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Reference: University of Maryland, Maryland University Training Reactor ("MUTR"), Docket No. 50-166, License No. R-70 Request for Additional Information ("RAI")

Enclosed is a hard copy of General Atomics' final report and response, titled "University of Maryland Reactor Analysis and Support," to Request #2 to the NRC's April 6, 2010 Request for Additional Information. A soft copy was e-mailed to NRC officials on February 1, 2011. With this submission, all outstanding requests for the renewal of the MUTR license have been answered.

If there are questions about the information submitted, please write to me at: Department of Materials Science and Engineering, University of Maryland, College Park, MD 20742-2115 or email me at mohamad@umd.edu. Please copy Prof. Robert Briber on any such correspondence: Department of Materials Science and Engineering, University of Maryland, College Park, MD 20742-2115; rbriber@umd.edu.

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

Sincerely,

Mohamad Al-Sheikhly
Professor and Director
Maryland University Training Reactor

cc: Robert Briber
Enclosure

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NRR

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UNIVERSITY OF MARYLAND REACTOR ANALYSIS AND SUPPORT

Response for MUTR's Request for Additional Information

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GENERAL ATOMICS

GENERAL ATOMICS

GA 1485 (REV. 08/06E)

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CONTINUE ON GA FORM 1485-1

NEXT INDENTURED DOCUMENT(S)

COMPUTER PROGRAM

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LIST OF ABBREVIATIONS/ACRONYMS

BOL	Beginning-of-Life
GA	General Atomics
IFE	Instrumented Fuel Element
MHR	Modular Helium Reactor
MUTR	University of Maryland TRIGA Reactor
RAI	Request for Additional Information
SAR	Safety Analysis Report
TRIGA	Training Research Isotope General Atomics
TS	Technical Specifications

1 PURPOSE

The purpose of this report is to document the nuclear and thermal hydraulic analyses that were performed by General Atomics (GA) in supporting the activities necessary to answer a Request for Additional Information (RAI) question regarding the University of Maryland TRIGA Reactor (MUTR) Safety Analysis Report (SAR) submittal. The nuclear and thermal hydraulic analysis was performed to directly address question #2a and question #2b:

#2a

Section 4.6 of the 2000 SAR states that the maximum fuel temperature would approach 400 degrees Celsius (°C) when the pool water would approach the boiling point of water, 100 °C. Please provide a reference analysis or data supporting the maximum fuel temperature value.

#2b

"Section 4.5.3 of the 2000 SAR states that the safety limit for the MUTR is for a maximum fuel temperature of 1000 °C. To preclude reaching this point the limiting safety system setting for the MUTR has been defined at less than 175 °C as measure by the instrumented fuel element (IFE). Proposed Technical Specifications (TS), dated December 18, 2006, section 2.2 also sets the limiting safety system setting at 175 °C as measured by the IFE in the periphery of the core and allows locating the IFE at any location in the core. Please provide a reference analysis or data supporting the bounding power peaking analysis that the temeperature at the hottest fuel element would be no grater that 350 °C, when the IFE is at 175 °C at any allowable location."

2 METHODS

2.1 DIF3D CODE

Work was performed using GA engineering procedures and verified computer codes (Ref. 2). Power peaking analyses were performed using the DIF3D multi-dimensional diffusion theory code which solves the neutron diffusion equations with arbitrary group scattering. The 3-dimensional MUTR DIF3D model is large with 151 x-axis, 182 y-axis and 125 z-axis mesh points. Requested convergences on eigenvalue and fission source were 1E-5 and 1E-4, respectively. The analyses used the cross-sections generated for beginning-of-life (BOL) concentrations at the approximate average fuel temperature of 200°C, the closest nuclear data available.

2.2 RELAP5 /MOD3.3-Patch03 CODE

RELAP has been an industry standard code for the analysis of power reactors. Its development and improvements have occurred over at least 25 years. The code performs steady state and transient reactor neutronics, thermal hydraulics and fuel rod thermal analysis. It allows for very general modeling – multiple rod and channel configurations and heat structures for rod thermal performance. The code has sophisticated single and two phase flow modeling both in its continuity, momentum and

energy formulations, as well as for wall and interfacial friction and heat transfer correlations. The version RELAP5/MOD3.3-Patch03 is the latest available validated version of the code (Ref. 3).

3 RESULTS

3.1 Request for Additional Information – Question 2a

Section 4.6 of the 2000 SAR states that the maximum fuel temperature would approach 400 °C when the pool water would approach the boiling point of water, 100 °C. Please provide a reference analysis or data supporting the maximum fuel temperature value.

This case was analyzed using the latest version of the RELAP Code (RELAP5/MOD3.3-Patch03) as shown in Table 3-1. The assumptions in deriving the results shown in Table 3-1 are:

- RELAP5 code model
- Model includes one average powered rod representing 93 rods and one hot rod.
- 120 gpm water enters near bottom of core
- 14.7 psia pressure above tank water
- RELAP model incorporates IFE fuel temperatures supplied by MUTR
- RELAP model similar to models used in previous SAR analyses

Table 3-1: RELAP Summary Results as Pool Temperature Approaches 100 ° C

Case	Reactor Power (kW)	Hot Rod Peaking Factor*	Max Hot Rod Fuel Temp (°C)	Max Allowable Pool Temp Achieved** (°C)	Fig. No.	Comment
1	300	1.6	251.77	105	3-1	300 kW is the max design power. Fuel-clad gap conductance adjusted per MUTR IFE measurement.

*Peaking factors to be updated once nuclear calculations are completed

** RELAP was run with a 105 ° C pool temperature with the hot rod maximum temperatures below the limit of 400 ° C. RELAP oscillatory conditions at higher pool temperatures prevented higher temperature calculations

3.2 Result - Question #2a

The analyses performed have shown that for a case where the pool temperature approaches the boiling point of water the fuel temperature will not exceed 252 ° C, regardless of whether there is primary coolant flow, as shown in Figure 3-1, and is well within the 400 ° C stated in the 2000 SAR.

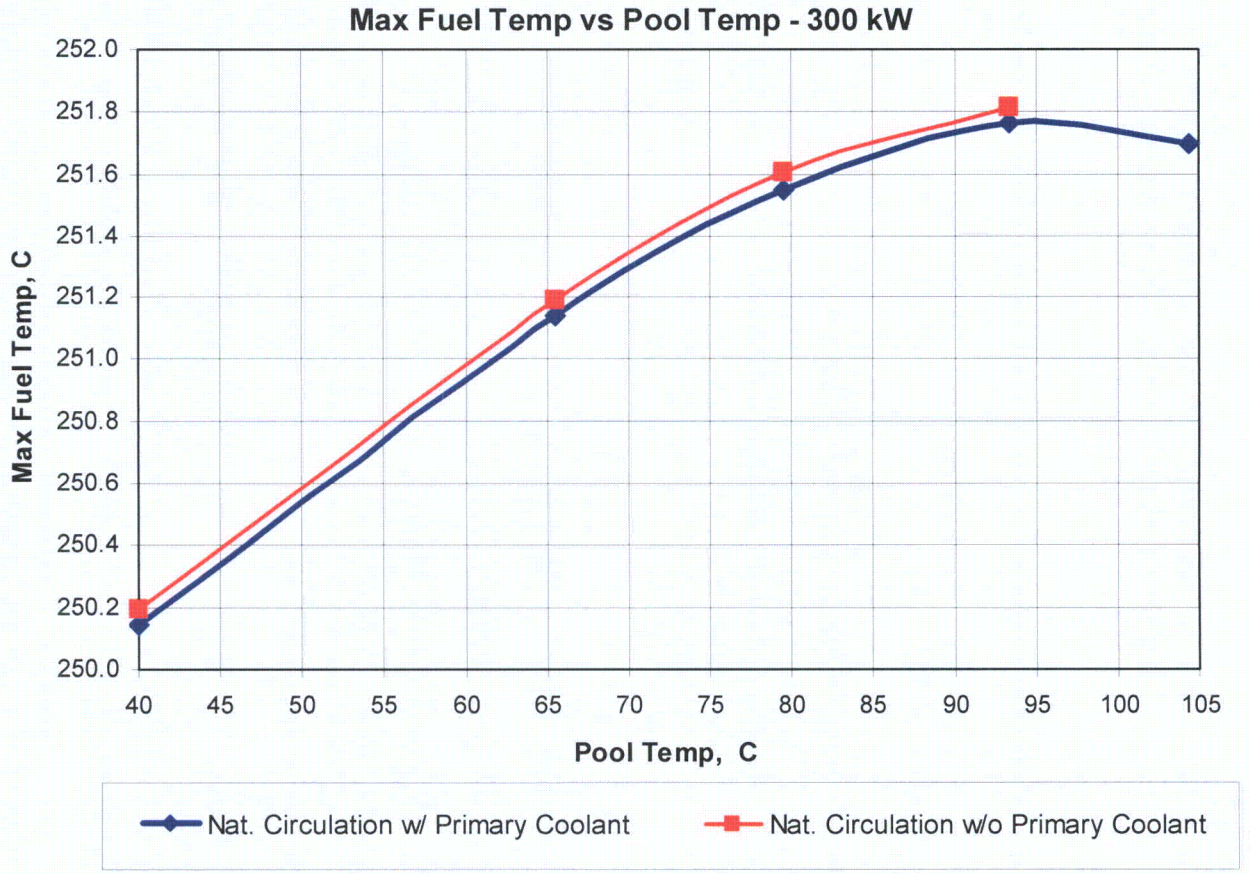

Figure 3-1: Hot Rod Maximum Fuel Temperature as Pool Water Approaches 100 °C Modified Conditions with Reactor Power at 300 kW

3.3 Request for Additional Information – Question #2b

"Section 4.5.3 of the 2000 SAR states that the safety limit for the MUTR is for a maximum fuel temperature of 1000 °C. To preclude reaching this point the limiting safety system setting for the MUTR has been defined at less than 175 °C as measure by the instrumented fuel element (IFE). Proposed Technical Specifications (TS), dated December 18, 2006, section 2.2 also sets the limiting safety system setting at 175 °C as measure by the IFE in the periphery of the core and allows locating the IFE at any location in the core. Please provide a reference analysis or data supporting the bounding power peaking analysis that the temeperature at the hottest fuel element would be no grater that 350 °C, when the IFE is at 175 °C at any allowable location."

This condition was also analyzed using the latest version of the RELAP computer code (RELAP5/MOD3.3-Patch03). Three fuel rods were considered. A low powered rod for the IFE, an average rod to represent the core average conditions, and a high powered rod for the hot rod. To run

the RELAP code, power factors are required to obtain the temperatures for the three rods considered. Since a power peaking map was not available a nuclear calculation using the DIF3D code was performed to obtain the peaking factors. This calculation is summarized in the Appendix A.

Using the above RELAP model and the peaking factors from the DIF3D analysis, calculations were made for fuel temperatures as a function of reactor power over the range of 250 kW to 1,100 kW. Pool temperatures of 27 ° C and 40 ° C and fuel-clad gap conductances adjusted per the MUTR IFE measurements, Figure 3-2, were included in the RELAP model.

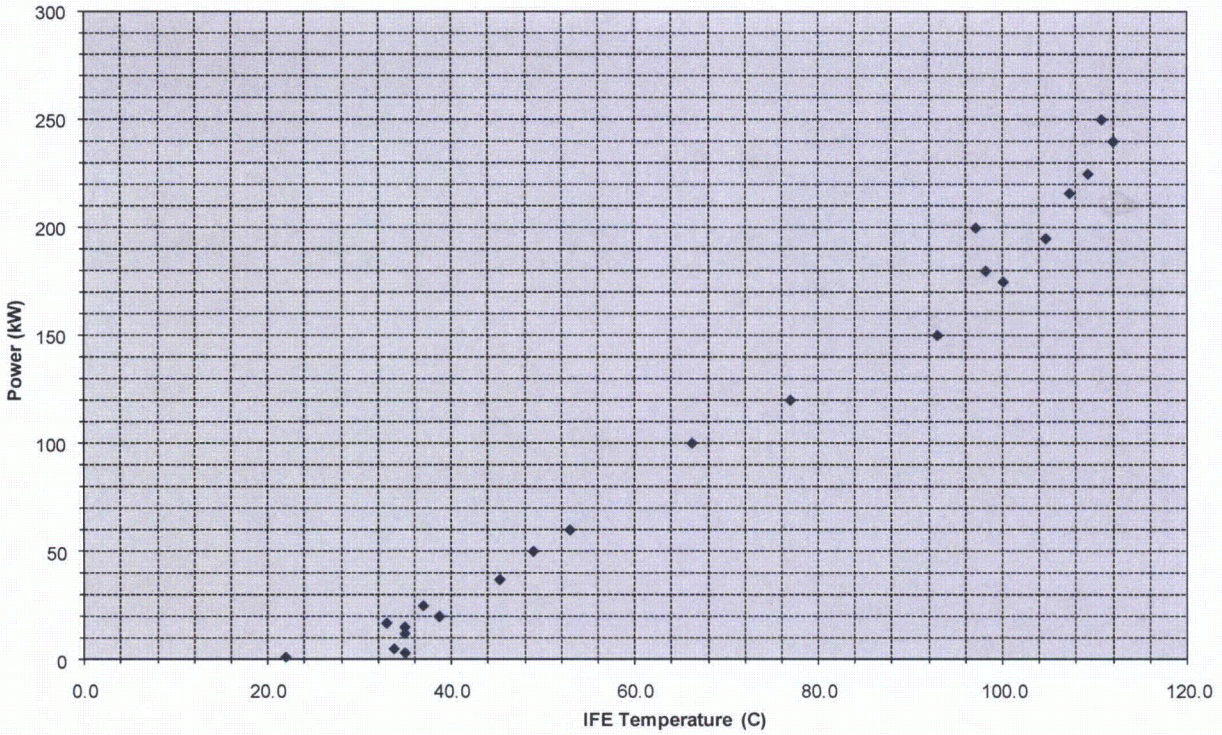

Figure 3-2: IFE Temperatures as a Function of Reactor Power

Fuel-clad gap adjustments were made by performing a pre-analysis RELAP run at a reactor power of 250 kW, a pool temperature of 27 ° C and an IFE peaking factor of 0.841, shown in Figure A-3 in Appendix A. These conditions resulted in a predicted IFE temperature of 135 ° C, somewhat higher than the MUTR measured value of 113° C. Therefore a further analysis was run to predict the power factor that would result in the measured 113 ° C temperature. This yielded a power factor of 0.67 for the IFE. Thus a combination of uncertainties in the DIF3D calculations and in the IFE measurements gave rise to this temperature difference. Further calculations were run using the 0.67 rod peaking factor for the IFE.

The RELAP results for the IFE are shown in Figure 3-3, which show that the reactor would have to operate at 490 kW to obtain a temperature of 175 ° C in an IFE operating with a power factor of 0.67.

The power generated in the IFE at the predicted 490 kW would be: 490 kW * 0.67 / 93 = 3.53 kW. Moving the IFE to a different location having a different peaking factor will give a temperature of 175 ° C whenever the power generated in the IFE is equal to or greater than 3.53 kW. Figure 3-3 also shows that at the design power of 300 kW, the IFE temperature would be about 136 ° C. Finally, to achieve an IFE temperature reading of 175 ° C at 300 kW, the power factor would have to be 1.09.

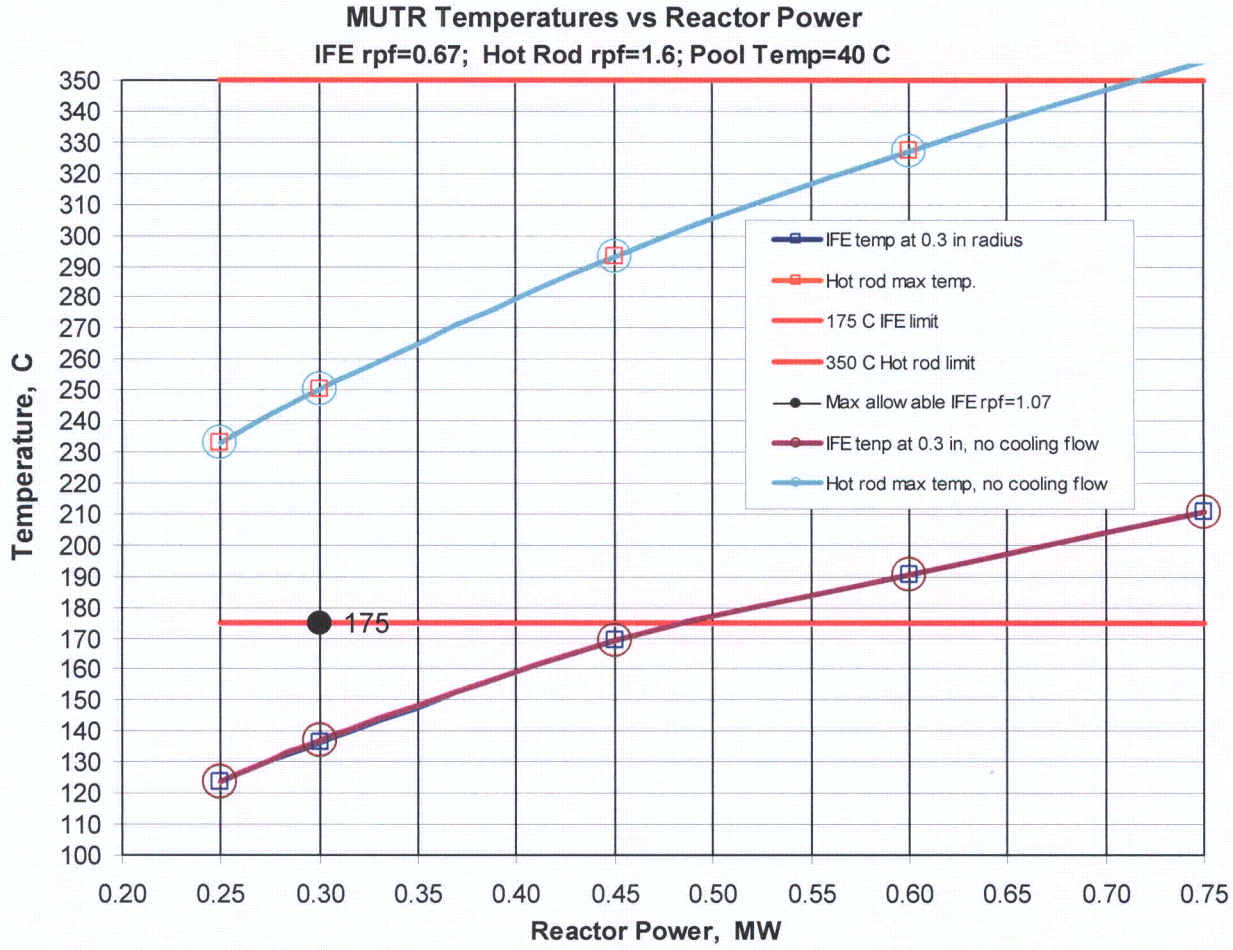

Figure 3-3: IFE and Hot Rod Temperatures as a Function of Reactor Power

In addition, the RELAP results for the hot rod (rod power factor 1.6) are also shown in Figure 3-3. The results show that for the hot rod, the reactor would have to operate at 720 kW to exceed the maximum hot rod fuel temperature of 350° C. When the core has a power level of 720 kW, an average fuel rod generates 720 kW/93 = 7.74 kW. If the IFE at this power level was only generating 3.53 kW, then its power factor is 3.53/7.74 = 0.456.

3.4 Result - Question #2b

The nuclear and thermal hydraulic analyses that were performed show that the IFE could be placed in any core location as long as its power factor is at least 0.456. As shown in Figure 3-4, there are four peripheral locations in clusters B4, C3, and C8 where the minimum peaking factor calculated using

DIF3D is less than 0.456. At these locations the fuel temperature limit of 350 ° C would be exceeded while the IFE still measure 175 ° C. Thus the IFE could be placed in any other core location and NOT exceed the fuel temperature limit of 350 ° C.

F			0.544	0.659	0.813	0.937	1.204	1.069	1.065	1.013	0.923	0.800	0.653	0.549		
			0.620	0.771	1.000	1.270	1.270	1.302	1.295	1.249	1.238	0.974	0.758	0.614		
E			0.705	0.889	1.259		1.576	1.508	1.496	1.538		1.212	0.864	0.663		
			0.746	0.945	1.226	1.520	1.557	1.597	1.578	1.505	1.443	1.155	0.900	0.678		
D			0.732	0.930	1.183	1.375	1.497	1.549	1.523	1.423	1.270	1.078	0.858	0.650		
			0.661	0.841	1.069	1.238	1.361	1.493	1.365	1.272	1.260	1.076	0.767	0.585		
C			0.542	0.686	0.871	1.006	1.193		1.187	1.137			0.711	0.490		
			0.447	0.569	0.719	0.828	0.916	1.209	0.916	0.934			0.587	0.411		
B											0.620	0.520				
											0.358	0.297				
	9		8		7		6		5		4		3		2	

Figure 3-4: IFE Locations where 350° C Temperature Limit could be exceeded

The analyses also show that there is a limit on where the IFE can be located in the core and still maintain the capability to meet the Tech Spec Requirements of the 300 kW safety scram. To make sure that this is not limiting, the IFE would have to be located in any location that has a peaking factor less than 1.09 as shown in Figure 3-5.

F			0.544	0.659	0.813	0.937	1.204	1.069	1.065	1.013	0.923	0.800	0.653	0.549		
			0.620	0.771	1.000	1.270	1.270	1.302	1.295	1.249	1.238	0.974	0.758	0.614		
E			0.705	0.889	1.259		1.576	1.508	1.496	1.538		1.212	0.864	0.663		
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			0.447	0.569	0.719	0.828	0.916	1.209	0.916	0.934			0.587	0.411		
B											0.620	0.520				
											0.358	0.297				
	9		8		7		6		5		4		3		2	

Figure 3-5: IFE Locations where 175° C Temperature Limit can be exceeded prior to reaching 300 kW

Finally, Figure 3-6 shows all the core locations that the IFE can be located and not exceed either the 350° C fuel temperature limit or the 175° C IFE temperature trip setting.

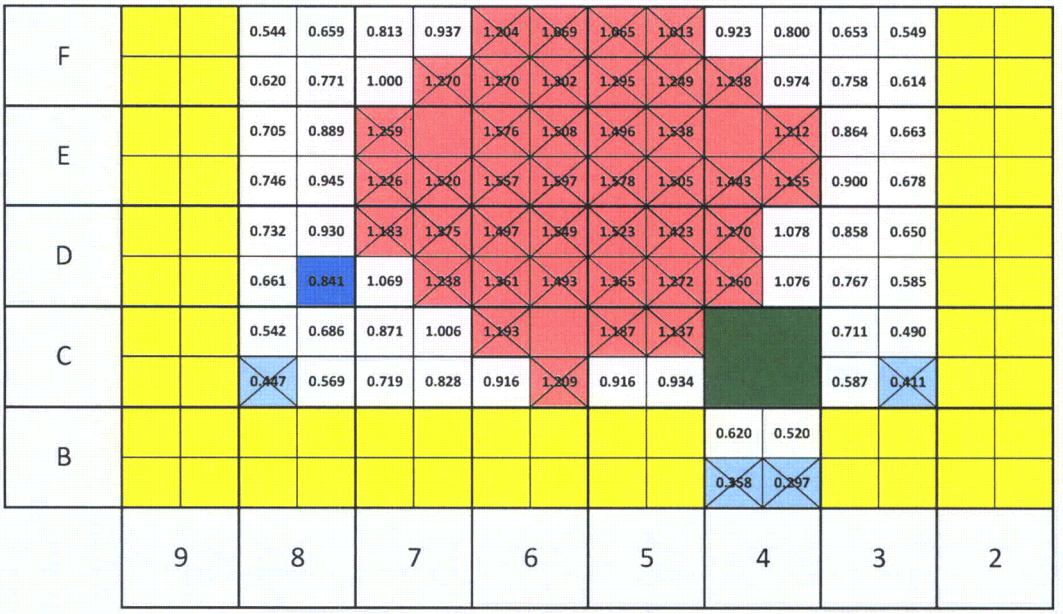

Figure 3-6: Possible IFE Location

4 REFERENCES

1. Ellis, C., "Support Calculations for MUTR's Request for Additional Information," General Atomics, GA-911199, December 2010.
2. Sherman, R. "DIF3D Code Validation Report," 21C023 Rev. 0, April 2000.
3. "RELAP5/MOD3.3 Code Manual Volume 3: Developmental Assessment Problems," NUREG-CR-5535/Rev. P3 – Volume III.

APPENDIX A: SUMMARY DIF3D NUCLEAR ANALYSIS

Power peaking analyses were performed using the DIF3D multi-dimensional diffusion theory code. The analyses used the cross-sections generated for Beginning-of-Life concentrations at the approximate average fuel temperature of 200°C, the closest nuclear data available. The core model for the DIF3D analyses is based on the MUTR core arrangement shown in Figure A-1.

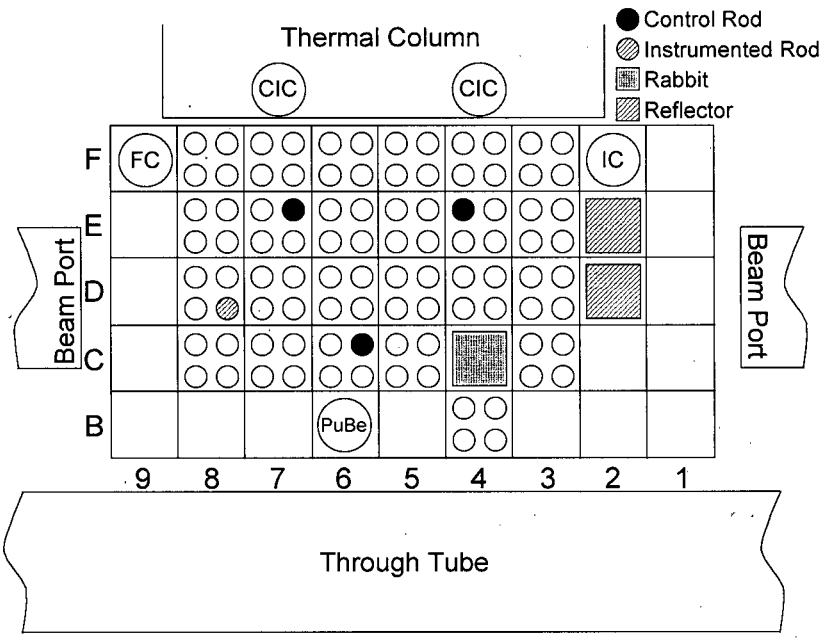

Figure A-1: University of Maryland Core Design

The fuel and cluster assemblies are shown in Figure A-2. The key parameters for the fuel and components are given in Table A-1. Power peaking in the core was analyzed on the basis of rod power factor, which is the power generation in a fuel rod (element) relative to the core averaged rod power generation. Since maximum fuel temperature is the limiting operational parameter in the core, this rod power factor is of greatest importance for steady-state operation.

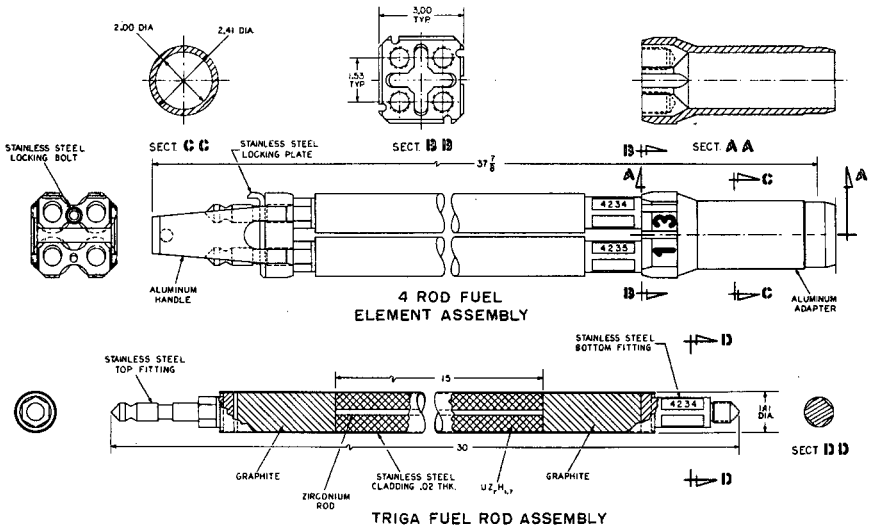

Figure A-2: Fuel Element and Cluster Assemblies

Table A-1: Key Fuel and Reactor Parameters

DESIGN DATA	MUTR CORE
Number of Fuel Rods	93
Fuel Type	UZrH
Uranium Enrichment, %	19.75
Zirconium Rod Outer Diameter, mm	5.715
Fuel Meat Outer Diameter, mm	34.823
Fuel Meat Length, mm	381
Clad Thickness, mm	0.508
Clad Material	304 SS
REACTOR PARAMETERS	
Reactor Steady State Operation, kW	250
Testing, kW	300
Maximum Fuel Temperature at 300 kW, °C	250
Maximum Fuel Temperature at 250 kW, °C	233
Maximum Rod Power at 300 kW, kW/element	5.15
Maximum Rod Power at 250 kW, kW/element	4.29
Average Rod Power at 300 kW, kW/element	3.23

The results from the calculations gave rod power factors that ranged from a low of 0.297 for a rod in the cluster in core location B4, to a maximum of 1.597 for the hot rod which is located in cluster E6. The corresponding rod powers for these peaking values give rod powers of 0.8 kW when the reactor is operating at the nominal power of 250 kW (0.96 kW at the design power of 300 kW) for the lowest powered rod and 4.29 kW (5.15 kW) for the hot rod. The IFE which is located in cluster D8 has a peaking factor of 0.841 which corresponds to a rod power of 2.26 kW (2.71 kW). The peaking power profile for the MUTR core is shown in Figure A-3. All files associated with these analyses are stored on the GA Modular Helium Reactor (MHR) Group's O-Drive ('Torna02.ga.com') as described in Appendix B of the calculation file that supplements this document (Ref. 1).

F			0.544	0.659	0.813	0.937	1.204	1.069	1.065	1.013	0.923	0.800	0.653	0.549		
			0.620	0.771	1.000	1.270	1.270	1.302	1.295	1.249	1.238	0.974	0.758	0.614		
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D			0.732	0.930	1.183	1.375	1.497	1.549	1.523	1.423	1.270	1.078	0.858	0.650		
			0.661	0.841	1.069	1.238	1.361	1.493	1.365	1.272	1.260	1.076	0.767	0.585		
C			0.542	0.686	0.871	1.006	1.193		1.187	1.137			0.711	0.490		
			0.447	0.569	0.719	0.828	0.916	1.209	0.916	0.934			0.587	0.411		
B											0.620	0.520				
											0.358	0.297				
	9		8		7		6		5		4		3		2	

Figure A-3: Peaking Factors, MUTR Core - 200°

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