact composition of individual fuel meats and trace elements contained therein) or applicability of cross section data sets used to model the reactor (i.e., interpolation between temperatures). A large source of error is the uncertainty of the contents of the BOL fuel meats, as all of the fuel (except for the FFCRs) was previously irradiated. Without knowing the exact burnup and previous grid location of these elements, it is nearly impossible to accurately determine their fuel compositions.

This bias will be used to determine reactivity values in the following sections.

6. Burnup Calculations

After performing the initial model bias calculations, a series of MCNP BURN calculations were performed to burn the NETL fuel to its current core configuration which was established in February 2018. This was a very detailed process as NETL is a very active facility and experienced many different core configurations. Using the fuel move logs, it was determined that there were 18 significant different core configurations that needed to be modeled (see Table 2). Each burnup step involved the fuel burnup for the specified amount of MW-days, parsing of the output fuel isotopics, then subsequent core model reconfiguration.

Table 2 – Summary of Burnup Steps

Burnup Step	From	To	MW-days	Total MW-days	FEs	Note
1	3/19/1992	10/12/1995	9.201	9.201	87	Initial Fuel Load
2	10/12/1995	1/20/1998	5.276	14.477	87	New IFE
3	1/20/1998	6/19/1998	2.789	17.266	87	Fuel Swapped Out/Add Rabbit
4	6/19/1998	3/4/1999	6.376	23.642	87	New IFE
5	3/4/1999	11/12/1999	7.671	31.313	90	Add 3 Fuel Elements
6	4/6/2000	6/29/2000	3.444	34.757	89	Core Reload
7	6/29/2000	1/29/2001	1.919	36.676	92	3L Experiment
8	1/29/2001	7/30/2001	9.138	45.814	92	3L Experiment with New IFE
9	7/30/2001	7/22/2002	21.508	67.322	95	Add 3 Fuel Elements
10	7/22/2002	11/13/2002	13.966	81.288	95	Fuel Shuffle
11	11/13/2002	4/1/2004	24.933	106.221	103	Add 8 New Fuel Elements
12	7/26/2004	7/13/2005	15.71	121.931	102	3L Experiment Core Reload
13	7/13/2005	7/11/2006	22.983	144.914	104	Add 2 Fuel Elements
14	7/11/2006	7/24/2007	41.732	186.646	104	Fuel Shuffle
15	7/24/2007	6/12/2008	18.347	204.993	108	Add 4 Fuel Elements
16	6/12/2008	6/24/2010	21.288	226.281	110	7L Experiment
17	6/24/2010	1/15/2016	73.587	299.868	114	Remove 7L Experiment
18	1/15/2016	2/22/2018	38.026	337.894	114	New IFE

7. Current Core Configuration

Once the burnup calculations were complete, the core was reconfigured to the current core configuration (as of 2/22/2018, see Figure 6). The next series of calculations were then performed to determine various neutronic characteristics of the NETL.

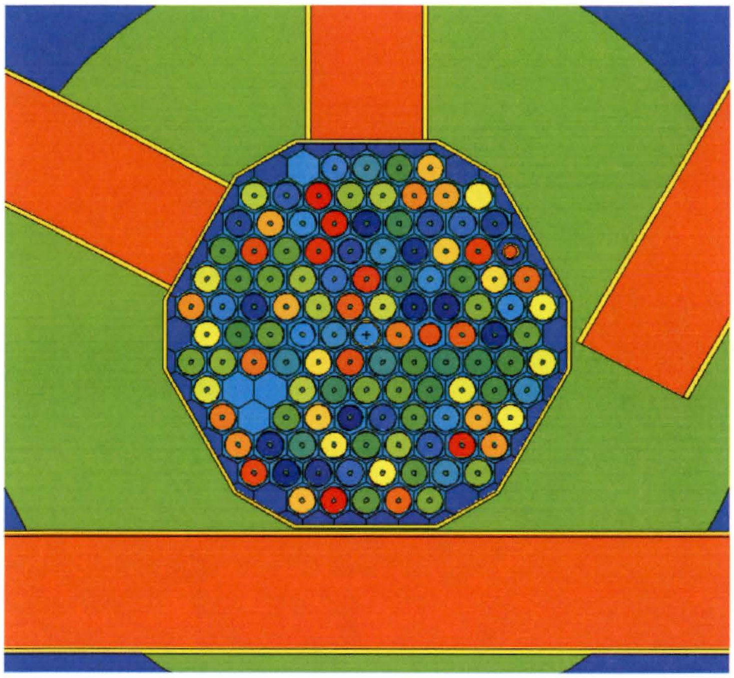

Figure 6 – Vertical Cross-section of Current Core Configuration MCNP Model

Core Power Distribution

F4 flux tallies were used to determine the power-per-element. The tallies output as a fluence per fission neutron. These units were converted to power density (W/cm^3) which were then converted to power-per-element. The individual power-per-element values (in kW) are shown in Figure 7.

G26	Empty	G27	5.74	G28	5.61	G29	5.54	G30	13.75														
G24	5.13	F21	6.80	F22	7.61	F23	7.89	F24	7.97	F25	7.34	F26	6.61	G32	Source								
G23	5.49	F20	7.34	E17	8.98	E18	10.10	E19	10.40	E20	10.15	E21	9.07	F27	7.42	G33	6.31						
G22	5.76	F19	7.77	E16	10.06	D13	11.66	D14	12.97	D15	12.55	D16	11.71	E22	10.28	F28	8.44	G34	Rabbit				
G21	5.34	F18	7.79	E15	10.16	D12	12.14	C09	13.49	C10	13.92	C11	13.71	D17	13.13	E23	10.74	F29	8.37	G35	6.38		
G20	5.02	F17	7.03	E14	9.61	D11	11.95	C08	13.70	B05	15.54	B06	15.39	C12	15.12	D18	13.20	E24	10.60	F30	7.79	G36	5.74
F16	6.10	E13	8.39	D10	10.94	C07	13.72	B04	15.4	A01	CT	B01	15.93	C01	Trans	D01	12.70	E01	9.62	F01	7.18		
G18	4.97	F15	7.23	E12	10.56	D09	12.26	C06	13.50	B03	15.82	B02	15.93	C02	14.14	D02	13.18	E02	11.38	F02	8.23	G2	5.91
G17	5.69	F14	Empty	E11	Empty	D08	12.16	C05	13.75	C04	14.91	C03	13.82	D03	13.01	E03	11.32	F03	8.93	G3	6.52		
G16	6.53	F13	Empty	E10	10.95	D07	11.27	D06	13.22	D05	12.99	D04	12.39	E04	11.05	F04	9.07	G4	6.84				
G15	5.59	F12	7.44	E09	8.71	E08	10.01	E07	10.95	E06	10.90	E05	10.39	F05	8.39	G5	6.57						
G14	4.96	F11	6.36	F10	7.47	F09	8.34	F08	8.53	F07	8.06	F06	7.35	G6	5.95								
G12	5.20	G11	5.79	G10	6.14	G9	6.02	G8	5.71														

Figure 7 – Current Core Power-Per-Element (in kW) Distribution at 1.1 MW

The red highlighting indicates the hottest fuel element locations, which are in B-1 and B-2, with a maximum power of 15.93 kW (at a total maximum core power of 1.1 MW). B-2 is actually slightly higher than B-1 (15.931 kW vs. 15.929 kW) but both are within the 2-sigma error of 0.04 kW.

Effective Delayed Neutron Fraction and Prompt Neutron Generation Time

MCNP outputs effective delayed neutron fraction (β_{eff}) and prompt neutron lifetime when using the KOPTS card. Nine different MCNP calculations (the same calculations used in the following Core Excess section) were used to determine β_{eff} and prompt neutron lifetime (see Table 3).

Table 3 – β_{eff} and Prompt Neutron Lifetimes for Current Core Configuration

Case	Prompt Neutron Generation Time (s)	Error (s)	β_{eff}
Trans fully in	47.62	7.543	0.00705
Trans fully out	46.868	7.111	0.00716
Reg fully in	48.08	7.824	0.00707
Reg fully out	46.718	6.961	0.00707
Shim I fully in	48.023	7.748	0.00702
Shim I fully out	46.777	6.974	0.00705
Shim II fully in	48.104	7.684	0.00717
Shim II fully out	46.708	7.086	0.00713
All Rods Out	45.824	6.626	0.00720
Average	47.191	7.284	0.00710

The average effective delayed neutron fraction β_{eff} was calculated to be 0.00710 ± 0.00007. This is in reasonable agreement with values predicted in other LEU TRIGA® cores (i.e., Oregon State University $\beta_{eff} = 0.0076$ [3], University of Maryland $\beta_{eff} = 0.007$ [4]) and also the value historically used for the NETL of $\beta_{eff} = 0.007$. The value $\beta_{eff} = 0.007$ will be used to express all dollar values of reactivities in this report.

The average prompt neutron generation time is 47.191 ± 7.284 seconds.

Core Excess, Control Rod Worth and Shutdown Margin

Nine different MCNP calculations were performed to determine core excess, control rod worth, and shutdown margin. Core excess is calculated as the reactivity of all rods withdrawn from the core. Control rod worths and shutdown margin were calculated by determining a critical state of the reactor with one rod full inserted and the other three rods banked at the same height, then fully withdrawing the previously-inserted rod. The resulting values (with comparison to values measured at NETL) are shown in Table 3.

Table 4 – Current Core Rod Worth Calculations

Case	MCNP k-effective Rod Full-In	MCNP k-effective Rod Full-Out	MCNP Rod Worth	Experimental Reactivity	Difference
Transient	1.00035	1.02354	\$3.24	\$3.44	-\$0.20
Regulating	0.99978	1.02214	\$3.13	\$3.18	-\$0.05
Shim 1	1.00078	1.02248	\$3.03	\$3.09	-\$0.06
Shim 2	1.00014	1.0211	\$2.93	\$2.94	-\$0.01
All Rods Out (Core Excess)	-	1.04118	\$6.75	\$6.06	\$0.69

MCNP appears to accurately calculate the individual rod worths. The Regulating, Shim 1 and Shim 2 rods are all within the margin of error (which is approximately ±\$0.06 for each case).

These calculations show a core excess of \$6.75 ± \$0.03. This is below the technical specification limit of \$7.00. The core excess was measured by NETL to be \$6.06 on 3/6/18. MCNP appears to have over-estimated core excess by approximately \$0.70. This could be due to a variety of reasons, such as only modeling the fuel elements as one single material per element, thus some burnup resolution is lost as the fuel does not burn uniformly throughout.

The technical specification definition of shutdown margin is "the minimum reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition (the highest worth MOVEABLE EXPERIMENT in its most positive reactive state, each SECURED EXPERIMENT in its most reactive state), with the most reactive rod in its most reactive position, and that the reactor will remain subcritical without further operator action." The most reactive rod is the Transient rod.

Total rod worth minus the Transient rod is \$9.09 ± \$0.06. NRC shutdown margin is this value minus the core excess, which would be \$2.34 ± \$0.06, which is far above the technical specification limit of \$0.29.

Prompt Fuel Temperature Coefficient

The prompt-temperature coefficient associated with the NETL fuel, α_F, was calculated by varying the fuel meat temperature while leaving other core parameters fixed. The MCNP model was used to simulate the reactor with all rods out at 293, 600, 900, 1200 and 2500 K. The prompt-temperature coefficient for the fuel was calculated at the mid-point of the four temperature intervals. The results are shown in Figure 8 and tabulated in Table 5. Results from GA were added to show similarity [5]. The prompt-temperature coefficient is observed to be negative for all evaluated temperature ranges with decreasing magnitude as temperature increases. The coefficient has a value of -1.3¢/°C at 446.8 K, which is similar to the value of -0.01%/°C stated in the original SAR [6].

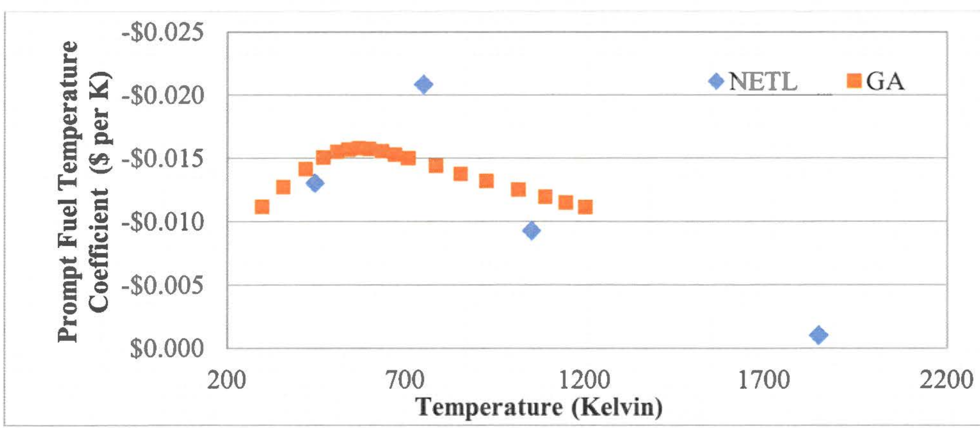

Figure 8 – Current Core Configuration Prompt Temperature Coefficient, α_F, as a Function of Temperature

Table 5 – Current Core Configuration Prompt Temperature Coefficient

Fuel Temperature [K]	Prompt Temperature Coefficient [$/°C]
446.8	-$0.0130
750	-$0.0208
1050	-$0.0092
1850	-$0.0010

Moderator Void Coefficient

The moderator void coefficient of reactivity was also determined using the MCNP model. The voiding of the core was introduced by uniformly reducing the density of the liquid moderator in the entire core. The calculation was performed from 0% to 100% voiding at 10% intervals. The void coefficient was negative for every interval and steadily decreased, as can be seen in Figure 9.

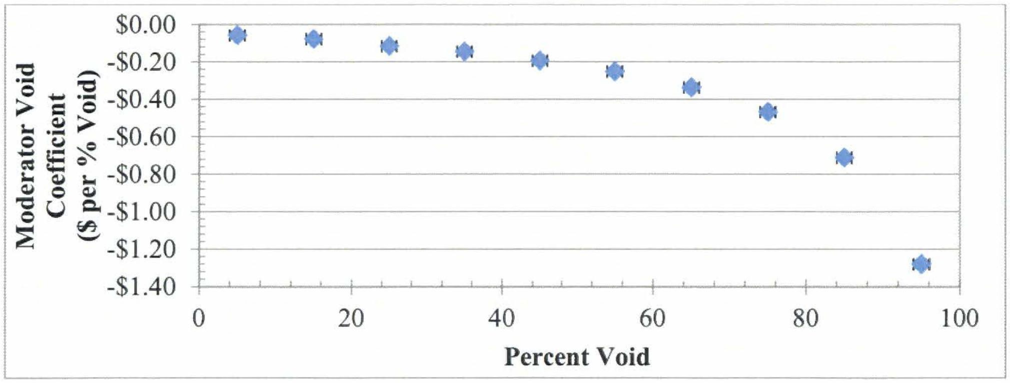

Figure 9 – Current Core Configuration Moderator Void Coefficient

Moderator Temperature Coefficient

The moderator temperature coefficient of reactivity, α_M, was determined by varying the moderator density with respect to temperature within the MCNP model from the expected operating temperature range of 20°C to 50°C (using Engineering Toolbox [7] to determine water density). The results are shown in Figure 10. The moderator temperature coefficient is calculated to be slightly positive from 25˚C to 30 ˚C and from 45 ˚C to 50 ˚C, but these changes are less than $0.01/˚C and both points (with 2-sigma error) are bounded around zero. The moderator temperature coefficient appears to be negligible.

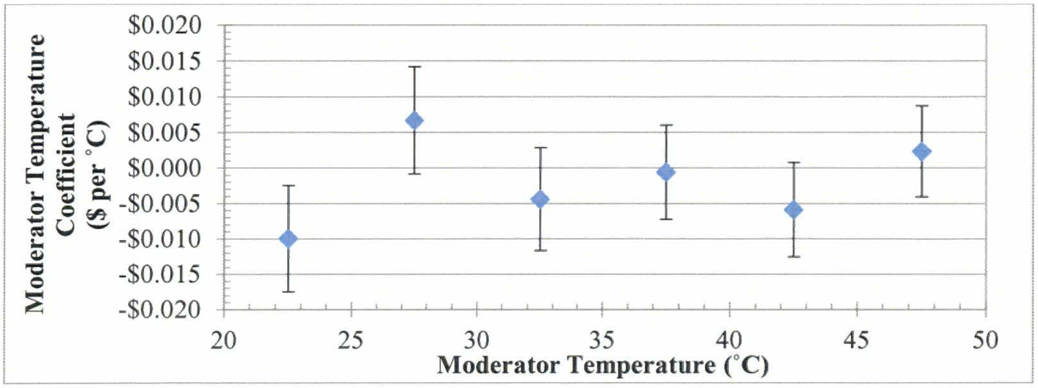

Figure 10 – Current Core Configuration Moderator Temperature Coefficient

Power Coefficient of Reactivity

The power coefficient of reactivity, otherwise known as power defect, is the amount of reactivity required to overcome the temperature feedback during the rise to power. This is modeled by analyzing two MCNP decks that are similar except for the neutron cross-sections used. Two k-effective calculations were performed with all rods out, one using cross sections at 293K (low power) and one using cross sections at 600K (full power). The results are seen in Table 6.

Table 6 – K-Effective Calculations Used to Determine Current Core Power Defect

Case	MCNP k-effective	Standard Deviation	Reactivity	Error (2-sigma)
Low Power	1.04118	0.00012	$6.75	$0.03
Full Power	1.01327	0.00010	$2.94	$0.03

Power defect is simply the difference in reactivity between these two cases; thus the power defect is \$3.81 ± \$0.05.

8. Limiting Core Configuration

This section will suggest a limiting core configuration that utilizes fresh fuel to improve reactor efficiency while maintaining proper safety margins. The NETL limiting core configuration is a core that completely consists of fresh fuel.

Figure 11 shows the suggested limiting core configuration. For this analysis, it is suggested that the core is loaded with 84 fresh fuel elements (including FFCRs), which will provide just under the license limit of \$7.00 core excess (\$6.93 ± \$0.07). This is comparable to the original 1992 BOL core configuration, which was measured to have a \$6.38 core excess on a core of 87 lightly-irradiated fuel elements. This configuration will provide maximum flux to the beam port facilities while maintaining safety margins.

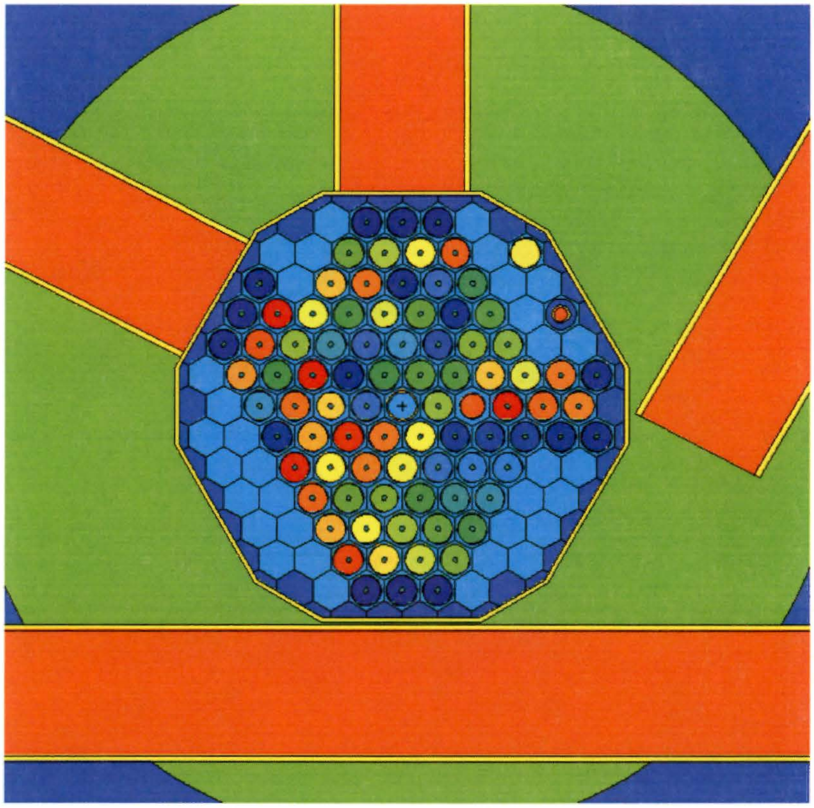

Figure 11 – Vertical Cross-section of Limiting Core Configuration MCNP Model

Core Power Distribution

Figure 12 shows the power-per-element (in kW) in the suggested limiting core configuration.

G26	Empty	**G27**	6.21	**G28**	5.93	**G29**	5.90	**G30**	18.01														
G24	Empty	**F21**	Empty	**F22**	10.37	**F23**	10.17	**F24**	9.88	**F25**	9.49	**F26**	Empty	**G32**	Source								
G23	6.74	**F20**	Empty	**E17**	13.18	**E18**	13.64	**E19**	14.05	**E20**	12.77	**E21**	11.52	**F27**	Empty	**G33**	Empty						
G22	5.76	**F19**	10.65	**E16**	14.12	**D13**	16.05	**D14**	14.99	**D15**	16.87	**D16**	14.61	**E22**	12.74	**F28**	Empty	**G34**	Rabbit				
G21	5.76	**F18**	9.70	**E15**	13.76	**D12**	17.09	**C09**	19.22	**C10**	19.62	**C11**	18.13	**D17**	15.33	**E23**	13.08	**F29**	Empty	**G35**	Empty		
G20	Empty	**F17**	9.36	**E14**	12.68	**D11**	16.75	**C08**	19.75	**B05**	22.14	**B06**	21.59	**C12**	18.75	**D18**	15.70	**E24**	11.85	**F30**	8.55	**G36**	4.72
F16	Empty	**E13**	11.45	**D10**	14.75	**C07**	15.98	**B04**	22	**A01**	CT	**B01**	21.48	**C01**	Trans	**D01**	14.41	**E01**	9.72	**F01**	6.43		
G18	Empty	**F15**	Empty	**E12**	12.69	**D09**	15.69	**C06**	18.87	**B03**	21.58	**B02**	21.22	**C02**	18.52	**D02**	15.56	**E02**	11.81	**F02**	8.53	**G2**	4.67
G17	Empty	**F14**	Empty	**E11**	Empty	**D08**	15.35	**C05**	18.01	**C04**	18.79	**C03**	17.52	**D03**	14.90	**E03**	12.93	**F03**	Empty	**G3**	Empty		
G16	Empty	**F13**	Empty	**E10**	12.46	**D07**	14.34	**D06**	13.87	**D05**	15.90	**D04**	14.04	**E04**	12.43	**F04**	Empty	**G4**	Empty				
G15	Empty	**F12**	Empty	**E09**	11.06	**E08**	12.17	**E07**	13.02	**E06**	12.02	**E05**	11.02	**F05**	Empty	**G5**	Empty						
G14	Empty	**F11**	Empty	**F10**	8.90	**F09**	9.20	**F08**	9.20	**F07**	8.90	**F06**	Empty	**G6**	Empty								
G12	Empty	**G11**	5.45	**G10**	5.45	**G9**	5.46	**G8**	Empty														

Figure 12 – Limiting Core Configuration Power-Per-Element Distribution at 1.1 MW

The hottest fuel element in now in location B-5. This makes sense as the core is more shifted to the northwest, which would better centralize the location of the maximum power production around B-5. Also, the hottest power-per-element at 1.1 MW is now 22.14 ± 0.06 kW, which is higher than the current core hot channel, due to a lower fuel loading concentrating more power at the center of the core.

Effective Delayed Neutron Fraction and Prompt Neutron Generation Time

Once again using the "KOPTS" card and running nine cases, the effective delayed neutron fraction β_{eff} and prompt neutron generation times were calculated

Table 7 – β_{eff} and Prompt Neutron Lifetimes for Limiting Core Configuration

Case	Prompt Neutron Generation Time (s)	Error (s)	β_{eff}
Trans fully in	42.828	5.531	0.00743
Trans fully out	42.721	5.024	0.00725
Reg fully in	43.764	5.502	0.00732
Reg fully out	41.951	4.985	0.00742
Shim I fully in	43.546	5.616	0.00737
Shim I fully out	42.407	5.104	0.00737
Shim II fully in	43.614	5.458	0.00733
Shim II fully out	42.261	5.200	0.00728
All Rods Out	42.024	4.965	0.00742
Average	42.791	5.265	0.00735

The average β_{eff} was calculated to be 0.00735 ± 0.00007. There is a slight increase in β_{eff} compared to the current core configuration, but for consistency, 0.007 will continue to be used to express all dollar values of reactivities in this report.

The average prompt neutron generation time is 42.791 ± 5.265 seconds.

Core Excess, Control Rod Worth, and Shutdown Margin

The same nine MCNP rod worth calculations were performed again for the limiting core configuration: Core excess, shutdown margin, and individual rod worths were calculated from these outputs and the reactivity values (with the bias taken into account) of each of these calculations are shown in Table 7.

Table 8 – Limiting Core Configuration Rod Worth Calculations

Case	MCNP k-effective Rod Full-In	MCNP k-effective Rod Full-Out	MCNP Rod Worth
Transient	0.99886	1.02191	$3.22
Regulating	1.00024	1.03222	$4.43
Shim 1	1.00003	1.02431	$3.39
Shim 2	1.0003	1.02857	$3.93
All Rods Out (Core Excess)	-	1.04257	$6.93

These calculations show a core excess of $6.93 ± $0.07. This is below the technical specification limit of $7.00.

Now the most reactive rod is the Regulating, due to having more fuel near its vicinity and the power shifted to the northwest side of the core. Total rod worth minus the Regulating Rod is $10.53 ± $0.16. NRC shutdown margin is this value minus the core excess, which would be $3.60 ± $0.16, which is still far above the technical specification limit of $0.29.

Prompt Fuel Temperature Coefficient

The results of the limiting core configuration prompt fuel temperature coefficient calculations are shown in Figure 13 and tabulated in Table 9.

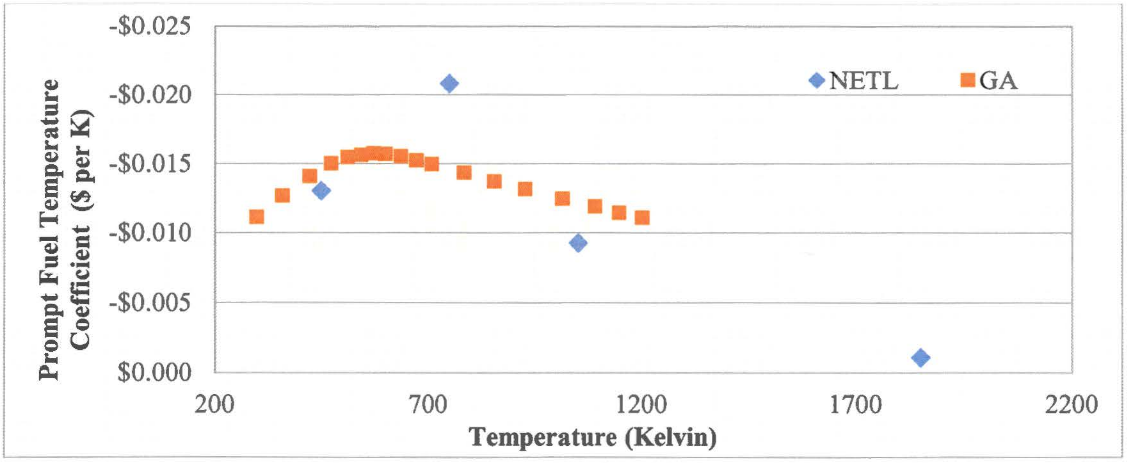

Figure 13 – Limiting Core Configuration Prompt Temperature Coefficient, α_F, as a Function of Temperature

Table 9 – Limiting Core Configuration Prompt Temperature Coefficient

Fuel Temperature [K]	Prompt Temperature Coefficient [$/°C]
446.8	-$0.01302
750	-$0.02081
1050	-$0.00928
1850	-$0.00105

These values are similar to the original BOL coefficients.

Moderator Void Coefficient

Figure 14 shows the moderator void coefficient in the suggested limiting core configuration.

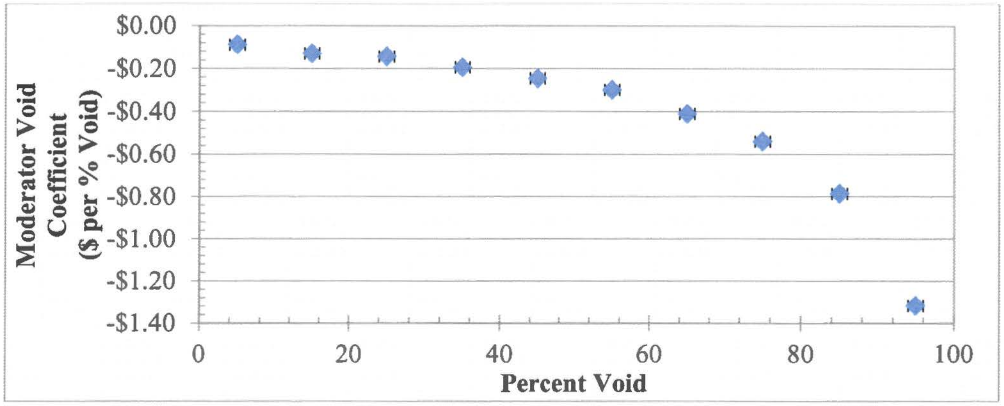

Figure 14 – Limiting Core Configuration Moderator Void Coefficient

The void coefficient was negative for every interval and steadily decreased, similar to the current core configuration. The void coefficient is slightly more negative in the limiting core configuration, likely due to having more moderator in the core configuration.

Moderator Temperature Coefficient

Figure 15 shows the moderator temperature coefficient in the suggested limiting core configuration.

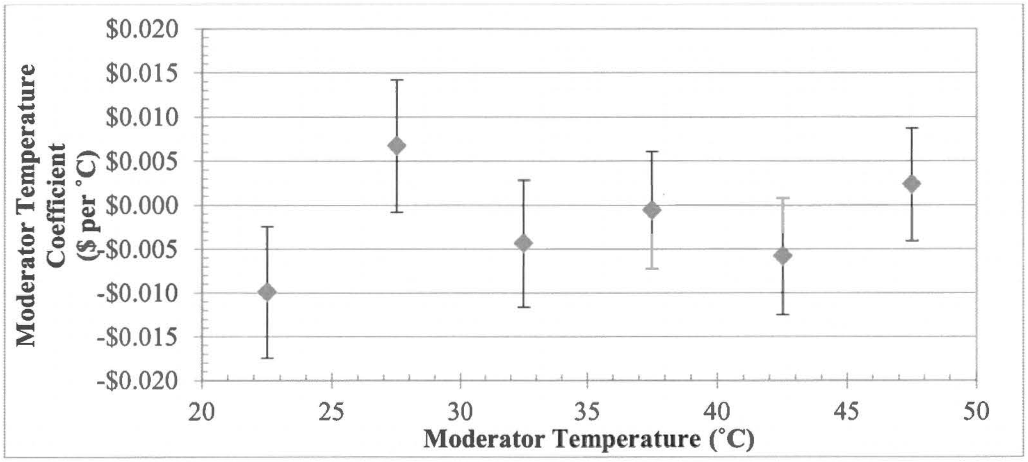

Figure 15 – Limiting Core Configuration Moderator Temperature Coefficient

Once again the moderator temperature coefficient appears to be negligible as it bounds around $0.00 at all observed temperature ranges.

Power Coefficient of Reactivity

The power coefficient of reactivity results are seen in Table 10.

Table 10 – K-Effective Calculations Used to Determine Limiting Core Power Defect

Case	MCNP k-effective	Standard Deviation	Reactivity	Error (2-sigma)
Low Power	1.04231	0.00015	$6.90	$0.04
Full Power	1.01921	0.00010	$3.79	$0.03

Thus the power defect is \$3.11 ± \$0.05. This is lower than the current core configuration's power defect, likely due to less resistance at the point-of-adding-heat due to the lower amount of zirconium-hydride in the core.

Hot Channel Power Summary

The hot channel in the limiting core configuration was determined to be B-5. An fmesh calculation was performed to analyze a 20 by 20 mesh array to determine axial and radial power distributions. Table 11 summarizes the results of this calculation.

Table 11 – Limiting Core Hot Channel Power Summary

Core Configuration	Hot Rod Location	Hot Rod Thermal Power [kW]	Hot Rod Peak Factor [P_{max}/P_{avg}]	Hot Rod Axial Peak Factor [P_{max}/P_{avg}]	Hot Rod Radial Peak Factor [P_{max}/P_{avg}]	Effective Peak Factor
Limiting Core	B6	22.14	1.691	1.296	1.017	2.229

9. Summary

MCNP6.2 was used to calculate fundamental and operational parameters for the Nuclear Engineering Teaching Laboratory Reactor to demonstrate the reactor's adherence to safety margins in the technical specifications. Values of fundamental parameters agree well with theoretical values. Values of operational parameters agree well with measured values, giving confidence in the model's ability to predict the viability of future core configurations. The results of this study indicate that the NETL can be operated safely within the Technical Specification bounding envelope and that its MCNP model can be used to predict future core configuration changes.

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March 31, 2020

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Washington, D.C. 20555-0001

Geoffrey Wertz, P.E.
Non-Power Production and Utilization Facility Licensing Branch
Division of Advance Reactors and Non-Power Utilization
Nuclear Reactor Regulation

SUBJECT: Docket No. 50-602, Facility Operating License R-129 - Submission of Neutronic and Thermal Hydraulic Analysis for the University of Texas at Austin Research Reactor

REFERENCE: October 18, 2018 letter: University- of Texas at Austin - Summary of Site Visit and Request for Schedule for Completion of the Reactor Analyses RE: Renewal of Facility Operating License No. R-129 for The University of Texas at Austin Research Reactor (EPID NO. L-2017-RNW -0032)

Sir:

We respectfully submit neutronics and thermal-hydraulic analysis, attached. If you have any questions, please contact me at 512-232-5373 or whaley@mail.utexas.edu.

P. M. Whaley

I declare under penalty of perjury that the foregoing is true and correct.

W. S. Charlton

ATT:
(1) Analysis of the Neutronic Behavior of the Nuclear Engineering Teaching Laboratory at the University of Texas
(2) Thermal Hydraulic Analysis of the University of Texas (UT) TRIGA Reactor