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50-166

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United States Nuclear Regulatory Commission
Washington, D.C. 20555-0001

RE: Maryland University Training Reactor (License #R-70) responses to Request for Additional Information dated April 16, 2019

Enclosed please find the University of Maryland's responses to the 13 requests for additional information (ML19024A291) regarding the license amendment request dated the March 28, 2018 (ML18092A086).

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

Sincerely,

Amber S. Johnson

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NRR

RAI #1: Provide a copy of the OSU Radiation Center report titled, "Analysis of the Neutronic Behavior of the Maryland University Training Reactor," dated July 2017.

See attachment 1.

RAI #2: Provide a thermal-hydraulic analysis for the proposed core configuration or explain why the thermal-hydraulic analysis for the current core configuration, as referenced in the LAR, bounds the proposed core configuration. Alternatively, justify why additional information is not necessary.

A new thermal-hydraulic analysis is unnecessary. A comparison of Figure 1, the power distribution of the proposed core with Figure 2, the current core power distribution (from the neutronics analysis report), shows that the average power per element at full power will be reduced from 2.00 kW to 1.95 kW, and the power of the most reactive single element will be reduced from 3.86 kW to 3.52 kW. This indicates that the power peaking factor will be reduced and the current thermal hydraulic analysis is a conservative estimate.

	8		7		6		5		4		3	
F	7390	7391	7378	7379	7354	7355	7395	7393	7168	7169	7333	7335
	1.30	1.65	2.03	2.35	2.56	2.70	2.70	2.58	2.35	2.06	1.67	1.36
	7389	7392	7377	7380	7353	7356	7397	7396	7167	7166	7334	7336
	1.48	1.89	2.41	3.07	3.09	3.20	3.19	3.09	3.07	2.45	1.93	1.55
E	7161	7026	7398	259	7368	7365	7374	7375	304	7406	7342	7343
	1.68	2.16	2.97	0.00	3.74	3.66	3.66	3.71	0.00	2.97	2.20	1.69
	7028	7027	7399	7400	7367	7366	7373	7376	7404	7405	7341	7344
	1.78	2.29	2.95	3.65	3.74	3.85	3.86	3.67	3.58	2.89	2.29	1.75
D	7408	7409	7345	7346	7382	7383	7371	7372	7290	7330	7164	7165
	1.76	2.27	2.89	3.33	3.63	3.77	3.73	3.51	3.16	2.73	2.21	1.70
	7407	7160	7348	7347	7381	7384	7370	7369	7332	7331	7163	7162
	1.60	2.09	2.63	3.02	3.33	3.64	3.36	3.13	2.84	2.47	2.00	1.55
C	7360	7357	7352	7349	7401	260	7388	7385			7362	7363
	1.31	1.71	2.16	2.48	2.88	0.00	2.89	2.60			1.75	1.32
	7359	7358	7351	7350	7403	7402	7387	7386			7361	7364
	1.06	1.38	1.75	1.98	2.08	2.34	2.21	2.18			1.55	1.12
B									7338	7337		
									1.57	1.42		
									7339	7340		
									0.82	0.94		

Figure 1: Power distribution of the current core. The upper numbers of each pair are the element serial numbers, and the lower numbers are their power in kW at full power. The highest power element is highlighted in red.

	8		7		6		5		4		3	
F	7390	7391	7378	7379	7354	7355	7395	7393	7168	7169	7333	7335
	1.12	1.40	1.71	1.99	2.19	2.28	2.27	2.16	1.98	1.73	1.41	1.16
	7389	7392	7377	7380	7353	7356	7397	7396	7167	7166	7334	7336
	1.28	1.63	2.08	2.64	2.65	2.73	2.71	2.62	2.58	2.06	1.63	1.32
E	7161	7026	7398	259	7368	7365	7374	7375	304	7406	7342	7343
	1.48	1.90	2.60	0.00	3.24	3.17	3.15	3.18	0.00	2.52	1.86	1.46
	7028	7027	7399	7400	7367	7366	7373	7376	7404	7405	7341	7344
	1.63	2.09	2.67	3.29	3.34	3.44	3.41	3.25	3.16	2.54	2.02	1.55
D	7408	7409	7345	7346	7382	7383	7371	7372	7290	7330	7164	7165
	1.68	2.15	2.71	3.11	3.36	3.47	3.42	3.22	2.91	2.51	2.03	1.56
	7407	7160	7348	7347	7381	7384	7370	7369	7332	7331	7163	7162
	1.63	2.10	2.62	2.99	3.22	3.52	3.23	3.01	2.76	2.42	1.93	1.50
C	7360	7357	7352	7349	7401	260	7388	7385			7362	7363
	1.49	1.90	2.37	2.66	3.01	0.00	3.02	2.71			1.82	1.38
	7359	7358	7351	7350	7403	7402	7387	7386			7361	7364
	1.27	1.63	2.00	2.21	2.37	2.60	2.33	2.30			1.61	1.18
B	6286	6284	5861	6281			6287	6289	7338	7337	6282	6277
	1.01	1.28	1.55	1.78			1.86	1.76	1.76	1.59	1.21	0.93
	6283	6285	5862	5864			6279	6290	7339	7340	6288	6268
	0.73	0.92	1.13	1.36			1.43	1.28	1.19	1.05	0.86	0.68

Figure 2: Power distribution of the proposed core. The upper numbers of each pair are the element serial numbers, and the lower numbers are their power in kW at full power. The highest power element is highlighted in red.

RAI #3: Confirm that TS 4.1, Specification 4, in the LAR should include the inspection of bundle C6. If not, explain the apparent discrepancy between the information provided in the LAR and current MUTR TSs.

This is a typo in the LAR. Inspection of bundle C6 is required. The current MUTR Tech Spec reads "A visual inspection of a representative group of fuel bundles from row C column 8,7,6,5,3 and row B column 4 shall be performed annually, at intervals not to exceed 15 months. If any are found to be damaged, an inspection of the entire MUTR core shall be performed."

RAI #4: Provide an explanation of the substantial difference between the calculated and measure reactivity worths for Shim 1 and Shim 2. Additionally, state any ranges of acceptability between the simulated and measured reactivity worths and what action will be taken if the range is exceeded.

In the neutronics report individual simulated rod worths are simply the absolute value of "all rods in" minus the absolute value of the worth of the individual rod. Thus the control rod worths of the rods are Shim I: \$3.80 ± \$0.05, Shim II: \$3.87 ± \$0.06, Reg: \$2.82 ± \$0.06 and the total rod worth of \$10.48 ± \$0.10.

Rod worths are typically calculated using the asymptotic period method, but this can suffer from the effects of rod shadowing. We tried measuring the rod worths again by the rod drop method to reduce these effects. This resulted in worths of $3.22 for Shim 1, $4.25 for Shim 2, and $2.48 for the Reg Rod, or a total worth of $9.95. This is a much better agreement with the simulated values (within 20% in all cases, and within 6% in total worth.

Rod worths can differ by approximately 20% from year to year so this is considered an acceptable deviation from the simulated values.

RAI #5: Provide a justification that the rod withdrawal analysis provided by the December 18, 2006, letter bounds the proposed excess reactivity LCO of $3.50. If not, provide a revised rod withdrawal analysis that considers an excess reactivity of $3.50. Alternatively, justify why additional information is not necessary.

The rod withdrawal analysis from the December 18, 2006 RAI response does not provide a bounding condition for the proposed excess reactivity LCO of $3.50. A hypothetical rapid insertion of $3.70 in excess reactivity is described in sections 13.1.2 and 13.2.2 of the SAR.The hypothetical rapid insertion of $3.70 at 0.01kW and 250kW corresponds to peak fuel temperatures of 692°C and 988°C respectively. This temperature is lower than the safety limit temperature of the fuel cladding, 1000°C, thus the fuel integrity would not be lost. Therefore, this hypothetical instantaneous insertion of $3.70 would not result in the release of any fission products from the primary barrier. A slow ramp insertion of excess reactivity might be expected to produce higher fuel surface temperatures in a fuel with smaller or slower negative temperature feedback. However, this is not a concern for TRIGA® fuel, which introduces a large prompt negative feedback as the fuel meat heats up, greatly reducing the power density at that location. A reactor with fuel designed for the harsh conditions produced by a pulse can withstand a continuous rod withdrawal. Further information is not necessary.

RAI #6: Provide an analysis that shows the relationship between the measured fuel temperature in the IFE D8 position and the maximum fuel temperature within the proposed core configuration. Alternatively, justify why additional information is not necessary.

The IFE is located in position D8 NE corner. The power of this element is 2.10 kW in the proposed core. The hottest element in the proposed core is D6 NE with a power of 3.52 *kW (Figure 3). The power of these elements differs by less than a factor of 2. The basis for TS 2.2 establishes that the temperature of the IFE can differ from the hottest element by no more than a factor of 2 to allow for a 650C safety margin with an IFE trip setting of 175C. Since the power of D6 NE is less than twice the power of D8 NE, the basis for TS 2.2 is still being met and further analysis is unnecessary.*

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	1.48	1.90	2.60	0.00	3.24	3.17	3.15	3.18	0.00	2.52	1.86	1.46
	7028	7027	7399	7400	7367	7366	7373	7376	7404	7405	7341	7344
	1.63	2.09	2.67	3.29	3.34	3.44	3.41	3.25	3.16	2.54	2.02	1.55
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	1.68	2.15	2.71	3.11	3.36	3.47	3.42	3.22	2.91	2.51	2.03	1.56
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	1.49	1.90	2.37	2.66	3.01	0.00	3.02	2.71			1.82	1.38
	7359	7358	7351	7350	7403	7402	7387	7386			7361	7364
	1.27	1.63	2.00	2.21	2.37	2.60	2.33	2.30			1.61	1.18
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	1.01	1.28	1.55	1.78			1.86	1.76	1.76	1.59	1.21	0.93
	6283	6285	5862	5864			6279	6290	7339	7340	6288	6268
	0.73	0.92	1.13	1.36			1.43	1.28	1.19	1.05	0.86	0.68

Figure 3: Power distribution of the proposed core. The upper numbers of each pair are the element serial numbers, and the lower numbers are their power in kW at full power. The IFE is serial number 7160.

RAI #7: Clarify the units of measure used in Figure 2 of the LAR or explain why units of measure are not needed.

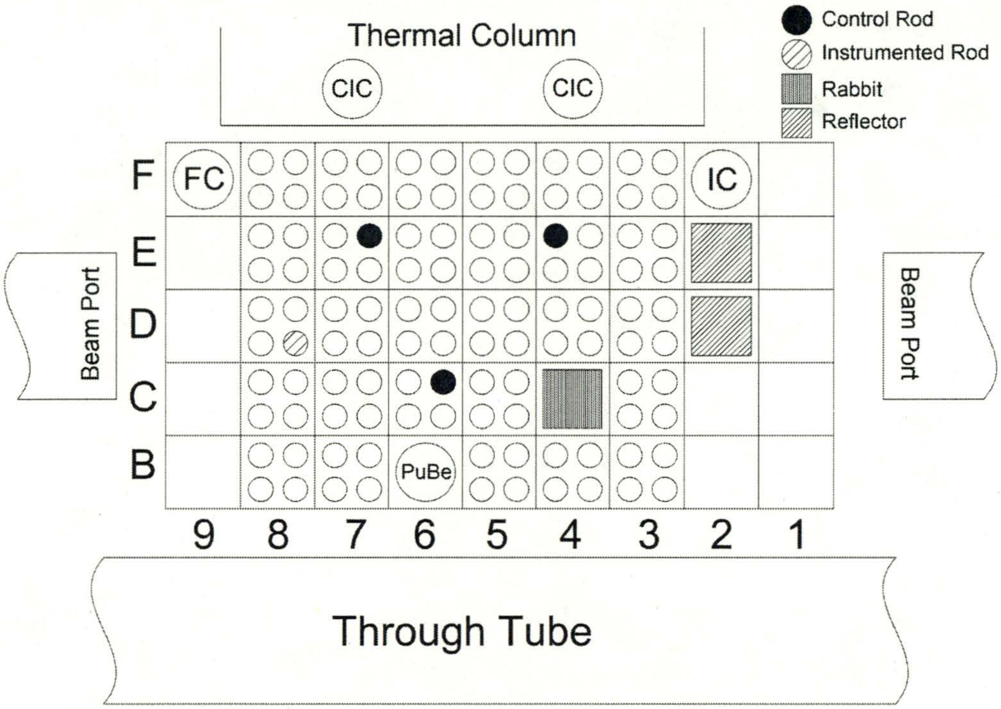
The units of Figure 2 are kW for the lower of each pair of numbers. The upper of the number pairs are element serial numbers.

RAI #8: LAR Figure 3, "Proposed core configuration power peaking factors," provides information on power peaking factors. However, the ratio used to develop these peaking factors are not clear. Provide an explanation of the ratio used to develop the power peaking factors presented in LAR Figure 3. Alternatively, justify why additional information is not necessary

Figure 3 is miscaptioned; it should be labeled "Proposed core power distribution". The units of Figure 3 are kW for the lower of each pair of numbers. The upper of the number pairs are element serial numbers.

RAI #9: Provide an illustration of the proposed core configuration showing the locations of fuel bundles that may be displaced for the in-core pneumatic experimental system, plutonium-beryllium source, neutron detectors, and graphite reflector elements. Alternatively, justify why additional information is not necessary.

See below. Elements are displaced from grid position B6 for the plutonium-beryllium source and grid position C4 for the in-core pneumatic experimental system. Neutron detectors and graphite reflectors are located on the periphery of the core.



RAI #10: Provide a shutdown margin analysis that includes relevant uncertainties, error limits, and worst-case conditions and takes into account the difference between the simulated and measured control rod worths for the current core configuration. Provide an explanation of why the proposed core configuration simulation provides acceptable predictions of control rod worths for determining that the shutdown margin will be maintained for the proposed core configuration. Alternatively, justify why additional information is not necessary.

Further information is not necessary, adding fuel to the core will increase the control rod worths. Assuming a worst case scenario where the worths of the control rods do not change from their current values (LAR table 2), while the excess reactivity increases to *the proposed limit of $3.50, the shutdown margin of the reactor, given by the total rod worth minus the most reactive rod minus the excess reactivity is: ($2.08+$2.75+$2.20)-($2.75)-($3.50)=$0.78. Comfortably above the Tech Spec limit of $0.50.*

RAI #11: Provide an explanation for the apparent discrepancy between the statements in the LAR above referring to a $4.00 insertion of excess reactivity and MUTR TS 3.6, Specification 2. In addition, for the proposed core configuration, provide the basis for selecting $4.00 of reactivity as a bounding analysis for a credible prompt insertion of reactivity, especially given that your proposed excess reactivity is $3.50.

$4.00 was chosen as a conservative estimate of maximum possible reactivity insertion. The central fuel bundle has an estimated worth of $4.70. (SAR 13.2.2.3) If the central bundle were to be dropped into the core while critical, the reactivity insertion above prompt critical would be approximately $4.00. This was analyzed as the maximum possible reactivity insertion although there is no conceivable scenario in which this could happen. This situation is analysed in section 13.2.2.3 of the SAR with an insertion of $3.70.

RAI #12: Clarify the request for the grammar change of the current LC 2.B.2.d.

The licensee request was to update current license condition 2.B.2.e. However, due to the requested deletion of 2.B.2.b, 2.B.2.e would become 2.B.2.d. The licensee now understands that the NRC prepares the updates to the license and providing the text is unnecessary.

RAI #13: Provide an explanation if UMD can perform an adequate visual inspection of the remaining fuel bundles (i.e., other than fuel bundles in rows B and C) in the core. If so, state which fuel bundles can be adequately inspected. Include additional information in TS 4.1, Specification 4, to explicitly state which and how many fuel bundles will be inspected on an annual frequency. Alternatively, justify why additional information is not necessary.

UMD proposes a change to MUTR TS 4.1, "Reactor Core Parameters," to clarify the fuel bundles that are inspected annually as follows:

4.1 Reactor Core Parameters

4. A visual inspection of 2 fuel bundles from rows B and C shall be performed annually at intervals not to exceed 15 months. The bundles inspected shall change each year so that in a 5 year period the entire group will be inspected. If any are found to be damaged, an inspection of the entire MUTR core shall be performed.

In response to RAI Nos. 4, 9, and 31 (Refs. 15, 55, and 96), we provided justification for not inspecting and measuring all fuel elements in the core, or performing length and bend measurements. The MUTR does not pulse, does not use a forced circulation coolant system, has relatively low fuel burn up given its operating history, uses stainless steel fuel elements, has a low risk of damage to instrumentation, and is only currently licensed for a power of 250 kW. Therefore visually inspecting the fuel in grid plate locations listed in TS 4.1, Specification 4 provides an adequate representative profile of all other fuel elements in the core. However, if an annual inspection identifies damaged fuel, then the entire core would be visually inspected for damage in accordance with TS 3.1 Specification 4. In order to measure the fuel elements, the fuel bundles would need to be disassembled, which has never been undertaken because it presents risk of fuel and instrumentation damage. Therefore, TS 4.1, Specification 4, is a reasonable alternative to the guidance in NUREG-1537, Appendix 14.1, Section 4.1.

Attachment 1

"Analysis of the Neutronic Behavior of the Maryland University Training Reactor"