October 10, 2002

Dr. Mohamad Al-Sheikhly, Director
Maryland University Training Reactor
University of Maryland
Department of Materials and Nuclear Engineering
Building 090
College Park, MD 20742-2115

SUBJECT: UNIVERSITY OF MARYLAND - REQUEST FOR ADDITIONAL INFORMATION RE: LICENSE RENEWAL (TAC NO. MB1788)

Dear Dr. Al-Sheikhly:

We are continuing our review of your license renewal request for Facility Operating License No. R-70 for the University of Maryland, Maryland University Training Reactor, which you submitted on May 12, 2000. During our review of your request, questions have arisen for which we require additional information and clarification. Please provide responses to the enclosed request for additional information within 180 days of the date of this letter. In accordance with 10 CFR 50.30(b), your response must be executed in a signed original under oath or affirmation. Following receipt of the additional information, we will continue our evaluation of your request.

If you have any questions regarding this review, please contact me at (301) 415-1127.

Sincerely,

/RA/

Alexander Adams, Jr., Senior Project Manager
Research and Test Reactors Section
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 50-166

Enclosure: As stated

cc w/enclosure: See next page

cc:

Director, Dept. of Natural Resources
Power Plant Siting Program
Energy & Coastal Zone Administration
Tawes State Office Building
Annapolis, MD 21401

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Maryland Department of Environment
201 West Preston Street
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Department of Materials and
Nuclear Engineering
University of Maryland
College Park, MD 20742

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REQUEST FOR ADDITIONAL INFORMATION
UNIVERSITY OF MARYLAND RESEARCH REACTOR
DOCKET NO. 50-166

Safety Analysis Report (SAR)

2.0 SITE CHARACTERISTICS

1. Section 2, Site Characteristics. Have significant changes described in Section 2.0 of the MUTR SAR (such as the peak daytime population increase from 20,000 to 45,000) been evaluated against the existing design and analyses to see whether they have any impact? Do these changes have any effect on the design and analyses presented in other chapters of the SAR, e.g., Chapter 3, "Design of Structures, Systems, and Components"; Chapter 11, "Radiation Protection Program and Waste Management"; and Chapter 13, "Accident Analyses"?

2. Section 2.2.2, Air Traffic, page 2-7. Some discussion is provided regarding the nearby small, single runway airport that is approximately 1.5 km from the MUTR, the types of planes that use the airport, and the relatively minor damage that would be expected if a small aircraft was to strike the MUTR. Discuss why this type of impact would not cause any significant damage to the pool tank and fuel.

3. Section 2.3.1, Meteorology - General and Local Climate, page 2-8. This section of the SAR should be updated to describe the recent tornado that hit this area in 2001 and to discuss the frequency and consequences of such a tornado on the MUTR.

4. Section 2.5.4, Vibratory Ground Motion, page 2-12. The magnitude of vibratory ground motion is presented in Section 2.5.4 of the MUTR SAR based on 1999 US Geological Survey (USGS) estimates. The value of 18% g at 0.5 second period in the SAR does not appear to match the information from the USGS (18% g at 0.2 second period). Please explain this difference and indicate whether the specified maximum earthquake potential/vibratory ground motion have been considered in the design or the basis for acceptance of these values.

5. Sections 2.0 and 3.0. These sections of the MUTR SAR contain statements which present staff conclusions such as "...the staff concludes that the meteorological conditions at the reactor site neither pose a significant risk of damage to the reactor nor render the site unacceptable for the facility." Please replace these statements with an analysis and basis of why you find the sections under discussion to be acceptable.

3.0 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

6. Section 3.1, Design Criteria, page 3-1. This section of the MUTR SAR indicates that the reactor building was designed and built to meet or exceed building codes existing at the time of construction. Please provide a summary of the codes, standards, and guides that were followed for structures, systems, and components that are required to ensure reactor facility safety and protection of the public.

7. Section 3.5, Systems and Components. This section does not provide the design bases for electro-mechanical systems and components that are required to function. Please provide a summary of the design criteria (codes/standards, loadings, operating environment, etc.).

4.0 REACTOR DESCRIPTION

8. Section 4.1, Summary Description. This section discusses water chemistry at MUTR. Have there been any water chemistry excursions at MUTR which could result in the corrosion or material degradation of the fuel elements or clusters?

9. Section 4.0, Reactor Description. This section does not discuss fuel inspection, though Section 9.2.4 (Fuel-Rod Inspection Tool) does. Are any routine inspections done of fuel element condition? If not, please justify.

10. Section 4.2.4, Neutron Startup Source. This section describes the upper grid plate. From the discussion provided, it is not clear what the upper grid plate is. Please clarify.

11. Section 4.2.4, Neutron Startup Source. This section also describes the startup source at MUTR. Is the source regenerative and, if not, what is the lifetime of the source? When depleted, is the source removed from site, or stored in the pool?

12. Section 4.2.2, Control Rods. Are any depleted control rods stored in the pool? If so, how are they stored, and how many are stored? If not, what is the process for disposition of the depleted control rods?

13. SAR Figure 4.15, Reactor Core Grid Plate. This figure illustrates three sets of holes. The larger one is for the Fuel Clusters. What are the purposes of the other holes? How is the grid plate supported above the floor of the tank?

14. Section 4.3, Reactor Tank. This section describes the design and function of the reactor pool tank. How would you detect leakage from the reactor tank? What is the minimum leakage rate that you can detect and what is the maximum time duration that this rate of leakage could occur before detection? What is the impact on the public health and safety of a pool leak?

15. Section 4.4, Biological Shield. This section states that the biological shield consists of concrete, steel, and reactor pool water and serves to protect personnel from over-exposure. Have there been any operational radiological exposures at MUTR attributed to inadequate, or degradation of the biological shield?

16. Section 4.5.1, Normal Operating Conditions. The MUTR can place experiments in the beam-tubes, the thermal column, and in-core. TS 3.7 limits each experiment individually to 1.00$ and the total of in-core experiments to 3.00$ (including flooding). It is not clear how many experiments MUTR can have in-core and how many outside. Please specify how many experiments may be conducted simultaneously. What is the total reactivity worth of all experiments (in-core and other)?

17. Section 4.5.2, Reactor Core Physics Parameters. The peak thermal/fast flux is measured in the pneumatic transfer tube. Was any other location analyzed or measured for its respective flux value?

18. Section 4.5.3, Operating Limits. Identify how the value 0.90$ was obtained when a hypothetical 3.50$ excess reactivity is present and the two control rods are introduced. Is there a common mode failure for multiple experiments that could introduce additional excess reactivity?

5.0 REACTOR COOLANT SYSTEMS

19. Section 5.2, Primary Coolant System, page 5-1. The MUTR is designed for natural convection cooling without forced flow. There is also a heat exchanger (HX) in a forced flow loop present in the primary coolant system, but it is not required for safe operation of the reactor and can be bypassed. The function of this system is to maintain the temperature and chemistry quality of the pool water. The HX is cooled by an open loop of city water which discharges into the city sewer system. The use and safety design of this system is not clear. Please discuss the following:

    a. When the reactor is operating, what is the normal mode of operation for the primary coolant system relative to the cleanup system and the HX? Is the operation of this system controlled by a plant operating procedure?

    b. How often is the HX used and on what conditions would the HX be used and bypassed?

    c. If there were a reactor coolant piping/component failure outside of the reactor pool, describe what would prevent the pool water from draining or limit the amount of water lost?

    d. If a primary to secondary leak were to develop in the HX, which way would the leakage flow, and how would your design prevent the escape of primary water into the city water system? Please consider this question with the primary pump both running and shut down. If you cannot show that pressure on the secondary side is higher than pressure on the primary side of the heat exchanger at all times, analyze the impact of a HX leak from a radiological standpoint. Is there any radiological monitoring of the discharge of the city water before it goes into the city sewer system?

20. Section 5.6, Nitrogen-16 Control System, page 5-4. In section 5.6 discussing the N-16 control system, it is stated that the outlet pipe is equipped with a siphon break to preclude a significant loss of primary coolant in the event of a piping failure outside of the pool tank. Figure 5.1 does not show the location of the siphon break. Indicate where the siphon break is located in this figure. Also, explain how this siphon break precludes a significant loss of pool water in the event of a piping failure outside the pool tank.

21. Section 5.7, Reactor Sump, page 5-7. Figure 5.4 presents the reactor sump water handling system. In this figure the well and the sump structure are shown as separate structures. Describe the physical connection between the well and the concrete sump pit. Does the spring check valve on the city water line in Figure 5.4 provide assurance that no backflow into the city water system occurs? If so, what testing or inspection is performed on this valve to ensure no degradation and that it is operating as designed?

22. Section 5.5, Primary Makeup Water System, page 5-4. Describe the method for adding water to the primary system. Is the system normally valved off by manual valves when water is not being added (or is there a physical break in the piping when not in use)? Figures 5.2 and 5.3 show a city water feed that does not have a check valve. Is this city water feed different from that city water feed shown in Figure 5.4? If yes, then provide assurance that no backflow into the city water system occurs.

6.0 ENGINEERED SAFETY FEATURES

23. Chapter 6.0, Engineered Safety Features. The SAR states there are six external entrances into the confinement. Technical Specification 4.4, Confinement, states that prior to reactor startup, isolation of these doors is visually verified. Once verified and the reactor is made critical, are these entrances alarmed in some way to alert the operator that confinement is unsecured if someone opens one inadvertently, or are they locked during the visual? Disposition of these doors is not specified in the SAR.

7.0 I&C SYSTEMS

24. Section 7.2, Design of Instrumentation and Control Systems. This section describes a system performance/reliability analysis for the bistable trips, console scram, and rod control circuits. Have there been any notable problems in the operating experience for these systems that would call into question the analytical results?

25. Section 7.4, Reactor Protection System. As detailed in this section, the reactor protection system provides a number of redundant and diverse inputs into the scram logic. Please provide a description as to the separation/isolation these various inputs are afforded throughout the reactor facility.

26. Sections 7.3, Reactor Control System. In the manual mode, rod up movement is interlocked such that only one rod can be raised at a time. Is it possible to raise one rod and inadvertently lower another simultaneously, and if so, could this present a problem?

27. Section 7.3.3, RCS Interlocks. Please clarify the first sentence in paragraph 5, where it states that "the regulating rod motor leads are <u>removed</u> from manual operation and <u>connected</u> to the output of the servo-amplifier." Is this just the turning of a switch or a more involved procedure?

28. Section 7.7, Radiation Monitoring Systems. This section has a discussion regarding the multiple roles of the area radiation monitors (i.e., reactor protection, engineered safety features actuation, and health physics protection for the facility inhabitants). Upon the loss of the normal electrical power supply, RPS and ESF actuation is automatically

initiated. However, it appears that the protection function for the facility inhabitants is lost. Please provide comments as to the acceptability of this situation.

8.0 ELECTRICAL POWER SYSTEMS

29. Section 8.0, Electrical Power Systems. Describe how the separation of electrical power cables and those cables associated with the experiments is accomplished in order to prevent any electromagnetic interference with the reactor protection instrumentation.

30. Section 8.0, Electrical Power Systems. Describe any needs for electrical power that may be required for placing/maintaining experimental equipment in a safe condition.

9.0 AUXILIARY SYSTEMS

31. Chapter 9, Auxiliary Systems and Chapter 11, Radiation Protection Program and Waste Management. There is no mention of the reactor building floor drain system in either of these SAR chapters. Decontamination activities and minor valve and fitting leakages of reactor coolant would be potentially contaminated. Are the floor drains all routed to the sump located in the water handling room or are they routed to the sanitary system? Please discuss.

32. Section 9.2.3, Fuel-Rod Transfer Cask. The SAR states that the fuel rod transfer cask weighs approximately 5700 lbs. Provide a discussion on the procedures, equipment, and lifting capacities associated with this load handling at MUTR.

33. Section 9.3, Fire Protection. This section states that a water sprinkler system provides the fire suppression in the reactor building. Is the drain system sized to handle the water volume created by actuation of the sprinkler system to prevent both migration of radiological contamination outside the building and potential flooding of the water handling room?

34. Section 9.3, Fire Protection. If the sprinkler system actuates when the building is unoccupied, how will the fire department or security be made aware of it?

35. Section 9.4, Communication System. Is the audible evacuation alarm separate from the intercom system? If it is, please provide a brief description of the system design.

36. Figure 9.5, page 9-8. The locations of the fire alarm pull boxes and speaker intercom locations are not clear. Figure 9.5 should be revised, with the locations of the fire alarm pull boxes and intercom speakers clearly identified.

37. Section 9.5, Possession and Use of Byproduct, Source, and Special Nuclear Material. Please confirm that you want to maintain the current license special nuclear material and byproduct material limits in your renewed license.

10.0 EXPERIMENTAL FACILITIES AND UTILIZATION

38. Section 10.2, Experimental Facilities. Provide a more detailed description of the functional design of the Thermal Column, Beam Ports and Through Tube experimental facilities. For example, from the description provided it is not clear if these facilities require an air exhaust system or if beam tubes must be filled with demineralized water to provide shielding. Additionally, Figures 10.3, “Beam Tube,” 10.4, “Beam Tube Plugs,” and 10.5, “Through Tube” are not legible (apparently due to their reduced size). Please provide clearer illustrations that are of a larger size. Enlarging each figure to 8½ by 11 inches should be sufficient.

39. Section 10.2.4, Pneumatic Transfer System. Provide a more detailed description of the pneumatic transfer system design and operation and the administrative controls governing its use. Specific topics to be addressed include the source(s) of CO_2, potential consequences of a stuck/immovable rabbit assembly and design features and/or administrative controls provided to preclude or mitigate this occurrence, manual and automatic timing modes of operation, system venting, and design features which preclude the potential for a failure within this system to result in a loss of pool water inventory.

40. Section 10.2.5, Other Locations. This section states that the reactor grid plate and reactor pool tank may be utilized to conduct experiments. Provide a more detailed description of the functional design of these facilities, the type of experiments that are typically conducted at these locations, and if any special precautions or limitations are needed to ensure their safe use.

42. Section 10.3, Experiment Review. Regulatory Position C.3.a(1) of Regulatory Guide 2.2 states that no experiment should be performed without review and approval by a technically competent Safety Review Group or Committee. From the discussion presented in Section 10.3 of the SAR and Technical Specification 6.5, it is not clear if all experiments categorized as “routine” have been previously reviewed and approved by RSC. Please clarify.

43. Section 10.3, Experiment Review. Please clarify in the SAR that all new (modified routine and special) experiments will go through a 50.59 review.

44. Section 10.3, Experiment Review. This section notes that quantities of TNT less than 25 mg can be irradiated but calculations must show that the pressure produced (if detonation occurs) is less than the failure pressure of the container. The overpressure from detonation of 25 mg of TNT (from Regulatory Guide 1.91, Rev. 0) in small containers can be significant (e.g., > 1000 psi for a 2" container and 250 psi for a 4" container). For larger containers the overpressure is not significant. Please discuss any administrative controls you have to ensure that the necessary calculations are performed and performed correctly, if TNT were to be irradiated.

11.0 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

45. Section 11.1.2.5, Principal User. This section defines a principal user and lists their responsibilities. Please specify who the principal users are, as related to the operation and experimental programs at the MUTR.

46. Section 11.1.4, Radiation Monitoring and Surveying. Please provide calculations to show that doses to the reactor staff and members of the public from the production of normal gaseous effluents from reactor operations is acceptable. The calculations should be based on continuous reactor operation and consider both argon-41 and nitrogen-16. Doses should be determined for staff members, the maximum exposed member of the public, at the closest residence to the reactor, and at any other points of special interest (e.g., dormitories).

47. Section 11.1.4, Radiation Monitoring and Surveying. Are there any environmental radiation measurements taken outside of the reactor room (e.g., TLDs posted outside of the reactor building or samples taken of vegetation, water or soil from the environment)?

48. Section 11.1.4, Radiation Monitoring and Surveying. The second to last paragraph states, “Both monitors are capable of sending scram signals to the reactor as well as secure the ventilation system in the event of a high reading.” Please describe the basis for the set points of the radiation monitoring system. Is this an automatic scram signal (Chapter 7 alludes to it being automatic, but does not explain how it is incorporated into trip logic)? Or is it just an alarm to the operators so that they can manually scram the reactor? If it does automatically scram the reactor, provide details as to how it is accomplished electronically. If it does not automatically scram the reactor, Section 11.1.4 should be clarified.

49. Section 11.1.4, Radiation Monitoring and Surveying. Some discussions of the radiation monitors appears in SAR Sections 6, 7.7, and 11.1.4, and in TS 3.6 and 4.6. They do not provide a consistent picture of the system. Please coordinate the discussions in order to clarify.

50. Section 11.1.5.3, Expected Exposures and Dosimetry. This section discusses expected exposures and dosimetry associated with the facility. Please provide a summary of actual staff radiation exposure at the reactor facility over the past five years similar to the information given in Table 12.1 of NUREG-1043, “Safety Evaluation Report Related to the Renewal of the Operating License for the Training and Research Reactor at the University of Maryland,” dated March 1984.

12.0 CONDUCT OF OPERATIONS

51. Section 12.1.1, Structure. Clarify that the Nuclear Reactor Director shown in the organization charts and in the SAR is the Facility Director used throughout the TS.

52. Section 12.1, Organization. There is no discussion in the SAR of radiation protection worker staffing, qualification, or training. Please provide.

53. Section 12.2.2, Charter and Rules. It is not clear whether the RSC has a formal charter, including the items of Section 6.2.2 of ANS 15.1. Please clarify.

13.0 ACCIDENT ANALYSIS

54. Section 13.2.1, Maximum Hypothetical Accident, page 13-2. What is the reference for the isotopic loading in one fuel element of the MUTR after an infinite operation at 250 kW?

55. Section 13.2.1, Maximum Hypothetical Accident, page 13-3. What is the basis for assuming a value of 0.01 for the atmospheric dispersion factor (x/Q)? What are the release pathways to the environment for the HMA? If the release point is elevated, has the possibility been examined that the highest dose may be from overhead cloud shine instead of cloud immersion?

56. Section 13.2.1, Maximum Hypothetical Accident, page 13-4. What is the basis for assuming a release fraction of $1x10^{-6}$ for cesium and strontium?

57. Section 13.2.1, Maximum Hypothetical Accident, page 13-4. Are the fission product activities, listed in Tables 13.1 to 13.3, derived from NUREG/CR-2387?

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.

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?

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

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$?

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?

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.

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 6.687×10^{-4} mCi/ml in a fuel cladding failure?

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.

15.0 FINANCIAL QUALIFICATIONS

66. Please submit a copy of the latest financial statements of the University.

16.0 PRIOR USE OF REACTOR COMPONENTS

67. Section 16, Prior Use of Reactor Components. For reactor building and biological structural elements (including steel, concrete, aluminum liner, foundations, and equipment supports), summarize the operating experience relating to degradation and/or any malfunctions. Have there been any inspection/examination of the conditions of these items? If so, what are the results (e.g., items examined, aging mechanisms such as water infiltration, cracking, corrosion, etc., and any required repairs/replacements)? Section 16.1 of the MUTR SAR indicates that the reactor building (structure, potable water systems, non-reactor control electrical systems, HVAC systems, and fire protection systems) is maintained by the “campus.” What has been the maintenance experience with these items? No discussion of reactor building maintenance was noted in the 1999-2000 Annual Report. Would maintenance on these items be reported in the TS required Annual Report for the MUTR?

68. Section 16.2, Biological Shield. This section describes the design and safety functions of the biological shield. Has the reactor tank ever overflowed? If so, this could be indicative of future corrosion.

69. Section 16.3, Reactor Fuel. This section states that the fuel at MUTR was fabricated 28 years ago. Have you considered fuel lifetime in your decision to use the fuel for the next 20 years? Discuss potential fuel degradation due to radiation, gas pressure build-up internal to the cladding, and erosion of cladding and why these phenomenon will not be a concern.

70. Section 16.4, Reactor Control Systems. This section states that reactor control system “age-related failures” have occurred, and that it is likely such failures will continue to occur. Please describe the specific age-related failures which have occurred, the consequence of these failures, and how these occurrences were detected? What changes, if any, were made to procedures in an attempt to detect such age-related degradation before components fail?

71. Section 16, Prior Use of Reactor Components. The 1999-2000 Annual Operating Report for MUTR indicates that several CRDM's were replaced with new and refurbished drives. What necessitated these changes?

ENVIRONMENTAL REPORT

72. Section V of the 1999-2000 Annual Operating Report states that continuous monitoring for the year was accomplished using fixed-mounted film badges throughout the interior of the reactor building. Facility Technical Specification 3.6.4 specifies that the campus radiation safety organization maintain an environmental monitor at the site boundary as well. Explain how compliance is demonstrated, and if any abnormal radiation levels were ever detected.

73. Discuss actual releases of airborne, liquid and solid waste from the facility for the past 10 years and if these trends are expected to continue in the future.

TECHNICAL SPECIFICATIONS

74. TS 1.1, ALARA. Your definition differs from that given in 10 CFR Part 20. Please address.

75. TS 1.24, REACTOR SECURED. The definition you have used from ANS 15.1 is generic. Please modify this definition to make it specific to your facility (e.g., in TS 1.24.2.a state the minimum number of control rods needed in the full down position).

76. TS 1.31, SCRAM TIME. The definition you have used from ANS 15.1 is generic. Please modify this definition to make it specific to your facility (see your current TSs).

77. TS 2.1, SAFETY LIMIT. By stipulating the safety limit for the fuel fully immersed in water, are you ensuring that the cladding temperature will be less than 500°C at all times? Please give a more detailed explanation and specific references to support your proposed safety limit.

78. TS 2.2, LIMITING SAFETY SYSTEM SETTING. Please provide the calculations referenced in the basis for this TS that shows that the LSSS is sufficient to protect the SL with the instrumented fuel element at any position in the reactor core and the calculations that support the statement that sufficient margin is present to account for uncertainty in the accuracy of the fuel temperature measurement channel and any overshoot in reactor power resulting from a reactor transient during steady state mode operation. Section 4.5.3 of the SAR discusses a LSSS of 175 °C (however, Table 3.1 contains a scram set-point of 175 °C) while TS 2.2 has a value of 350 °C. Please explain the difference in the values.

79. TS 3.1.3.b, REACTOR CORE PARAMETERS. Your TS uses the terms "fuel elements" and "fuel bundles." Please define a fuel bundle. Is fuel normally handled as elements or bundles? If fuel is handled in bundles, explain how the reactor will remain sub-critical if the core is sub-critical by the worth of the most reactive fuel element and a fuel bundle is added to the reactor.

80. TS 3.1.3.c, REACTOR CORE PARAMETERS. Please provide a calculation that shows that the reactor will remain sub-critical if the most reactive control rod is removed from the core if the four least reactive fuel bundles are removed. Would a requirement that enough fuel bundles are to be removed from the core prior to control rod removal such that the reactor remains at some minimal sub-critical level after removal of the control rod be simpler?

81. TS 3.1.4, REACTOR CORE PARAMETERS. Please define what constitutes damaged fuel.

82. TS 3.1.5, REACTOR CORE PARAMETERS. Should these values be stated as less than or greater than rather than single values? Also consider moving this TS to Section 5 because these are design criteria rather than LCOs.

83. TS 3.2.1 and 4.2.3, REACTOR CONTROL AND SAFETY SYSTEMS. Are the results of the control rod drop time tests trended to detect any indication of degradation prior to the time limit being exceeded? If so, please discuss any trends seen.

84. TS 3.2.2, REACTOR CONTROL AND SAFETY SYSTEMS. Please discuss the maximum power ramp that would result from adding $0.30 per second of reactivity to the reactor, starting from a low power condition. Also, discuss the reactor safety system response to the reactivity addition, including power overshoot.

85. TS 3.2.4, REACTOR CONTROL AND SAFETY SYSTEMS. Technical Specification 3.2.4 states: "*The safety interlocks shall be operable in accordance with Table 3.2, including the minimum number of interlocks*." With regard to experimental facilities, Table 3.2 describes the Plug Electrical Connection interlock as a means of disabling magnet power when the Beam Port or Through Tube plug is removed. Table 3.4 states that the purpose of this interlock is to assure that the reactor cannot be operated with Beam Port or Through Tube plugs removed without further precautions. Technical Specification 3.2.5 states: *"The Beam Port and Through Tube Interlocks may be bypassed during a reactor operation with permission of the Reactor Director*." The Basis for this specification (Basis 5) states that this "*ensures that the reactor interlocks will always serve their intended purpose*." This basis does not appear correct, since the intent of TS 3.2.5 is to bypass the interlock not ensure it serves its purpose. Please clarify. Also, describe the circumstances under which the Beam Port and Through Tube interlocks would be bypassed and the precautions that are implemented when this interlock is bypassed.

86. TS Table 3.1, REACTOR SAFETY CHANNELS: SCRAM CHANNELS. This Table does not include the reactor period scram function. Thus, only 9 of the 10 reactor scram functions are addressed by the TS. Please provide your basis for the exclusion of the period scram from this table.

87. TS Table 3.2, REACTOR SAFETY CHANNELS: INTERLOCKS. It is not clear what interlock is provided by the log power channel. Please clarify. TSs usually contain an additional table which lists required minimum measuring channels. For example, there is a requirement in the TS for two reactor power level scrams. However, these scrams

originate in different measuring channels. Please consider adding this additional table to the TS.

88. TS 3.3.3, COOLANT SYSTEMS. This TS, as written, is a surveillance requirement and should be in Section 4.3 of the TS (it is partially in as 4.3.1 now). TS 3.3.3 in this section should contain the acceptable limits of the measurements/samples/analyses.

89. TS 3.3.4, COOLANT SYSTEMS. The last sentence of this TS is a surveillance requirement and should be in Section 4.3 of the TS (it is partially in as 4.3.2 now).

90. TS 3.3, COOLANT SYSTEMS. Is there any limitation on the bulk temperature of the reactor coolant?

91. TS 3.4, CONFINEMENT. Your proposed TS appears to be design features that should be in Section 5 of the TS. This TS should discuss under what conditions confinement is needed (e.g., reactor operation, fuel movement, radioactive materials handling, etc.) and what constitutes confinement being established.

92. TS 3.5, VENTILATION SYSTEMS. Are there any minimum ventilation performance requirements, such as minimum fan flow rates that must be met by the ventilation system to maintain confinement and meet the objective of TS 3.4. If so, they should be stated in this TS and verified in TS 4.5.

93. TS 3.6, RADIATION MONITORING SYSTEM. While it is acceptable for the actual alarm set points to be in a procedure because they can change with changes in such parameters as detector efficiency, the bases for the set points should be given in the specification of the TS. Please include this information in the specification of TS 3.6. Also, the current Bases of TS 3.6 refer to TS 3.3.6, which is missing. Please clarify.

94. TS Table 3.5, MINIMUM RADIATION MONITORING CHANNELS. In the “Minimum Number Operable” column, the placement of the wording makes it unclear as which monitors it applies. Please clarify. Also, if it is intended to say that you only need one monitor overall, please justify.

95. TS 3.7, LIMITATIONS ON EXPERIMENTS. Please explain the difference between TS 3.7(1) and 3.7(2), and the need for both. In TS 3.7(3) should the limitation be on the absolute worth of the sum of experiments? Are potentially explosive materials discussed in TS 3.7(4) also subject to the requirements of TS 3.7(5)? TS 3.7(6)(a) and (b) are standard TS for experiment failure. However, you have stated two of the four standard requirements (see page 28 of Appendix 14.1 of NUREG-1537, Part 1). Please explain why the other two standard requirements are not applicable to your experimental program. The basis for TS 3.7(7) refers to an analysis in the SAR. Please provide the analysis.

96. TS 4.0, SURVEILLANCE REQUIREMENTS. There is usually an introduction to this section that defines the standard surveillance intervals. This introduction also may specify that certain surveillance requirements may be postponed during reactor shutdown and performed before the reactor is restarted or as soon as practicable after

reactor start up if reactor operation is needed to perform the surveillance. For example, if the reactor is not in operation, it may not be necessary to calibrate the control rods until the reactor is restarted. Further, some surveillances may become due during a period of extended operation, and the performance of the surveillance may need to be postponed until the reactor is shut down. You would need to determine what surveillances can be postponed and provide a justification. Please address.

97. TS 4.1, REACTOR CORE PARAMETERS. Consider adding to this TS a requirement to measure the excess reactivity and shutdown margin after changes in control rods and experiments that exceed the value of the minimum shutdown margin. Also, how do you ensure that TS 3.1(3)(a) is met?

98. TS 4.2.4, REACTOR CONTROL AND SAFETY SYSTEMS. Please justify the need not to do a channel test following a reactor shutdown of less than 24 hours.

99. TS 4.2.7 and 8, REACTOR CONTROL AND SAFETY SYSTEMS. Are the TS required inspections of the control rods and control rod drive mechanisms performed to procedures to ensure adequate and consistent inspections?

100. TS 4.5.1, VENTILATION SYSTEM. TS 4.5.1 appears to be a LCO which is given in Section 3, or a design feature which is given in Section 5, because it does not contain a surveillance requirement. Please address and add surveillance requirements if needed.

101. TS 4.6.2.1, EFFLUENTS. Please provide additional discussion about these air samples. How are they taken? What are the limits? Is there a requirement in the TS for the samples to be taken?

102. TS 5.4.2, FISSIONABLE MATERIAL STORAGE. This applies to fuel storage when not in the reactor core. What monitoring system, if any, is used for this pit to detect criticality, fuel temperature, etc?

103. TS Figure 6.1 and 6.2. Please clarify the meaning of solid and dotted lines on the structure diagrams. The solid line shown on Figure 6.2 between the Chairman of the Department of Materials and Nuclear Engineering and the Reactor Safety Committee is not on Figure 6.1 (similar comments on SAR Figures 12.1 and 12.2). Please explain.

104. TS 6.1.3.1, FACILITY STAFF REQUIREMENTS. The TS differs from the corresponding area of the SAR (12.1.3). The TS wording is less conservative than the SAR wording. SAR Section 2.1.3 specifies staffing for when the reactor is "not secured," which includes both operation and shutdown; while the TS specifies staffing only for when the reactor "is operating." ANS 15.1 agrees with the SAR wording rather than the TS. It appears that the TS staffing should apply for both operating and shutdown conditions (i.e., not secured). This would then agree with the SAR and the ANS standard.

105. TS 6.1.3.3.d, FACILITY STAFF REQUIREMENTS. This TS requires supervision by an SRO on "Resumption of operation following an unscheduled shut down. (This requirement is waived if the shutdown is initiated by an interruption of electrical power to the plant.)" This provision is included to meet the requirements of 10 CFR 50.54

(m)(1). However, 50.54 does not contain the waiver noted in the MUTR TS. ANS 15.1 does not include this waiver either. Please justify the need for the waiver and why an SRO is not required in this case. Alternatively revise the proposed TS to comply with the regulations. ANS 15.1 also requires the presence of an SRO during recovery from an unplanned or unscheduled significant power reduction. This is not included in your TS. Please add or justify the omission.

106. TS 6.1.4 SELECTION AND TRAINING OF PERSONNEL. Your TS differs from ANS-15.1. Please discuss. Also, the requalification program is a stand alone program and need not be referenced in the TS.

107. TS 6.2.3, REACTOR SAFETY COMMITTEE REVIEW FUNCTION AND SAR SECTION 12.2.3, REVIEW FUNCTION. There are a number of items, specified for review by the safety review committee in ANS 15.1, that are not included in the responsibility of the review committee (RSC) or are significantly different from the items given in ANS-15.1. Some examples are: (1) all new procedures and major revisions thereto having safety significance, (2) proposed changes to reactor facility equipment, or systems having safety significance, (3) new experiments that could affect reactivity or result in the release of radioactivity, and (4) violations of internal procedures or instructions having safety significance. Please modify the committee review functions to match those in ANS-15.1 or justify your proposed differences.

108. TS 6.2.4, REACTOR SAFETY COMMITTEE AUDIT Function. Two areas noted in ANS 15.1 for inclusion in the TS on the audit function were not in the MUTR TS, specifically: (1) results of actions taken to correct deficiencies in reactor facility equipment, systems, structures or methods of operations that affect reactor safety; and (2) the emergency plan and implementing procedures. Please justify or modify TS to include these items.

109. TS 6.4, OPERATING PROCEDURES. TS 6.4 addresses most of the required procedure types of ANS 15.1, but a few were not covered by the TS, specifically: administrative controls for conduct of irradiations and experiments that could affect reactor safety or core reactivity, implementation of the emergency and security plans, and personnel radiation protection (including commitment to ALARA per ANSI/ANS-15.11). Please justify the reason these are not addressed or add them to the TS.

110. TS 6.4, OPERATING PROCEDURES. NRC has determined that procedures are necessary for shipping, possession, and transfer of radioactive material. Please add this requirement to TS 6.4 or justify not needing these procedures.

111. TS 6.4, OPERATING PROCEDURES. ANS 15.1 recommends that substantive changes to previous procedures be made effective only after review by the RSC and appropriate approval. The MUTR TS and SAR do not require review by the RSC prior to implementing the change. Please justify this or add to the TS.

112. TS 6.4, OPERATING PROCEDURES. The SAR and the TS only address substantive changes to procedures. Is there a need for minor or temporary changes? If such activities are anticipated, then they would also need to be approved in the same manner

as substantive changes, unless a more streamlined method is documented and approved in the SAR.

113. TS 6.4, OPERATING PROCEDURES. Section 1.6 of the SAR notes that occasional irradiation work is performed at MUTR for local government and industry organizations. Clarify if any byproduct material is generated or used in these efforts. If so then procedures should be developed and added to the list in TS 6.4 governing this use of any byproduct material.

114. TS 6.4, OPERATING PROCEDURES. ANS 15.1 permits temporary deviations from procedures in special circumstances, but states that such deviations shall be documented and reported to management. The MUTR TS permit this in TS 6.4, but do not specify the documentation of such cases or the reporting to the Reactor Director. Please justify this omission or add it to the TS.

115. TS 6.6.1, ACTION TO BE TAKEN IN CASE OF SAFETY LIMIT VIOLATION. ANS 15.1, Section 6.6.1 requires that a safety limit violation be promptly reported to the Level 2 manager (facility director for MUTR). Please add this to TS 6.6.1 or justify not needing this reporting.

116. TS 6.6.2 and 6.7.1. It would help the operators at MUTR to reference Section 1.27 of the Technical Specifications in TS 6.6.2 and 6.7.2.1, since it is needed to implement these two specifications.

117. TS 6.7.2, SPECIAL REPORTS. NRC has changed administrative policy in a few areas related to this TS as follows. Provide a telephone report, confirmed in writing by fax (no telegraph), within 24 hours to the NRC operations center or the MUTR NRC project manager. Provide the 14 day written report to the NRC document control desk (no need for copies to director of NRR or Region I). Please revise the TS accordingly.

118. TS 6.8, RECORDS. Under the category of records to be kept for five years, the TS do not list audit reports as recommended by ANS 15.1. Please justify or modify TS.

119. TS 6.8, RECORDS. Under the category of lifetime records, the TS do not list either gaseous radioactive effluents released to the environment or offsite environmental monitoring surveys required by the TS, as recommended by ANS 15.1. We note that there were gaseous releases of Ar-41 reported in the Annual Report. Also we note that TS 3.6.4 requires environmental monitoring at the site boundary. Thus, these two items should be included in TS 6.8.3 as records that shall be retained for the lifetime of the facility. The regulations in 10 CFR 50.36 require records of violations of SL, LSSS, and LCOs to be retained for the life of the facility. Please modify your TSs or justify not making these changes.