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

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. In response to the acceptance review letter dated March 9, 2018, we have revised the original request to include a concise purpose, proposed license changes, justifications for proposed technical specifications, proposed technical specifications and a list of references.

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
A001
NRR

**Purpose**

The Maryland University Training Reactor (MUTR) has struggled to reach licensed operating power of 250kW since 2012. Operational experience and simulation from Oregon State University confirmed that our original style TRIGA fuel is unable to overcome poison build-up and uranium depletion after 40 years of operations. The installation of these lightly used elements is expected to allow the MUTR to achieve licensed power.

**Background Information**

The Department of Energy selected the 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, is 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.

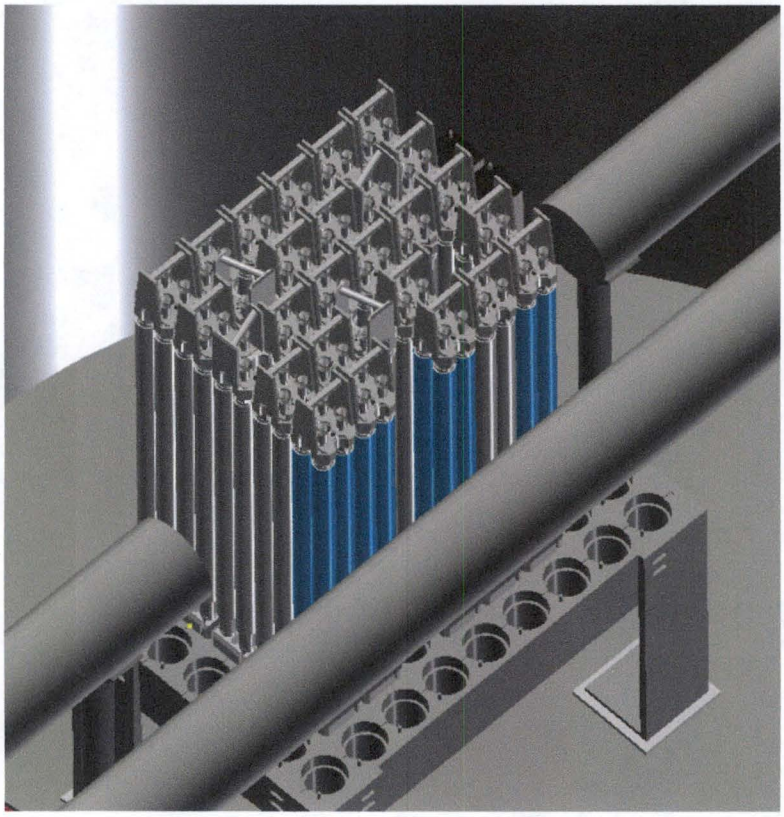

*Figure 1: Proposed location of additional elements.*

MCNP calculates an expected core excess reactivity of $2.88 versus the current measurement of $0.64 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 necessary changes to the technical specifications in order for the fuel to be installed in the core for use. Updated pages from the technical specifications are attached.

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 of core configuration will need to be updated with the installation of the additional fuel bundles. The new fuel will be assembled into THREE or FOUR ELEMENT FUEL BUNDLES, arranged in a close-packed configuration to match the grid plate. This ensures that the core fuel arrangement is closely-packed such that there are no open internal positions except as identified for the in-core pneumatic experimental systems, PuBe source, neutron detectors, and graphite reflector elements. This definition provides that internal core lattice positions are occupied with fuel elements to help reduce the probability of an accidental reactivity insertion at multiple lattice locations. 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 array. Bundles may be displaced for the pneumatic experimental system, PuBe source, neutron detectors, and graphite reflectors.*

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 ($\Delta k/k\text{-}sec^{-1}$)

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. A lower 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 shall 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.*

The specification as written requires the inspection of all bundles in rows B&C. With the addition of the new fuel bundles, rows B&C will be completely populated. The MUTR facility does not pulse, does not use a forced circulation coolant system, has relatively low fuel burn up given the operating history, and uses stainless steel fuel elements. This leads to a low risk of damage to instrumentation at the current licensed operating power of only 250kWt. Thus, visually inspecting the fuel elements in rows B&C would provide an adequate representation of the core. However, if an annual inspection should find damaged fuel, then the entire core would be inspected in accordance with TS 3.1 specification 4.
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.*

The analysis shows that the MUTR can safely 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 allow the MUTR to return to 250 kW operations as well as improve the flux in the experimental facilities. 5.3.1 limits the fuel to TRIGA elements assembled into defined THREE or FOUR ELEMENT FUEL BUNDLES. 5.3.2 is expanded to include the new fuel bundles and matches the definition of CORE CONFIGURATION. The new fuel will be assembled into THREE or FOUR ELEMENT FUEL BUNDLES, arranged in a close-packed configuration which matches the grid plate. This ensures that the core fuel arrangement is closely-packed such that there are no open internal positions except as identified for the in-core pneumatic experimental systems, PuBe source, neutron detectors, and graphite reflector elements. This definition provides that internal core lattice positions are occupied with fuel elements to help reduce the probability of an accidental reactivity insertion at multiple lattice locations. 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 array, with bundles displaced for the pneumatic experimental system, PuBe source, neutron detectors, and graphite reflectors.*

**Proposed R-70 License Changes**

Proposed changes are attached.

Updated 2.B.2.a to reflect the correct amount of U-235 available for use.

2.B.2.a to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 4,501 grams of contained uranium-235 enriched to less than 20 percent in the form of TRIGA-type reactor fuel;

Deleted 2.B.2.b, U-235 is accounted for in 2.B.2.a.

to receive, possess, but not use, and not separate, in connection with the operation of the facility, up to 1,060 grams of contained uranium-235 enriched to less than 20 percent in the form of "Alternate Reactor Fuel" TRIGA-type reactor fuel;

Deleted 2.B.2.f

to receive, possess, but not use, and not separate, in connection with the operation of the facility, such special nuclear material as may be produced by the operation of other facilities in the form of "Alternate Reactor Fuel."

Updated 2.B.2.d for grammar.

to receive, possess, and use, but not separate, in connection with the operation of the facility, such special nuclear material as may be produced by the operation of the facility.

Deleted 2.B.3.b

to receive, possess, but not use, and not separate, in connection with operation of the facility, such byproduct materials as may be produced by operation of other facilities in the form of "Alternate Reactor Fuel."

**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:

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

**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.

References:

GA Technologies. *Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors*. Washington DC: US Nuclear Regulatory Commission, 1987.

General Atomics. *Kinetic Behavior of TRIGA Reactors*. San Diego, CA: General Atomics, 1967.

*University of Maryland Reactor Analysis and Support*. San Diego, CA: General Atomics, 2011.

R. Schickler, S. Reese. *Analysis of the Neutronic Behavior of the Maryland University Training Reactor*. Corvallis, OR: Oregon State University, 2017.

"Technical Specifications License No. R-70, Docket No. 50-166." 2 December 2016.

U.S. Nuclear Regulatory Commission. "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content (NUREG 1537, Part 1)." February 1996.

"University of Maryland Docket No. 50-166 Renewed Facility Operating License." 2016.

University of Maryland Radiation Facilities. "RAI #84." College Park, MD, 18 December 2006.

U.S. Nuclear Regulatory Commission. *Safety Evaluation Report ML16075A214*. Washington DC: US Nuclear Regulatory Commission, 2016.

U.S. Nuclear Regulatory Commission. *Reed College Renewed Facility Operating License ML120530018*. Washington DC: US Nuclear Regulatory Commission, 2012.

J.D. Randall, D.R. Schad. *Operational Reactivity Considerations of Texas A&M TRIGA*. College Station, TX: Texas A&M University.

# 1 Introduction

## 1.1 Scope

Included in this document are the Technical Specifications and the "Bases" for the Technical Specifications. These bases, which provide the technical support for the individual technical specifications, are included for information purposes only. They are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

## 1.2 Format

These specifications are formatted to NUREG-1537 and ANSI/ANS 15.1-2007

## 1.3 Definitions

**ALARA** — An acronym for "as low as reasonably achievable", ALARA means making every reasonable effort to maintain exposures to radiation as far below the dose limits in 10 CFR Part 20 as is practical, consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest.

**AUTOMATIC MODE** — Automatic mode operation shall mean operation of the reactor with the mode selector switch in the automatic position.

**CHANNEL** — A channel is the combination of sensors, lines, amplifiers, and output devices which are connected for the purpose of measuring the value of a parameter.

**CHANNEL CALIBRATION** — A channel calibration is an adjustment of the CHANNEL such that its output corresponds with acceptable accuracy to known values of the parameter which the CHANNEL measures. Calibration shall encompass the entire CHANNEL, including equipment actuation, alarm, or trip and shall be deemed to include a channel test.

**CHANNEL CHECK** — A channel check is a qualitative verification of acceptable performance by observation of CHANNEL behavior, or by comparison of the CHANNEL with other independent CHANNELS or systems measuring the same variable.

**CHANNEL TEST** — A channel test is the introduction of a signal into the CHANNEL to verify that it is operable.

**CONFINEMENT** — Confinement means a closure of the overall facility that controls the movement of air into it and out, thereby limiting release of effluents, through a controlled path.

**CONTROL ROD** — A control rod is a device fabricated from neutron-absorbing material which is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged.

**CONTROL ROD GUIDE TUBE** — Hollow tube in which a CONTROL ROD moves.

**CORE CONFIGURATION** — The core consists of TRIGA fuel elements assembled into THREE or FOUR ELEMENT FUEL BUNDLES, arranged in a close-packed array. Bundles may be displaced for the pneumatic experimental system, PuBe source, neutron detectors and graphite reflectors.

# 3 Limiting Conditions for Operation

## 3.1 Reactor Core Parameters

Applicability

These specifications shall apply to the reactor at all times it is OPERATING.

Objective

The objective is to ensure that the reactor can be controlled and shut down at all times and that the SAFETY LIMIT will not be exceeded.

Specifications

1. The EXCESS REACTIVITY relative to the REFERENCE CORE CONDITION, with or without experiments in place shall not be greater than $3.50.
2. The SHUTDOWN MARGIN shall not be less than $0.50 with:
   (a) The reactor in the REFERENCE CORE CONDITION; and
   (b) Total worth of all experiments in their most reactive state; and
   (c) Most reactive CONTROL ROD fully withdrawn.
3. Core configurations:
   (a) The reactor shall only be operated with a STANDARD CORE.
   (b) No fuel shall be inserted or removed from the core unless the reactor is subcritical by more than the worth of the most reactive FOUR ELEMENT FUEL BUNDLE plus $0.50.
   (c) No control rod shall be removed from the core unless a minimum of four fuel bundles are removed from the core, having reactivity greater than the control rod.
   (d) The reactor shall only be operated with three OPERABLE control rods.
4. The reactor shall not be operated with damaged fuel except to locate such fuel. Fuel shall be considered damaged if:
   (a) A cladding defect exists as indicated by release of fission products, or
   (b) A visual inspection reveals bulges, gross pitting or corrosion.
5. The burn-up of U-235 in the UZrH fuel matrix shall not exceed 50% of the initial concentration.

Bases

1. While specification 3.1.1, in conjunction with specification 3.1.2, tends to over constrain the excess reactivity, it helps ensure that the OPERABLE core is similar to the core analyzed in the FSAR.
2. The value of the SHUTDOWN MARGIN as required by specification 3.1.2 assures that the reactor can be SHUTDOWN from any OPERATING condition even if the highest worth CONTROL ROD should remain in the fully withdrawn position.
3. Specification 3.1.3 ensures that the OPERABLE core is similar to the core analyzed in the SAR. It also ensures that accidental criticality will not occur during fuel or CONTROL ROD manipulations.

after each time the core fuel configuration is changed, these changes include any removal or replacement of CONTROL RODS.

3. CORE CONFIGURATION shall be verified prior to the first startup of the day.
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.
5. Burnup shall be determined annually, not to exceed 15 months.

Bases

Experience has shown that the identified frequencies ensure performance and operability for each of these systems or components. For EXCESS REACTIVITY and SHUTDOWN MARGIN, long-term changes are slow to develop. For fuel inspection, visually inspecting the bundles annually will identify any developing fuel integrity issues throughout the core.

## 4.2 Reactor Control and Safety Systems

Applicability

These specifications apply to the surveillance requirements for reactor control and safety systems.

Objective

The objective of these specifications is to ensure that the specifications of Section 3.2 are satisfied.

Specifications

1. The reactivity worth of each CONTROL ROD shall be determined annually, at intervals not to exceed 15 months, and after each time the core fuel configuration is changed or a CONTROL ROD is inspected.
2. The CONTROL ROD withdrawal and insertion speeds shall be determined annually, at intervals not to exceed 15 months, or whenever maintenance or repairs are made that could affect rod travel times.
3. CONTROL ROD DROP TIMES shall be measured annually, at intervals not to exceed 15 months, or whenever maintenance or repairs are made that could affect their DROP TIME.
4. All scram channels listed in Table 3.1 shall have a CHANNEL TEST, including trip actions with CONTROL ROD release and specified interlocks as listed in Table 3.2 performed after each SECURED SHUTDOWN, before the first operation of the day, or prior to any operation scheduled to last more than 24 hours, or quarterly, with intervals not to exceed 4 months. Scram channels and interlocks shall be calibrated annually, at intervals not to exceed 15 months.
5. CHANNEL TESTS shall be performed on all affected safety and control systems after any maintenance is performed.
6. A CHANNEL CALIBRATION shall be made of the linear power level monitoring channels annually, at intervals not to exceed 15 months.
7. A visual inspection of one of the CONTROL ROD poison sections shall be made annually, at intervals not to exceed 15 months. In a 3 year period, all sections shall be inspected.
8. A visual inspection of the CONTROL ROD drive and scram mechanisms shall be made annually, at intervals not to exceed 15 months.

2. Specification 5.2.2 ensures that the pool water level can normally decrease only by 50.8 cm (20 in) if the coolant piping were to rupture and syphon water from the reactor tank. Thus, the core will be covered by at least 4.57 m (15 ft.) of water.

## 5.3 Reactor Core and Fuel

Applicability

This specification applies to the configuration of the core and in-core EXPERIMENTS.

Objective

The objective is to ensure that the CORE CONFIGURATION is as specified in the license.

Specifications

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

### 5.3.1 Reactor Fuel

Applicability

This specification applies to the FUEL ELEMENTS used in the reactor core.

Objective

The objective is to assure that the FUEL ELEMENTS are of such design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics, and that the fuel used in the reactor has characteristics consistent with the fuel assumed in the SAR and the license.

Specifications

The individual unirradiated standard TRIGA fuel elements shall have the following characteristics:

1. Uranium content: a maximum of 9.0% weight uranium enriched to less than 20% $^{235}$U.
2. Zirconium hydride atom ratio: nominal 1.5 - 1.7 hydrogen-to-zirconium, ZrHx.
3. Cladding: 304 stainless steel, nominal thickness of 0.508 mm (.020 in).
4. The overall length of a FUEL ELEMENT shall be 30 inches, and the fueled length shall be 15 inches.

Basis

The design basis of the standard TRIGA FUEL ELEMENT demonstrates that 250 kW steady state operation presents a conservative limitation with respect to SAFETY LIMITS for the maximum temperature generated in the fuel.

G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

H. The issuance of this license is in accordance with 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession and use of byproduct and special nuclear materials as authorized by this facility operating license will be in accordance with the Commission's regulations in 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

2. Accordingly, Facility Operating License No. R-70 is hereby renewed in its entirety to read as follows:

A. This license applies to the Maryland University Training Reactor (herein "the facility") TRIGA-type nuclear research reactor owned by the University of Maryland (herein "the licensee"). The facility is located on the campus of the University of Maryland in College Park, MD, and described in the licensee's application for license renewal, dated May 12, 2000, as supplemented.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the University of Maryland as follows:

1. Pursuant to Subsection 104c of the Act and 1O CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility as a utilization facility at the designated location in accordance with the procedures and limitations described in the application and set forth in this license.

2. Pursuant to the Act and 10 CFR Part 70, the following activities are included:

a. to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 4,501 grams of contained uranium-235 enriched to less than 20 percent in the form of TRIGA-type reactor fuel;

b. to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 15 grams of special nuclear material, of any enrichment, in the form of detectors, fission plates, foils, and solutions;

c. receive, possess, and use, but not separate, in connection with the operation of the facility, up to 80 grams of plutonium contained in encapsulated plutonium-beryllium neutron sources;

d. to receive, possess, and use, but not separate, in connection with the operation of the facility, such special nuclear material as may be produced by the operation of the facility.

3. Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," the following activities are included:

a. to receive, possess, and use in connection with operation of the facility, such byproduct material as may be produced by operation of the facility, which cannot be separated except for byproduct material produced in non-fueled reactor experiments.

C. This license shall be deemed to contain, and is subject to the conditions specified in 10 CFR Parts 20, 30, 40, 50, 51, 55, 70, and 73 of the Commission's regulations; is subject to all provisions of the Act, and to the rules, regulations and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below:

1. Maximum Power Level

The licensee is authorized to operate the reactor at a steady-state power level up to a maximum of 250 kilowatts (thermal) in accordance with the limitations in the Technical Specifications.

2. Technical Specifications

The Technical Specifications contained in Appendix A are hereby incorporated in their entirety in the license. The licensee shall operate the facility in accordance with the Technical Specifications.