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Subject: Docket No. 50-166, License No. R-70, University of Maryland. Submittal of request for license amendment authorizing changes to technical specifications for the use of 16 additional TRIGA fuel elements in the Maryland University Training Reactor

The Maryland University Training Reactor (MUTR) hereby requests an amendment to the Facility Operating License No. R-70 for the addition of 16 lightly used TRIGA fuel elements into the current 93 element core. Attached is a Safety Analysis report formatted in accordance with NUREG 1537, chapter 16.1-Prior Use of Reactor Components. The technical analysis for this report was provided by Oregon State University Radiation Center.

If there are any questions or concerns with this request, please contact Amber Johnson at ajohns37@umd.edu or 301-405-7756.

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

Sincerely,

Timothy W Koeth

Timothy W. Koeth

A020
NRR

The Department of Energy selected the Maryland University Training Reactor (MUTR) to receive the inaugural shipment of 19 lightly used fuel elements from storage at Idaho National Laboratory (INL). Fuel elements have been transferred directly between TRIGA installations during the decommissioning process, for example, from University of Arizona to Reed College. Based upon data taken when the elements were relocated from the University of Wisconsin Nuclear Reactor (UWNR) to INL in 2009, 25 elements were selected as potential shipment candidates. In early March 2017, 21 elements were individually removed from dry-cask storage at INL for radiation readings and visual confirmation of cladding integrity. Three experts employed at the Idaho Nuclear Technology and Engineering Center (INTEC) visually inspected the fuel elements as they were retrieved from storage before loading into the BRR cask. Two elements were rejected for possible pitting or surface damage.

The fuel elements entered into service between 1967 and 1970 when the UWNR began operation as a 1000kW TRIGA with pulsing capabilities. The elements were placed into storage when the reactor converted to FLIP fuel in the late 1970s. The TRIGA fuel was stored on site until INTEC removed the elements for dry-cask storage at INL in 2009. Information about the elements is gathered in Table 1.

*Table 1: Information about fuel elements transferred from UWNR to MUTR found in Required Shipper's Data*

| ID Number | Initially loaded in core | Removed from core | Element Burnup (MW-days) | Element Burnup (% U-235) |
|---|---|---|---|---|
| **4415** | 1/19/67 | 3/08/74 | 1.47 | 5.25 |
| **5859** | 9/12/69 | 6/15/79 | 0.70 | 2.46 |
| **5860** | 9/12/69 | 6/15/79 | 0.70 | 2.49 |
| **5861** | 9/12/69 | 6/15/79 | 0.64 | 2.28 |
| **5862** | 9/12/69 | 6/15/79 | 0.64 | 2.28 |
| **5864** | 9/12/69 | 6/15/79 | 0.64 | 2.28 |
| **6268** | 2/18/70 | 6/15/79 | 0.14 | 0.51 |
| **6277** | 2/18/70 | 6/15/79 | 0.14 | 0.51 |
| **6279** | 2/18/70 | 6/15/79 | 0.42 | 1.50 |
| **6281** | 2/18/70 | 6/15/79 | 0.42 | 1.50 |
| **6282** | 2/18/70 | 6/15/79 | 0.14 | 0.51 |
| **6283** | 2/18/70 | 6/15/79 | 0.66 | 2.34 |
| **6284** | 2/18/70 | 6/15/79 | 0.66 | 2.34 |
| **6285** | 2/18/70 | 6/15/79 | 0.66 | 2.34 |
| **6286** | 2/18/70 | 6/15/79 | 0.66 | 2.34 |
| **6287** | 2/18/70 | 6/15/79 | 0.42 | 1.50 |
| **6288** | 2/18/70 | 6/15/79 | 0.42 | 1.50 |
| **6289** | 2/18/70 | 6/15/79 | 0.42 | 1.50 |
| **6290** | 2/18/70 | 6/15/79 | 0.42 | 1.50 |

The fuel was transferred to storage buckets in the MUTR pool during the week of March 20, 2017. With support from the Department of Energy Office of Nuclear Energy and their

contractors, the elements were individually transferred from the BRR cask to a transfer cask and then lowered into the pool.

These additional fuel elements have the same characteristics as the stainless steel clad fuel continuously in use at MUTR since 1974. This original style TRIGA fuel, evaluated for safety in NUREG-1282, contains approximately 8.5% net weight Uranium enriched to less than 20% U-235. The elements will be assembled into bundles of 4 for installation into the core support structure.

To optimize reactor efficiency while maintaining safety margins, 16 elements will be added to the core, increasing the inventory to 109 elements. The new bundles will be added to grid plate positions nearest the through-tube, indicated in blue in Figure 1.

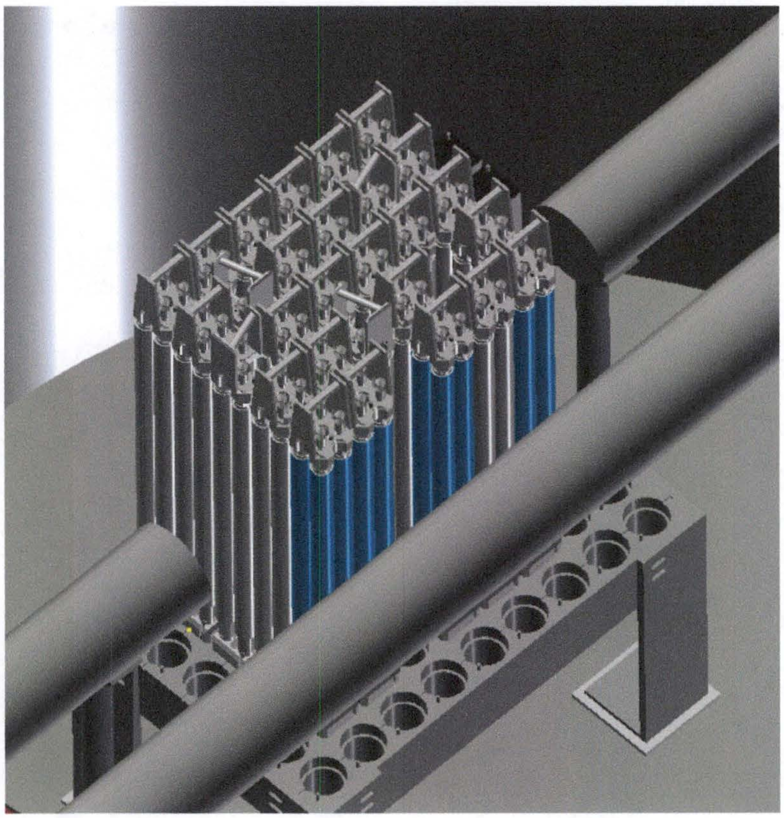

*Figure 1: Proposed location of additional elements.*

MCNP calculates an expected core excess of \$2.88 versus the current measurement of \$0.33 and the technical specification limit of \$1.12. All are less than the original technical specification limit of \$3.50. The most reactive rod is now expected to be the Regulating Rod. Rod worths are

simulated to be: Shim I $3.00, Shim II $3.09 and Regulating Rod $3.26. The expected shutdown margin of $3.22 would still far exceed the technical specification limit of $0.50.

The power per element was simulated to confirm placement of the instrumented fuel element (IFE) in D8. Current core power distribution can be seen in Figure 2. While the hottest element is located at E5, the power produced in the IFE is greater than 50% of the power produced in the hottest fuel element so it is acceptable in its current location.

| | 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 2: Current core power distribution.*

For the suggested 109 element core configuration, the hottest element is located at D6, shown in Figure 3. The IFE would still trip before another element exceeds the limiting safety system setting.

| | 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 3: Proposed core configuration power peaking factors.*

**Proposed Changes to the Technical Specifications**

This section contains suggested changes to the technical specifications in order for the fuel to be successfully installed in the core for use.

Change to Section 1.3

***CORE CONFIGURATION****-The core consists of 24 fuel bundles, with a total of 93 elements, arranged in a rectangular array with one bundle displaced for the pneumatic experimental system; three CONTROL RODS; and two graphite reflectors.*

The definition will need to be updated for the installation of the additional fuel bundles. The suggested definition is:

***CORE CONFIGURATION*** *– The core consists of TRIGA fuel elements assembled into THREE or FOUR ELEMENT FUEL BUNDLES, arranged in a close-packed rectangular 5x9 configuration. Bundles shall be displaced for the in-core pneumatic experimental system, PuBe source, neutron detectors, and graphite reflector elements.*

Change to Section 3.1

*3.1 Reactor Core Parameters*

1. *The EXCESS REACTIVITY relative to the REFERENCE CORE CONDITION, with or without experiments in place shall not be greater than $1.12.*

Previously, in response to RAI #84 asked on October 20, 2002 and answered on December 18, 2006, the excess reactivity was limited to $1.12. This question initially asked about the power ramp that would result if $0.30/second was added to the reactor starting from a low power condition. This can best be answered using a single delayed neutron group model with prompt jump approximation, power as a function of time is given by:

$$\frac{P(t)}{P_0} = e^{-\lambda t}\left[\frac{\beta}{\beta - \gamma t}\right]^{(1+\lambda\beta/\gamma)}$$

where $P(t)$= power at time t

$P_0$ = initial power level

$\beta$ = total delayed to neutron fraction=0.007

$\lambda$ = one group decay constant = 0.405 sec$^{-1}$

$t$ = time (sec)

$\gamma$ = linear insertion rate of reactivity (Δk/k-sec$^{-1}$)

Control rod data determined through annual surveillances is shown in table 2.

*Table 2: Control Rod Data*

| Rod | Total Worth ($) | Total Withdrawal Time (sec) | Average Insertion Rate ($/sec) | Total Reactivity at SCRAM ($) |
|---|---|---|---|---|
| Shim I | 2.08 | 48.77 | 0.0426 | |
| Shim II | 2.75 | 52.16 | 0.0527 | 0.98 |
| Regulating | 2.20 | 55.96 | 0.0393 | |
| - | - | - | 0.30 | 1.15 |

For our current core configuration, Shim II has the highest total worth and the highest average reactivity insertion rate. For the ramp insertion rate response of the reactor safety system, initial power levels of 1mW and 220kW will be considered. The SCRAM set point is 300 kW and 0.5 seconds delay time is assumed between reaching the SCRAM set point and the actual release of the control rods. For the case of 1mW and Shim II insertion rate, the reactor power was calculated to trip at 18.61 sec and the peak reactivity insertion was $0.98. Starting at 220kW, the reactor tripped after 3.71 sec and the peak reactivity insertion was $0.20. Using the technical specification limit of $0.30/sec at 1mW, the reactor tripped after 3.83 sec and the peak reactivity insertion was $1.15.

Using the Fuchs-Nordheim technique (GA-7882), the total reactivity values determined in Table 2 are shown to be well below the limits that would produce any adverse safety effects.
Average fuel temperature:

$$\Delta T = \frac{2\Delta k_p}{\alpha}$$

Total energy release:

$$E = \frac{2C\Delta k_p}{\alpha}$$

The peak power:

$$P_{max} = \frac{C(\Delta k_p)^2}{2l\alpha} + P_0$$

where:

$l$ = the prompt neutron lifetime = $7.3 \times 10^{-5}$sec

$\alpha$ = the prompt negative temperature coefficient = $1.25 \times 10^{-4} \Delta k/k$

$C$ = the total heat capacity of the core available to the prompt burst energy release = $9.6 \times 10^{4}$watt-sec/°C per core

$\Delta k_p$ = portion of the step reactivity insertion which is above prompt critical = 0.021 ($4.00)

As an upward bound, a $4.00 insertion of excess reactivity will be analyzed as the credible option for a prompt insertion of reactivity. This number is taken from technical specification 3.6.2, the total reactivity worth of an experiment. The reactor will be assumed to be operating at an initial power of 220kW. A total peaking factor of 1.6 from the GA thermal analysis completed on February 2, 2011.

| Average Final Fuel Temperature (°C) | Peak Final Fuel Temperature (°C) | Peak Power (MW) | Energy Released (MW-s) |
|---|---|---|---|
| 337 | 538 | 2320 | 32 |

Thus, this confirmatory calculation shows that the peak fuel temperature remains below the guidance stated in NUREG 1537 of 1,150°C. Thus, a peak reactivity insertion of $1.15 is determined to have no adverse safety effects.

The excess reactivity is calculated by bringing the reactor to low power critical and determining the amount of reactivity left in the core using the measured control rod worth curves. Due to poison build up throughout the forty years of operation, this number has been drastically reduced from the $3.50 initially licensed. Using the control rod values from Table 3, the Shutdown Margin is calculated from the total rod worth minus the most reactive rod minus the excess reactivity. An upper limit of $0.50 on the Shutdown Margin is defined in technical specification 3.1.2. Allowing for an excess reactivity of $3.50, guarantees that the shutdown margin will always be maintained. As such, it is suggested that the specification be rewritten:

*3.1 Reactor Core Parameters*

1. *The EXCESS REACTIVITY relative to the REFERENCE CORE CONDITION, with or without experiments in place shall not be greater than $3.50.*

Change to Section 4.1

*4.1 Reactor Core Parameters*

4. *A visual inspection of a representative group of fuel bundles from row C column 8,7,5,3 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.*

With the addition of the new fuel bundles, the visual inspection requirement should be updated to read:

*4.1 Reactor Core Parameters*

4. *A visual inspection of a representative group of fuel bundles from rows B and C 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.*

Change to section 5.3

*5.3 Reactor Core and Fuel*

1. *The core shall consist of 93 TRIGA fuel elements assembled into 24 fuel bundles, 21 bundles shall contain four fuel elements and 3 bundles shall contain three fuel elements and a CONTROL ROD guide tube.*
2. *The fuel bundles shall be arranged in a rectangular 4x6 configuration, with one bundle displaced for the in-core pneumatic experimental system.*
3. *The reactor shall not be operated at power levels exceeding 250 kW.*
4. *The reflector shall be a combination of two graphite reflectors.*

The analysis shows that the MUTR can support more fuel than was originally loaded. Due to the short core lifetime of standard TRIGA fuel elements, approximately 100 MW-days, the core needs to be overloaded to compensate for reactivity loss due to fuel depletion and poison buildup. The addition of fuel will all the MUTR to return to 250 kW operations as well as improve the flux in the experimental facilities. It is suggested that the specification be rewritten as such:

*5.3 Reactor Core and Fuel*

1. *The core shall consist of TRIGA fuel elements assembled into THREE or FOUR ELEMENT FUEL BUNDLES.*
2. *The fuel bundles shall be arranged in a close-packed rectangular 5x9 configuration, with bundles displaced for the in-core pneumatic experimental system, PuBe source, neutron detectors, and graphite reflector elements.*
3. *The reactor shall not be operated at power levels exceeding 250 kW.*
4. *The reflector shall be a combination of graphite reflectors and water.*

**Startup Plan-Additional Reactor Fuel**

Within 6 months following the completion of the loading of additional reactor fuel into the core, the following information will be summarized and submitted to the NRC.

1. **Initial Approach to Criticality**

   The loading of fuel bundles to obtain criticality shall be accomplished using the standard inverse multiplication curve (1/M) approach given by:

$$M = \frac{1}{1-k}$$

   which can be rearranged to yield:

$$1/_M = 1 - k$$

   where k ranges from 0 (no fuel) to 1 (criticality). The experimental values for 1/M are obtained by measuring the count rate at the initial core configuration, $C_0$, divided by $C_n$, the count rate after the $n^{th}$ bundle is loaded.

2. **Measurements to be Made After Achieving Criticality**

**2.1. Control Rod Calibrations**

The MUTR is equipped with 3 control rods that are routinely calibrated using the positive asymptotic method. Current measurements and simulation results complied in Table 3.

*Table 3: Rod worth measurements and calculations.*

| | BOL(MCNP) | Current (MCNP) | Current(Measured) | New Configuration(MCNP) |
|---|---|---|---|---|
| Reg Rod | \$2.75 | \$2.82 | \$2.20 | \$3.26 |
| Shim 1 | \$3.74 | \$3.80 | \$2.08 | \$3.00 |
| Shim 2 | \$3.75 | \$3.87 | \$2.75 | \$3.09 |

**2.2. Excess Reactivity**

Excess reactivity of the reactor will be determined.

**2.3. Calorimetric Power Calibration**

The calorimetric power calibration takes advantage of the fact that natural convection provides adequate cooling for a TRIGA core operating at power levels up to and including 2.0 MW. In the so-called "slope" method of calibration, the rate of temperature rise will be determined for the reactor pool water [dT/dt (°C/hr)] while the reactor is operating at power P and the tank water is stirred. Combined with the measured time rate of pool water temperature rise, the actual reactor power can be calculated from:

$$P(kW) = \left[\frac{dT/_{dt}\left({}^{\circ}C/_{hr}\right)}{TankConstant\left({}^{\circ}C/_{kW}\right)}\right]$$

### 2.4. Shutdown Margin

Shutdown margin shall be determined.

### 2.5 Primary Coolant Measurements

Results of any primary coolant water sample measurements for fission product activity taken during the first 30 days of operation after fuel loading.

### 2.6 Discussion of results

Discussion of the various results, including an explanation of any findings that could affect normal operations.