UNIVERSITY OF
MARYLAND

United States Nuclear Regulatory Commission
One White Flint North MS 12G13
11555 Rockville Pike
Rockville, Maryland 20852-2738
ATTN: Marcus Voth

November 9, 2006

Enclosed please find the University of Maryland's response to the request for additional information regarding questions 58 - 64 of the Safety Analysis Report.

**I declare under penalty of perjury that the foregoing is true and correct.**
**Executed on November 9, 2006**

Alsheikhly

**[Signature] Mohamad Al-Sheikhly, Director**
**University of Maryland Training Reactor**
**License Number R-70, Docket number 50-166**

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NRR

Rec'd DCD 5/27/10

## 13.0 ACCIDENT ANALYSIS

58. Section 13.2.1, Maximum Hypothetical Accident, page 13-4. The analysis only provides dose consequences for downwind locations (unrestricted areas). What is the projected dose for facility staff in the reactor bay (restricted area)? Doses in the unrestricted areas should be given for the maximum exposed person, the nearest residence, and other locations of interest such as the nearest dormitory.

*Response:*
*See LOCADOSE.xls as attached to previous section 13 submission.*

59. Section 13.2.2.3, Insertion of Fuel, page 13-5. Is there a reference for the calculated positive reactivity of 4.70$ from the insertion of a four-fuel element cluster into the most central location of the reactor core?

*Response:*
*Recent MCNP5 calculations indicate that the most reactive bundle is in fact worth $5.70. See attached.*

60. Section 13.2.2.3, Insertion of Fuel, page 13-6. The excess reactivity of MUTR is approximately 3.50$. Why does the insertion of a central fuel element cluster, with all control rods withdrawn, result in a reactivity addition of only 2.50$?

*Response:*
*The $2.50 is a conservative estimation and is based on a total reactivity worth of all control rods being approximately $8.00. MCNP calculations show that the most reactive bundle is in location* [redacted] *of the grid plate and has a worth of $5.70, of the $8.00, $6.70 is required to reach low power critical on a cold core, leaving $1.30, with a $1.00 experiment which is the most a single experiment is allowed, the pulse would maximize at $2.30.*

61. Section 13.2.2.3, Insertion of Fuel, page 13-7. Table 13.7 gives the calculated peak fuel temperatures for a 3.70$ reactivity pulse, at initial powers of 0.01 kW and 250 kW respectively. What is the basis for choosing a pulse of 3.70$? What is the location where a fuel cluster is added that results in a 3.70$ excess reactivity? Does this analysis form the technical basis for limiting the excess reactivity to 3.50$?

*Response:*
*The $3.70 is an extremely conservative estimation and is based on a total reactivity worth of all control rods being approximately $8.00. MCNP calculations show that the most reactive bundle is in location* [redacted] *of the grid plate and has a worth of $5.70, of the $8.00, $6.70 is required to reach low power critical on a cold core, leaving $1.30, with a $1.00 experiment which is the most a single experiment is allowed, the pulse would maximize at $2.30.*

62. Section 13.2.3, Loss of Coolant, page 13-7. The discussion on a loss of coolant accident (LOCA) noted that audible signals in the main reactor room, or on the west balcony, would warn persons entering those areas of high radiation conditions. When the building is unoccupied how would the high radiation condition be communicated to emergency response personnel? Is there an outside alarm to alert people to keep away from the facility? What is the projected dose for a person standing outside the reactor building? Please provide a copy of your calculations showing the dose rates from the LOCA. What are dose rates immediately following uncovering of the core?

[redacted]

63. Section 13.2.4.1, Fission Product Inventory, page 13-8. How does the fission product inventory listed in Table 13.8 compare with the source terms assumed for the Maximum Hypothetical Accident? Please calculate the fission product inventory for your fuel element.

*Response: These conversions are based on a conversion factor of* [redacted] *which is derived by dividing the NUREG inventory by a factor of four to account for the ratio of the power levels from the NUREG core to the Maryland core and a factor of* [redacted] *to account for the ratio of* [redacted] *versus 50 fuel elements as the basis for the calculations. This results in a combined ratio of* [redacted].

| | *NUREG/CR2387* | *University of Maryland* | *Conversion* | |
|---|---|---|---|---|
| Power kW | 1000 | 250 | 4 | Power Conversion |
| Fuel elements | 50 | [redacted] | [redacted] | Element Conversion |
| | | Combined | [redacted] | Combined Conversion |

| From NUREG/CR-2387 | | | | University of Maryland Maximum Inventory | |
|---|---|---|---|---|---|
| Isotope | $T_{1/2}$ | Unit | Activity (Ci) | Isotope | [redacted] |
| $^{83m}$Kr | 1.9 | h | 1.24E+02 | $^{83m}$Kr | [redacted] |
| $^{85m}$Kr | 4.4 | h | 2.87E+02 | $^{85m}$Kr | [redacted] |
| $^{85}$Kr | 10.8 | y | 4.80E+00 | $^{85}$Kr | [redacted] |
| $^{87}$Kr | 1.3 | h | 5.52E+02 | $^{87}$Kr | [redacted] |
| $^{88}$Kr | 2.8 | h | 7.89E+02 | $^{88}$Kr | [redacted] |
| $^{89}$Kr | 3.2 | m | 9.70E+02 | $^{89}$Kr | [redacted] |
| $^{90}$Kr | 32 | sec | 1.10E+03 | $^{90}$Kr | [redacted] |
| $^{89}$ Sr | 52.7 | d | 1.00E+03 | $^{89}$ Sr | [redacted] |
| $^{90}$ Sr | 27.7 | y | 3.10E+01 | $^{90}$ Sr | [redacted] |
| $^{91}$ Sr | 9.7 | h | 1.29E+03 | $^{91}$ Sr | [redacted] |
| $^{92}$ Sr | 2.7 | h | 1.46E+03 | $^{92}$ Sr | [redacted] |
| $^{93}$ Sr | 8.3 | m | 1.66E+03 | $^{93}$ Sr | [redacted] |
| $^{133m}$Xe | 2.3 | d | 3.90E+01 | $^{133m}$Xe | [redacted] |
| $^{133}$Xe | 5.3 | d | 2.27E+03 | $^{133}$Xe | [redacted] |
| $^{135m}$Xe | 0.3 | h | 5.98E+02 | $^{135m}$Xe | [redacted] |
| $^{135}$Xe | 9.1 | h | 1.03E+03 | $^{135}$Xe | [redacted] |
| $^{134m}$ Cs | 2.1 | y | 1.80E+00 | $^{134m}$ Cs | [redacted] |
| $^{134}$ Cs | 2.9 | h | 3.00E+00 | $^{134}$ Cs | [redacted] |
| $^{136}$ Cs | 13.7 | d | 2.60E+01 | $^{136}$ Cs | [redacted] |
| $^{137}$ Cs | 30 | y | 4.96E+02 | $^{137}$ Cs | [redacted] |
| $^{138}$ Cs | 32.2 | m | 2.06E+03 | $^{138}$ Cs | [redacted] |
| $^{131}$I | 8.1 | d | 1.08E+03 | $^{131}$I | [redacted] |
| $^{132}$I | 2.3 | h | 1.66E+03 | $^{132}$I | [redacted] |
| $^{133}$I | 20.3 | h | 1.93E+03 | $^{133}$I | [redacted] |
| $^{134}$I | 0.9 | h | 2.54E+03 | $^{134}$I | [redacted] |
| $^{135}$I | 6.7 | h | 2.21E+03 | $^{135}$I | [redacted] |

64. Section 13.2.4.2, Contamination of the Pool Water with Radioactivity, page 13-8. Is there a reference for the maximum water activity of $6.687 \times 10^{-4}$ mCi/ml in a fuel cladding failure?

*Response: These conversions are based on a conversion factor of* [redacted] *which is derived by dividing the NUREG inventory by a factor of four to account for the ratio of the power levels from the NUREG core to the Maryland core and a factor of* [redacted] *to account for the ratio of* [redacted] *versus 50 fuel elements as the basis for the calculations. This results in a combined ratio of* [redacted]. *The resultant numbers were converted to mCi and dispersed into the 6000 gallon capacity of the tank.*

| From NUREG/CR-2387 | | | [redacted] |
|---|---|---|---|
| Isotope | Activity (Ci) | Isotope | |
| $^{83m}$Kr | 1.24E+02 | $^{83m}$Kr | |
| $^{85m}$Kr | 2.87E+02 | $^{85m}$Kr | |
| $^{85}$Kr | 4.80E+00 | $^{85}$Kr | |
| $^{87}$Kr | 5.52E+02 | $^{87}$Kr | |
| $^{88}$Kr | 7.89E+02 | $^{88}$Kr | |
| $^{89}$Kr | 9.70E+02 | $^{89}$Kr | |
| $^{90}$Kr | 1.10E+03 | $^{90}$Kr | |
| $^{89}$Sr | 1.00E+03 | $^{89}$Sr | |
| $^{90}$Sr | 3.10E+01 | $^{90}$Sr | |
| $^{91}$Sr | 1.29E+03 | $^{91}$Sr | |
| $^{92}$Sr | 1.46E+03 | $^{92}$Sr | |
| $^{93}$Sr | 1.66E+03 | $^{93}$Sr | |
| $^{133m}$Xe | 3.90E+01 | $^{133m}$Xe | |
| $^{133}$Xe | 2.27E+03 | $^{133}$Xe | |
| $^{135m}$Xe | 5.98E+02 | $^{135m}$Xe | |
| $^{135}$Xe | 1.03E+03 | $^{135}$Xe | |
| $^{134m}$Cs | 1.80E+00 | $^{134m}$Cs | |
| $^{134}$Cs | 3.00E+00 | $^{134}$Cs | |
| $^{136}$Cs | 2.60E+01 | $^{136}$Cs | |
| $^{137}$Cs | 4.96E+02 | $^{137}$Cs | |
| $^{138}$Cs | 2.06E+03 | $^{138}$Cs | |
| $^{131}$I | 1.08E+03 | $^{131}$I | |
| $^{132}$I | 1.66E+03 | $^{132}$I | |
| $^{133}$I | 1.93E+03 | $^{133}$I | |
| $^{134}$I | 2.54E+03 | $^{134}$I | |
| $^{135}$I | 2.21E+03 | $^{135}$I | |

[redacted]

[redacted]

65. Section 13.3, Summary and Conclusions, page 13-9: The conclusion of Chapter 13 contains a statement that if the ventilation system were to function as designed, actual doses would be significantly reduced. Please discuss further.

*Response:*
*These scenarios were run assuming failure of confinement (that is the exhaust fans are running). If the exhaust fans are off, then the external dose is limited to what radiation penetrates the building plus what ever isotopes leak out of the non-air tight building (a far smaller amount than with the fans running).*

# Maryland Reactor Fuel Bundle Reactivity Worth

E. Burgett, D. Blaylock, N. Hertel, V. Adams, M. Al-Sheikhly

The reactivity worth of several fuel bundles in the University of Maryland Test Reactor (MUTR) has been investigated. A computer model of the reactor in full detail including fuel, control rods, support structures beam ports etc. was designed in the Monte Carlo N-Particle version 5 (MCNP5) neutral particle transport code. This highly benchmarked code developed by Los Alamos National Lab is a well accepted tool in the reactor engineering field. The reactivates of select fuel bundles being fully withdrawn and rapidly inserted will be described.

## MCNP5 Model

The MCNP5 model was created using a repeating fill universe, allowing for easy manipulation of the fuel bundle placements. The model was created using data from the operator log book. Due to unknown fuel enrichments and unknown fission product inventories at the this point in time in the reactor, the log book was used to benchmark the model. Using the critical rod heights described in an experiment on 27-July-05 for Xin Zhang's irradiation of cell cultures[*], the fission product inventory was adjusted to obtain a cold critical system. The general fission products for U-235 were used in the model[†]. Once a "just critical" system was created, the fuel rods were withdrawn to their upper limit specified in the UMTR tech specs[‡]. Fully removing the fuel element, MCNP5 was then allowed to run until a standard deviation of +/- 0.0009 or less was obtained. This procedure was repeated for several fuel elements of interest. Then, all of the fuel bundles were inserted to obtain a final $k_{eff}$ for the reactor system after the fuel bundle would be inserted. To further investigate this system, the reactor model was re-run for all cases to determine the reactivity worth of each of the desired fuel elements as if the reactor had come up to maximum operating temperature specified in the SAR[§].

## Results

The results of the MCNP5 Model are listed below. It can be noted the most reactive fuel bundle is located in positio[redacted] with a reactivity of $5.71. For the situation of a hot core, a maximum reactivity is once again obtained on bundle [redacted] with a lower $5.69. A table of the investigated fuel elements in descending reactivity can be seen in the second column of Table 1. A table depicting the reactivity of the hot fuel bundles can be seen in the third column of Table 1. Reactivity in dollars and cents was calculated using Equation 1 and 2. Equation 1 shows the derivation of rho also known as delta K over k (Δk/k). The dollar is defined in Equation 2. The various fuel locations can be seen in Figure 1.

$$\frac{k_j - k_i}{k_j} = \frac{\Delta k}{k} = \rho$$

**Equation 1**

$$Dollar(\$) = \frac{\rho}{\beta}$$

**Equation 2**

Here, the value of beta is defined as 0.007 as defined by the MUTR SAR.[**]

| [redacted] | Cold Fuel Reactivity ($) | Hot Fuel Reactivity ($) |
|---|---|---|
| | 5.714059078 | 5.689577985 |
| | 5.265036588 | 5.246142251 |
| | 5.11630674 | 5.094552399 |
| | 4.181575063 | 4.200597292 |
| | 3.374042817 | 3.356227651 |
| | 2.440585967 | 2.445271813 |
| | 2.429254169 | 2.395686348 |
| | 1.885327871 | 1.862996776 |
| | 1.311655604 | 1.28497192 |
| | 1.25358014 | 1.222635907 |
| | 0.181308766 | 0.165757128 |
| | 0.165727544 | 0.136005848 |

**Table 1**