UNIVERSITY OF MARYLAND

DOCKET NO. 50-166

RENEWED FACILITY OPERATING LICENSE

License No. R-70

1. The U.S. Nuclear Regulatory Commission ("the Commission") has found that:

   A. The application for renewal of Facility Operating License No. R-70 filed by the University of Maryland ("the licensee"), dated May 12, 2000, as supplemented by letters dated June 7, August 4, September 17, and October 7, 2004; April 18, 2005; April 25 (two letters), August 28 (two letters), September 7, November 9, and December 18, 2006; May 27, July 28, and September 22, 2010; January 31, February 2, May 2, July 5, July 29, September 26, September 28, and October 12, 2011; February 9, March 14, May 22, and August 29, 2012; March 21, 2013; April 10, June 18, and November 25, 2014 (two letters); July 1, November 23, and December 2, 2015; and January 5, February 29, November 1, November 2, November 10, November 17 (two letters), and December 2, 2016, ("the application"), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended ("the Act"), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

   B. Construction of the Maryland University Training Reactor ("the facility") TRIGA-type nuclear research reactor was completed in substantial conformity with Construction Permit No. CPRR-53 (dated June 29, 1960) and modified in conformity with CPRR-108 (dated March 25, 1970) and the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, as supplemented, the provisions of the Act, and the rules and regulations of the Commission;

   D. There is reasonable assurance that: (i) the activities authorized by this license can be conducted without endangering the health and safety of the public, and (ii) such activities will be conducted in compliance with the Commission's regulations;

   E. The licensee is technically and financially qualified to engage in the activities authorized by this license in accordance with the rules and regulations of the Commission;

   F. The applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," have been satisfied;

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 10 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. to 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; and

e. 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 other facilities in the form of TRIGA-type reactor fuel.

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.

b. to receive, possess, and use, but 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 TRIGA-type reactor fuel.

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. <u>Maximum Power Level</u>

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. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised by Amendment No. 10, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Physical Security Plan

The licensee shall maintain and fully implement all provisions of the Commission-approved physical security plan, including changes made pursuant to the authority of 10 CFR 50.54(p). The approved physical security plan, entitled “Security Plan for the University of Maryland Training Reactor February 10, 2016,” consists of documents withheld from public disclosure pursuant to 10 CFR 73.21, “Protection of Safeguards Information Performance Requirements.”

This license is effective as of the date of issuance and shall expire at midnight, 20 years from the date of issuance.

For the Nuclear Regulatory Commission

/RA/

William M. Dean, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendix A, Technical Specifications

Date of Issuance: December 22, 2016

Technical Specifications

License No. R-70, Docket No. 50-166

December 2, 2016

Contents

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.

DROP TIME — DROP TIME is the elapsed time between the initiation of a scram signal by either automated or operator action and the instant the slowest scrammable CONTROL ROD reaches its fully inserted position.

EXCESS REACTIVITY — Excess Reactivity is that amount of reactivity that would exist if all CONTROL RODS were moved to the maximum reactive condition from the point where the reactor is exactly critical ($k_{eff} = 1$).

EXPERIMENT — Any operation, hardware, or target (excluding devices such as detectors, foils, etc.), that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the EXPERIMENTAL FACILITIES, and that is not rigidly secured to a core or shield structure so as to be part of their design.

EXPERIMENTAL FACILITIES — Experimental facilities shall include the beam ports, pneumatic transfer systems and any in-core facilities.

EXPERIMENT SAFETY SYSTEMS — Experiment safety systems are those systems, including their associated input circuits, which are designed to initiate a scram for the primary purpose of protecting an EXPERIMENT or to provide information which requires manual protective action to be initiated.

FOUR ELEMENT FUEL BUNDLE — The four element fuel bundle consists of an aluminum bottom, 4 stainless steel clad FUEL ELEMENTS and aluminum top handle.

FUEL ELEMENT — A fuel element is a single TRIGA fuel rod.

FUELED DEVICE — An experimental device that contains fissionable material.

FULL POWER — Full licensed power is defined as 250 kW.

INSTRUMENTED FUEL ELEMENT (IFE) - An instrumented element is a special FUEL ELEMENT in which one or more thermocouples are embedded in the fuel.

ISOLATION — Isolation is the establishment of CONFINEMENT by closing of doors leading from the reactor bay area into the balcony area on the top floor, the door to the reception area on the ground floor, and the building exterior doors.

LIMITING CONDITIONS FOR OPERATION (LCO) - Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility.

LIMITING SAFETY SYSTEM SETTING (LSSS) - The limiting safety system setting is the FUEL ELEMENT temperature, which if exceeded, shall cause a reactor scram to be initiated, preventing the SAFETY LIMIT from being exceeded.

MEASURED VALUE — The measured value is the value of a parameter as it appears on the output of a CHANNEL.

ON CALL — A senior operator is "on call" if they are on the College Park campus or within 10 miles and can reach the facility within one half hour following a request.

OPERABLE — Operable means a component or system is capable of performing its intended function.

OPERATING — Operating means a component or system is performing its intended function.

REACTIVITY WORTH OF EXPERIMENT — The reactivity worth of an experiment is the value of the reactivity change that results from the EXPERIMENT, being inserted into or removed from its intended position.

REACTOR OPERATING — The reactor is operating whenever it is not secured or shutdown.

REACTOR OPERATOR (**RO**) — A reactor operator is an individual who is licensed by the U.S. Nuclear Regulatory Commission (NRC) to manipulate the controls of the reactor.

REACTOR SAFETY SYSTEM — Reactor safety systems are those systems, including their associated input CHANNELS, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

REACTOR SECURED — The reactor is secured when:

1. *Either* there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderator and reflection;

2. *Or* the following conditions exist:

   (a) All control devices (3 CONTROL RODS) are fully inserted;

   (b) The console key switch is in the off position and the key is removed from the lock;

   (c) No work is in progress involving core fuel, core structure, installed CONTROL RODS, or control rod drives unless they are physically decoupled from the CONTROL RODS;

   (d) No EXPERIMENTS in or near the reactor are being moved or serviced that have, on movement, a reactivity worth exceeding $1.00.

REACTOR SHUTDOWN — The reactor is shut down if it is subcritical by at least one dollar in the REFERENCE CORE CONDITION with the reactivity worth of all installed EXPERIMENTS included.

REFERENCE CORE CONDITION — The reference core condition is the reactivity condition of the core when it is at 20°C and the reactivity worth of xenon is negligible, $0.01 or less.

REPORTABLE OCCURRENCE — A REPORTABLE OCCURRENCE is any of the following:

1. Operation with actual safety system settings for required systems less conservative than the LIMITING SAFETY SYSTEM SETTINGS specified in Technical Specification 2.2;

2. Operation in violation of the LIMITING CONDITIONS FOR OPERATION established in the Technical Specifications;

3. Operation with a required REACTOR SAFETY SYSTEM component in an inoperative or failed condition which renders or could render the system incapable of performing its intended safety function. If the malfunction or condition is caused during maintenance, then no report is required;

4. An unanticipated or uncontrolled change in reactivity greater than one dollar. Reactor trips resulting from a known cause are excluded;

5. Abnormal and significant degradation in reactor fuel, or cladding, or both, coolant boundary, or CONFINEMENT boundary (excluding minor leaks);

6. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

SAFETY CHANNEL — A safety channel is a CHANNEL in the REACTOR SAFETY SYSTEM.

SAFETY LIMIT — Safety limits are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity.

SECURED SHUTDOWN — Secured shutdown is achieved when the reactor meets the requirements of the definition of "REACTOR SECURED" and the facility administrative requirements for leaving the facility with no licensed reactor operators present.

SENIOR REACTOR OPERATOR (SRO) — A senior reactor operator is an individual who is licensed by the NRC to direct the activities of reactor operators. Such an individual is also a reactor operator.

SHALL, SHOULD, MAY — The word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and the word "may" is used to denote permission, neither a requirement nor a recommendation.

SHUTDOWN MARGIN — Shutdown margin is the minimum reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible OPERATING condition and with the most reactive rod in its most reactive position, and that the reactor will remain subcritical without further operator action.

SHUTDOWN REACTIVITY — Shutdown reactivity is the value of the reactivity of the reactor with all CONTROL RODS in their least reactive position (*e.g.*, inserted). The shutdown reactivity includes the reactivity value of all installed EXPERIMENTS plus the reactivity of the REFERENCE CORE CONDITION.

STANDARD CORE — A standard core is an arrangement of standard TRIGA fuel in the reactor grid plate and the IFE in grid position D8.

STEADY STATE MODE — Steady state mode operation shall mean operation of the reactor with the mode selector switch in the steady state position.

THREE ELEMENT FUEL BUNDLE — The three element fuel bundle consists of an aluminum bottom, 3 stainless steel clad FUEL ELEMENTS, 1 CONTROL ROD GUIDE TUBE, and aluminum top handle.

TRUE VALUE — The true value is the actual value of a parameter.

UNSCHEDULED SHUTDOWN — An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the REACTOR SAFETY SYSTEM, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation, not to include shutdowns that occur during testing or checkout operations.

2 Safety Limit and Limiting Safety System Settings

2.1 Safety Limit

Applicability

This specification applies to the temperature of the reactor fuel.

Objective

The objective is to define the maximum FUEL ELEMENT temperature that can be permitted with confidence that no damage to the FUEL ELEMENT cladding will result.

Specification

The temperature in a standard TRIGA FUEL ELEMENT shall not exceed 1000°C under any conditions of operation, with the fuel fully immersed in water.

Basis

The most important SAFETY LIMIT for a TRIGA reactor is the UZrH FUEL ELEMENT temperature. This parameter is well suited as a single specification because it can be measured directly with a thermocouple. A loss in the integrity of the FUEL ELEMENT cladding could arise from a buildup of excessive pressure between the fuel-moderator and the cladding if the fuel temperature exceeds the SAFETY LIMIT. The pressure is caused by the presence of air, fission product gases, and hydrogen from the dissociation of the hydrogen and zirconium in the fuel-moderator. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium. The data indicate that the stress within the FUEL ELEMENT will not result in the compromise of the stainless steel cladding if the temperature in the fuel does not exceed 1000°C and the fuel cladding is water-cooled.

It has been shown by experience that operation of TRIGA reactors at a power level of 1000 kW will not result in damage to the fuel. Several reactors of this type have operated successfully for several years at power levels up to 1500 kW. Analysis and measurements on other TRIGA reactors have shown that a power level of 1000 kW corresponds to a peak fuel temperature of approximately 400°C.

2.2 Limiting Safety System Settings

Applicability

This specification applies to the reactor scram setting that prevents the reactor fuel temperature from reaching the SAFETY LIMIT.

Objective

The objective is to prevent the SAFETY LIMIT from being reached.

Specification

The LIMITING SAFETY SYSTEM SETTING shall be 175°C as measured by the INSTRUMENTED FUEL ELEMENT (IFE).

Basis

A LIMITING SAFETY SYSTEM SETTING of 175°C provides a safety margin of 650°C. A part of the safety margin is used to account for the difference between the temperature at the hot spot in the fuel and the measured temperature resulting from the actual location of the thermocouple. If the IFE is located in the hottest position in the core, the difference between the true and measured temperatures will be only a few degrees since the thermocouple junction is at the mid-plane of the element and close to the anticipated hot spot. If the IFE is located in a region of lower temperature, such as on the periphery of the core, the measured temperature will differ by a greater amount from that actually occurring at the core hot spot. Calculations have shown that if the IFE were located on the periphery of the core, the true temperature at the hottest location in the core will differ from the measured temperature by no more than a factor of two. Thus, with the IFE positioned in the location specified in the STANDARD CORE, when the temperature in the thermocouple reaches the setting of 175 °C, the true temperature at the hottest location would be no greater than 350°C, providing a margin to the SAFETY LIMIT of at least 650°C. This margin is ample to account for the remaining uncertainty in the accuracy of the fuel temperature measurement channel, and any overshoot in reactor power resulting from a reactor transient while the reactor is OPERATING.

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.

4. Specification 3.1.4 limits the fission product release that might accompany operation with a damaged fuel element. Fuel will be considered potentially “damaged” if said fuel is found to be leaking in the air and/or water sample or under such case that the fuel has been exposed to a temperature above 175°C as measured on the INSTRUMENTED FUEL ELEMENT. If either sample is found to contain I-129 through I-135, Xe-135, Kr-85,87, or Kr-89, Cs-135 or Cs-137, or Sr-89 through Sr-92, this meets the criteria for a leaking FUEL ELEMENT.

5. General Atomic tests of TRIGA fuel indicate that keeping FUEL ELEMENT burnup below 50% of the original U-235 loading will avoid damage to the fuel from fission product buildup. The elongation, bend, and burn-up limits are values that have been found acceptable to the NRC (NUREG-1537).

3.2 Reactor Control and Safety Systems

Applicability

These specifications apply to reactor control and safety systems and safety-related instrumentation that must be OPERABLE when the reactor is OPERATING.

Objective

The objective of these specifications is to specify the lowest acceptable level of performance or the minimum number of OPERABLE components for the reactor control and safety systems.

Specifications

1. The DROP TIME from the initiation of scram signal of each of the three CONTROL RODS from the fully withdrawn position to the fully inserted position shall not exceed one second.

2. Maximum positive reactivity insertion rate by CONTROL ROD motion shall not exceed $0.30 per second.

3. The reactor SAFETY CHANNELS shall be OPERABLE in accordance with Table 3.1, including the minimum number of channels and the indicated maximum or minimum set points for the scram channels.

4. The safety interlocks shall be OPERABLE in accordance with Table 3.2, including the minimum number of interlocks.

5. The Beam Port and Through Tube interlocks may be bypassed during a reactor operation with the permission of the Reactor Director.

6. A minimum of one reactor power channel, calibrated for reactor thermal power, shall be attached to a recording device sufficient for auditing of reactor operation history.

Bases

1. Specification 3.2.1 ensures that the reactor will be shutdown promptly when a scram signal is initiated. Experiments and analysis have indicated that for the range of transients anticipated for the MUTR TRIGA reactor, the specified CONTROL ROD DROP TIME is adequate to ensure the safety of the reactor.

2. Specification 3.2.2 establishes a limit on the rate of change of power to ensure that the normally available reactivity and insertion rate cannot generate OPERATING conditions that exceed the SAFETY LIMIT. (See FSAR)

3. Specification 3.2.3 provides protection against the REACTOR OPERATING outside of the SAFETY LIMITS. Table 3.3 describes the basis for each of the reactor SAFETY CHANNELS.

4. Specification 3.2.4 provides protection against the REACTOR OPERATING outside of the SAFETY LIMITS. Table 3.4 describes the basis for each of the reactor safety interlocks.

5. Specification 3.2.5 ensures that reactor interlocks will always serve their intended purpose. This purpose is to assure that the operator is aware of the status of both the beam ports and the through tube.

6. Specification 3.2.6 provides for a means to monitor reactor operations and verify that the reactor is not operated outside of its license condition.

Scram Channel	Minimum Required Operable	Scram Setpoint
Reactor Power Level	2	Not to exceed 120%
Fuel Element Temperature	1	Not to exceed 175°C
Power Supply for Reactor Power Levels	2	<90%
Manual Scram	1	N/A
Console Electrical Supply	1	Loss of electrical power to the control console
Rate of Power Change - Period	1	Not less than 5 seconds
Radiation Area Monitors	1	<50 mr/hr (bridge monitor) <10 mr/hr (exhaust monitor)
Pool Water Temperature	1	< 90°C, manual scram

Table 3.1: Reactor Safety Channels: Scram Channels

Interlock/Channel	Function
Log Power Level	Provides signal to period rate and minimum source channels. Prevent control rod withdrawal when neutron count rate is less than 1 cps.
Startup Count Rate	Prevent control rod withdrawal when neutron count rate is less than 1 cps.
Safety 1 Trip Test	Prevent control rod withdrawal when Safety 1 Trip Test switch is not in operate.
Plug Electrical Connection	Disable magnet power when Beam Port or Through Tube plug is removed unless bypass has been activated.
Rod Drive Control	Prevent simultaneous manual withdrawal of two or more control rods in the steady state mode of operation.

Table 3.2: Reactor Safety Channels: Interlocks

Interlock/Channel	Function
Reactor Power Level/Fuel Element Temperature	Provides protection to assure that the reactor can be shut down before the safety limit on the fuel element temperature will be exceeded.
Power Supply for Reactor Power Levels	Provides protection to assure that the reactor cannot be operated unless the neutron detectors that input to each of the linear power channels are operable.
Manual Scram	Allows the operator to shut down the reactor if an unsafe or abnormal condition occurs.
Console Electric Supply	Ensures that the reactor cannot be operated without a stable electric supply.
Rate of power change - Period	Ensures that the reactor is operated in a manner that allows the operator time to shut down the reactor before the licensed power restriction is exceeded.
Radiation Area Monitors	Assures that the reactor automatically scrams if a high airborne radiation level is detected.

Table 3.3: Reactor Safety Channels: Scram Channel Bases

Interlock/Channel	Function
Log Power Level	This channel is required to provide a neutron detector input signal to the startup count rate channel.
Startup Count Rate	Assures sufficient amount of startup neutrons are available to achieve a controlled approach to criticality.
Safety 1 Trip Test	Assures that the 1 cps interlock cannot be bypassed when not in operate position.
Plug Electrical Connection	Assures that the reactor cannot be operated with beamport or through tube plugs removed without further precautions.
Rod Drive Control	Limits the maximum positive reactivity insertion rate available for steady state operation.

Table 3.4: Reactor Safety Channels: Interlocks Bases

3.3 Primary Coolant System

Applicability

This specification applies to the quality and quantity of the primary coolant in contact with the fuel cladding at the time of reactor startup.

Objectives

1. To minimize the possibility for corrosion of the cladding on the FUEL ELEMENTS.
2. To minimize neutron activation of dissolved materials.
3. To ensure sufficient biological shielding during while the reactor is OPERATING.

4. To maintain water clarity.

Specifications

1. A minimum of 15 ft. of coolant shall be above the core.
2. Conductivity of the pool water shall be no higher than 5×10^{-6} mhos/cm.
3. The concentrations of radionuclides in the bulk pool water shall be no higher than the values presented for water in 10 CFR Appendix B to Part 20 Table 3.
4. The pool water temperature shall not exceed 90°C, as measured by thermocouples located in the pool.

Bases

1. Specification 3.3.1 ensures that both sufficient cooling capability and biological shielding are available for safe reactor operation.
2. A low rate of corrosion continuously occurs in a water-metal system. In order to limit this rate, and thereby extend the longevity and integrity of the fuel cladding, a filtration and demineralization system is required. Experience with water quality control at many reactor facilities has shown that maintenance within the specified limit provides acceptable control. In addition, by limiting the concentration of dissolved materials in the water, the radioactivity of neutron activation products is limited. This is consistent with the ALARA principle, and tends to decrease the inventory of radionuclides in the entire coolant system, which will decrease personnel exposure during maintenance and operation.
3. Specification 3.3.3 ensures that a fuel failure with release of radioactive materials into the pool will be detected.
4. Specification 3.3.4 ensures a departure from nucleate boiling ratio value greater than 2.

3.4 Confinement and Ventilation System

Applicability

This specification applies to that part of the facility that contains the reactor, its controls and shielding.

Objective

The objective of these specifications is to ensure that sufficient CONFINEMENT volume is available for the dilution of radioactive releases and to limit the rate of release of radioactive material to the outside environment.

Specifications

1. CONFINEMENT shall be considered established when the doors leading from the reactor bay area into the balcony area on the top floor, and the reception area as well as the building exterior are secured and ventilation system is off with louvers closed.
2. CONFINEMENT shall be established whenever the reactor is not secured or radioactive material with significant potential for airborne release is being handled, with the exception of the time that persons are physically entering or leaving the CONFINEMENT area.
3. Forced air ventilation to the outside shall automatically secure without operator intervention in such case that the radiation levels exceed a preset level as defined in Table 3.1.

Bases

1. This specification provides the necessary requirements for confinement, which ensures releases to the outside environment are within 10 CFR Part 20 requirements.
2. This specification provides the reactor status condition for confinement, as well as allows personnel to enter and leave the reactor building, as required, when the reactor is unsecured.
3. This specification ensures that radioactive release will be minimized by stopping forced flow to the outside environment.

3.5 Radiation Monitoring System and Effluents

3.5.1 Radiation Monitoring System

Applicability

This specification applies to the radiation monitoring information that must be available to the REACTOR OPERATOR while the reactor is OPERATING.

Objective

The objective is to assure that sufficient radiation monitoring information is available to the operator to assure safe operation of the reactor.

Specifications

1. The reactor shall not be operated unless the bridge monitor is OPERATING.
2. For a period of time not to exceed 48 hours for maintenance or calibration to the radiation monitor channel, the intent of specification 3.5.1 shall be satisfied if the bridge radiation monitor is replaced with portable gamma sensitive instrument having its own alarms or which shall be observable by the REACTOR OPERATOR.
3. The alarm set point shall be stated in a facility OPERATING procedure. The alarm set-point for the bridge monitor is <37 mR/hr (alert), <50 mR/hr (scram).
4. The campus radiation safety organization shall maintain environmental monitors at the greatest points of release.

Bases

1. Specification 3.5.1.1 ensures that a significant fuel failure with release of radioactive materials will be determined and that any large releases will be mitigated by the specified protective actions.
2. Specification 3.5.1.2 allows for continued reactor operation if maintenance and/or calibration of the radiation area monitor is required.
3. The alarm and scram setpoint shall be designed to ensure that dose rates delivered to areas accessible to members of the general public do not exceed the levels defined in 10 CFR Part 20. Additionally, the radiation area monitor provides information to OPERATING personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings.
4. The intent of Specifications 3.5.1.3 and 3.5.1.4 is to ensure that the facility does not generate a dose to the general public greater than that allowed by 10 CFR Part 20.

3.5.2 Effluents

Applicability

This specification applies to limits on effluent release.

Objective

The objective is to ensure that the release of radioactive materials from the reactor facility to unrestricted areas do not exceed federal regulations.

Specification

All effluents from the MUTR shall conform to the standards set forth in 10 CFR Part 20.

Basis

The intent of 3.5.2 is to ensure that, in the event that radioactive effluents are released, the dose to the general public will be less than that allowed by 10 CFR Part 20.

3.6 Limitations on Experiments

Applicability

This specification applies to EXPERIMENTs installed in the reactor and its EXPERIMENTAL FACILITIES.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive material in the event of an EXPERIMENT failure.

Specifications

The reactor shall not be operated unless the following conditions governing experiments exist.

1. The reactivity worth of any single EXPERIMENT shall be less than $1.00.
2. The total absolute reactivity worth of EXPERIMENTS shall not exceed $3.00, including the potential reactivity which might result from experimental malfunction and EXPERIMENT flooding or voiding.
3. Experiments containing materials corrosive to reactor components, compounds highly reactive with water, and liquid fissionable materials shall be doubly encapsulated. In the event of a failure, the potentially impacted reactor components shall be inspected for damage. The results of the inspection and any corrective action taken shall be reviewed by the Director or designated alternate and determined to be satisfactory before operation of the reactor is resumed.
4. Explosive materials or potential explosive materials in quantities greater than 25 mg TNT or its equivalent shall not be irradiated in the reactor or EXPERIMENTAL FACILITIES. Explosive materials in quantities equal to or less than 25 mg TNT or its equivalent may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than the failure pressure of the container. The failure pressure of the container is one half of the design pressure. Total explosive material inventory in the reactor facility may not exceed 100 mg TNT or its equivalent. In the event of a failure, the potentially impacted reactor components shall be

inspected for damage. The results of the inspection and any corrective action taken shall be reviewed by the Director or designated alternate and determined to be satisfactory before operation of the reactor is resumed.

5. EXPERIMENT materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (1) normal OPERATING conditions of the EXPERIMENT or reactor, (2) credible accident conditions in the reactor or (3) possible accident conditions in the EXPERIMENT shall be limited in type and quantity such that if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the airborne radioactivity in the reactor room or outside environment will not result in exceeding the applicable dose limits set forth in 10 CFR Part 20.

   In calculations pursuant to 3.6.5 above, the following assumptions shall be used:

   (a) If the effluent from an experimental facility exhausts through a holdup tank, which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced shall escape.
   (b) If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3μm particles, at least 10% of these particles shall escape.
   (c) If an EXPERIMENT fails and releases radioactive gases or aerosols to the reactor bay or atmosphere, 100% of the radioactive gases or aerosols shall escape.
   (d) If an EXPERIMENT fails that contains materials with a boiling point above 54°C, the vapors of at least 10% of the materials shall escape through an undisturbed column of water above the core.

6. Each fueled EXPERIMENT shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the EXPERIMENT is no greater than 5 mCi.

Bases

1. This specification is intended to provide assurance that the worth of a single EXPERIMENT will be limited to a value such that the SAFETY LIMIT will not be exceeded if the positive worth of the EXPERIMENT were to be inserted suddenly.

2. The maximum worth of a single EXPERIMENT is limited so that its removal from the cold critical reactor will not result in the reactor achieving a power level high enough to exceed the SAFETY LIMIT. Since experiments of such worth must be fastened in place, its inadvertent removal from the REACTOR OPERATING at FULL POWER would result in a relatively slow power increase such that the reactor protective systems would act to prevent high power levels from being attained.

   The maximum worth of all experiments is also limited to a reactivity value such that the cold reactor will not achieve a power level high enough to exceed the core temperature SAFETY LIMIT if the experiments were removed inadvertently.

3. This specification is intended to prevent damage to reactor components resulting from EXPERIMENT failure. If an EXPERIMENT fails, inspection of reactor structures and components shall be performed in order to verify that the failure did not cause damage. If damage is found, appropriate corrective actions shall be taken.

4. This specification is intended to prevent damage to reactor components resulting from failure of an EXPERIMENT involving explosive materials, especially the accidental detonation of the explosive. If an EXPERIMENT fails, inspection of reactor structures and components shall be performed in order to verify that the failure did not cause damage. If damage is found, appropriate corrective actions shall be taken.

5. This specification is intended to reduce the likelihood that airborne activities in excess of the limits of Table 2 of Appendix B of 10 CFR Part 20 will be released to the atmosphere outside the facility boundary.

6. The 5 mCi limitation on iodine 131 through 135 assures that in the event of failure of a fueled EXPERIMENT leading to total release of the iodine, the exposure dose at the exclusion area boundary will be less than that allowed by 10 CFR Part 20 for an unrestricted area. (See SAR)

4 Surveillance Requirements

Applicability

This specification applies to the surveillance requirements of any system related to reactor safety.

Objective

The objective is to verify the proper operation of any system related to reactor safety.

Specification

Surveillances shall be performed on a timely basis. In the event that the reactor is not in an OPERABLE condition, such as during periods of refueling, or replacement or repair of safety equipment, surveillances may be postponed, see Table 4.1, until such time that the reactor is OPERABLE. In such case that any surveillance must be postponed, a written directive signed by the Director, shall be placed in the records indicating the reason why and the expected completion date of the required surveillance. This directive shall be written before the date that the surveillance is due. Under no circumstance shall the reactor perform routine operations until such time that all surveillances are current and up to date. Any system or component that is modified, replaced, or had maintenance performed shall undergo testing to ensure that the system/component continues to meet performance requirements.

Technical Specification	Defer during shutdown?	Required prior to operations?
4.1 Reactor Core Parameters	Yes	Yes
4.2 Reactor Control and Safety Systems	Yes	Yes
4.3 Primary Coolant System	No	N/A
4.4 Confinement and Ventilation	Yes	Yes
4.5 Radiation Monitoring and Effluents	No	N/A
4.6 Experiments	Yes	Yes

Table 4.1: Surveillance Requirements

4.1 Reactor Core Parameters

Applicability

These specifications apply to the surveillance requirements for the reactor core.

Objective

The objective of these specifications is to ensure that the specifications of Section 3.1 are satisfied.

Specifications

1. The EXCESS REACTIVITY shall be determined annually, at intervals not to exceed 15 months, and after each time the core fuel configuration is changed, these changes include any removal or replacement of CONTROL RODS.

2. The SHUTDOWN MARGIN shall be determined annually, at intervals not to exceed 15 months, and 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 at least 4 FOUR ELEMENT FUEL BUNDLES from rows B and C shall be performed annually at intervals not to exceed 15 months. The bundles inspected shall rotate such that in a 2-year period all accessible FOUR ELEMENT FUEL BUNDLES in rows B and C are inspected. 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.

Bases

1. The reactivity worth of the CONTROL RODS, specification 4.2.1, is measured to assure that the required SHUTDOWN MARGIN is available and to provide a means to measure the REACTIVITY WORTH OF EXPERIMENTS. Long term effects of TRIGA reactor operation are such that measurements of the reactivity worths on an annual basis are adequate to insure that no significant changes in SHUTDOWN MARGIN have occurred.
2. The CONTROL ROD withdrawal and insertion rates, specification 4.2.2, are measured to insure that the limits on maximum reactivity insertion rates are not exceeded.
3. Measurement of the CONTROL ROD DROP TIME, specification 4.2.3, ensures that the rods can perform their safety function properly.
4. The surveillance requirement specified in specification 4.2.4 for the reactor safety scram channels ensures that the CHANNELS are OPERABLE.
5. The surveillance test performed after maintenance or repairs to the REACTOR SAFETY SYSTEM as required by specification 4.2.5 to ensure that the affected CHANNEL will be OPERABLE.
6. The linear power level CHANNEL CALIBRATION specified in specification 4.2.6 will assure that the reactor will be operated at the licensed power levels.
7. Specification 4.2.7 assures that a visual inspection of CONTROL ROD poison sections is made to evaluate corrosion and wear characteristics and any damage caused by operation in the reactor.
8. Specification 4.2.8 assures that a visual inspection of control drive mechanisms is made to evaluate corrosion and wear characteristics and any damage caused by operation in the reactor.

4.3 Primary Coolant System

Applicability

These specifications apply to the surveillance requirements of the reactor primary coolant system.

Objective

The objective of these specifications is to ensure the reactor primary coolant system is OPERABLE as described in Section 3.3.

Specifications

1. The primary coolant level shall be verified before each reactor startup or daily during operations exceeding 24 hours.
2. Pool water conductivity shall be determined prior to the first startup of the day.
3. Pool water gross gamma activity shall be determined monthly, at intervals not to exceed six weeks. If gross gamma activity is high (greater than twice historical data), gamma spectroscopy shall be performed. Gamma spectroscopy shall be performed quarterly, not to exceed 4 months.
4. Pool water temperature shall be measured prior to the reactor startup and shall be monitored during reactor operation.

Bases

1. Specification 4.3.1 ensures that sufficient water exists above the core to provide both cooling capacity and an adequate biological shield.
2. Specification 4.3.2 ensures that poor pool water quality could not exist for long without being detected. Years of experience at the MUTR have shown that pool water analysis on a monthly basis is adequate to detect degraded conditions of the pool water in a timely manner.
3. Gross gamma activity measurements are conducted to detect fission product releases from damaged FUEL ELEMENT cladding.
4. Specification 4.3.4 ensures that the maximum allowable pool water temperature is not exceeded.

4.4 Confinement and Ventilation System

Applicability

This specification applies to that part of the facility which contains the reactor, its controls and shielding.

Objective

The objective of this specification is to ensure that radioactive releases from the CONFINEMENT can be limited.

Specifications

1. Prior to putting the reactor in an unsecured mode, the ISOLATION of the CONFINEMENT building shall be visually verified.
2. The ability to close the louvers of the ventilation system shall be verified before the first reactor operation after a SECURED SHUTDOWN.

Bases

1. This specification ensures that the minimal leakage rate assumed in the SAR is actually present during reactor operations in order to limit the release of radioactive material to the environs.
2. The facility is designed such that in the event that excessive airborne radioactivity is detected the ventilation system shall be shutdown to minimize transport of airborne materials. Analysis indicates that in the event of a major FUEL ELEMENT failure personnel would have sufficient time to evacuate the facility before the maximum permissible dose as given in 10 CFR Part 20 is exceeded.

4.5 Radiation Monitoring System and Effluents

4.5.1 Radiation Monitoring System

Applicability

This specification applies to the surveillance requirements for the Radiation Area Monitoring System (RAMS).

Objective

The objective of these specifications is to ensure the operability of the radiation area monitoring CHANNEL required by Section 3.5 and to ensure that releases to the environment are kept below allowable limits.

Specifications

1. A CHANNEL CALIBRATION shall be made for the CHANNEL listed in Section 3.5 annually, at intervals not to exceed 15 months or whenever maintenance or repairs are made that could affect its calibration.
2. A CHANNEL TEST shall be made for the CHANNEL listed in Section 3.5 prior to starting up the reactor to ensure it is OPERABLE.

Bases

Specifications 4.5.1.1 and 4.5.1.2 ensure that the radiation area monitor is checked and calibrated on a routine basis, in order to assure compliance with 10 CFR Part 20.

4.5.2 Effluents

Applicability

This specification applies to the surveillance requirements for air and water effluents.

Objective

The objective of these specifications is to ensure that releases to the environment are kept below allowable limits.

Specifications

1. Reactor building air samples shall be counted for gross gamma activity monthly, at intervals not to exceed 6 weeks.
2. A sample of any water discharged from the reactor building sump shall be counted for gross gamma activity before its release to the environs.
3. Environmental monitors at the boundary shall be exchanged at least quarterly, not to exceed 4 months.

Bases

Specifications 4.5.2.1 and 4.5.2.2 ensure that the facility effluents comply with 10 CFR Part 20.

4.6 Experiments

Applicability

This specification applies to the surveillance requirements for EXPERIMENTS installed in the reactor and its irradiation facilities.

Objective

The objective of this specification is to prevent the conduct of EXPERIMENTS or irradiations that may damage the reactor or release excessive amounts of radioactive materials as a result of failure.

Specifications

1. The reactivity worth of an EXPERIMENT shall be estimated or measured, as appropriate, before reactor operation with said EXPERIMENT.

2. An EXPERIMENT shall not be installed in the reactor or its irradiation facilities unless a safety analysis has been performed and reviewed for compliance with Section 3.6 by the Reactor Safety Committee or Director, in full accord with Sections 6.1.2, 6.2.1, and 6.5 of these Technical Specifications and the procedures which are established for this purpose.

Bases

Experience has shown that experiments reviewed and approved by the Reactor Safety Committee or Director can be conducted without endangering the safety of the reactor, personnel, or exceeding Technical Specification limits.

5 Design Features

5.1 Site Characteristics

Applicability

This specification applies to the reactor facility and its site boundary.

Objective

The objective is to assure that appropriate physical security is maintained for the reactor facility and the radioactive materials contained within it.

Specifications

1. The reactor shall be housed in a closed room, with free air volume of 1700 m^3, designed to restrict leakage. The closed room does not include the west balcony area.
2. The licensed reactor site boundary shall consist of the outer walls of the reactor building and the area enclosed by the loading dock fence.
3. The restricted area shall consist of all areas interior to the reactor building including the west balcony and lower entryway.
4. The controlled area shall consist of all areas interior to the reactor building including the west balcony and lower entryway.

Basis

These specifications assure that appropriate control is maintained over access to the facility by members of the general public.

5.2 Reactor Primary Coolant System

Applicability

This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective

The objective is to assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specifications

1. The reactor core shall be cooled by natural convective water flow.
2. Any piping system, such as the primary demineralizer and diffuser system, immersed in the pool, capable of draining pool level below the minimum level shall be equipped with a syphon break to prevent accidental draining of pool below 29 inches of reference level.

Bases

1. Specification 5.2.1 is based on thermal and hydraulic calculations and operation of the MUTR that show that the core can operate in a safe manner at power levels up to 300 kW with natural convection flow of the coolant.

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 into THREE or FOUR ELEMENT FUEL BUNDLES.
2. The fuel bundles shall be arranged in a rectangular 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.

5.3.2 Control Rods

Applicability

This specification applies to the CONTROL RODS used in the reactor core.

Objective

The objective is to assure that the CONTROL RODS are of such a design as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications

1. The three CONTROL RODS shall have scram capability, shall be used for reactivity control, and shall contain borated graphite, B_4C, in powder form.
2. The CONTROL ROD cladding shall be aluminum with nominal thickness 0.028" and length 17".

Basis

The poison requirements for the control rods are satisfied by using neutron absorbing borated graphite, B_4C, powder. These materials must be contained in a suitable cladding material such as aluminum to ensure mechanical stability during movement and to isolate the poison from the tank water environments. Scram capabilities are provided for rapid insertion of the control rods, which is the primary safety feature of the reactor.

5.4 Fissionable Material Storage

Applicability

This specification applies to the storage of reactor fuel at times when it is not in the reactor core.

Objective

The objective is to assure that fuel that is being stored will not become critical and will not reach an unsafe temperature.

Specifications

1. All FUEL ELEMENTS and FUELED DEVICES shall be stored either in a geometrical array where the k_{eff} is less than 0.8 for all conditions of moderation and reflection or stored in an approved fuel shipping container.
2. Irradiated fuel elements and fueled devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the FUEL ELEMENT or FUELED DEVICE temperature will not exceed design values.
3. When fuel is in storage in any area other than the grid plate, that area shall be equipped with monitoring devices that both measure and record the radiation levels and temperature of the region surrounding the fuel.

Basis

The limits imposed by Specifications 5.4.l and 5.4.2 are conservative and assure safe storage.

6 Administrative Controls

6.1 Organization

The Maryland University Training Reactor (MUTR) shall be owned and operated by the University of Maryland, College Park. The position of the MUTR within the University of Maryland structure is show in Figure 6.1. The university shall provide whatever resources are required to maintain the facility in a condition that poses no hazard to the general public or to the environment.

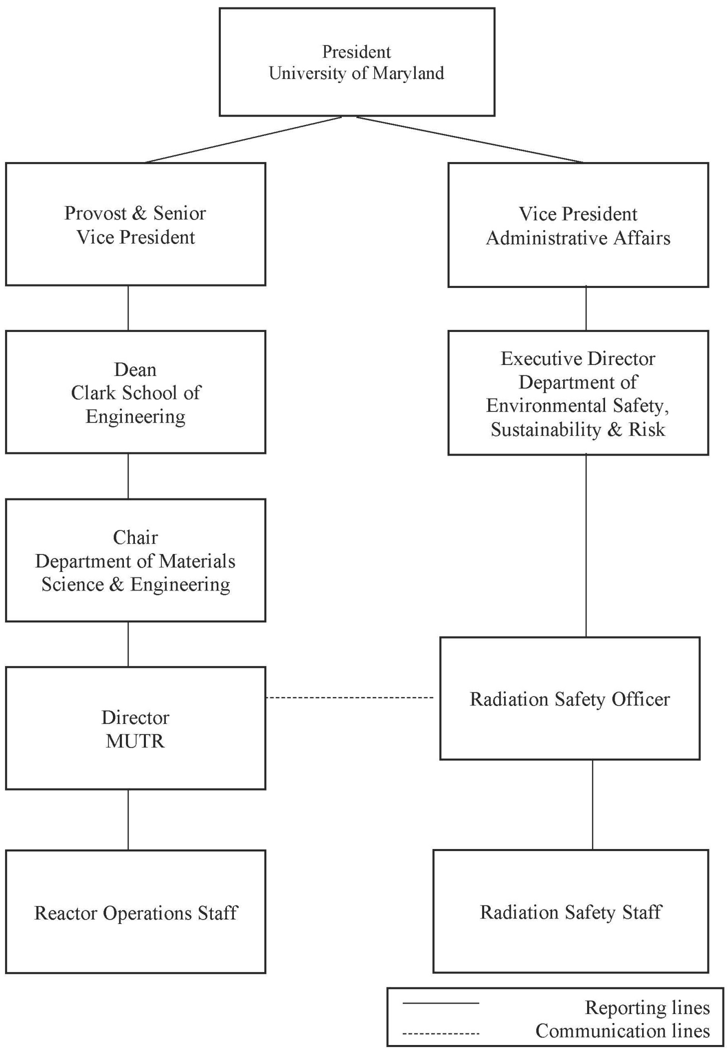

Figure 6.1: Position Chart

The Dean of the A. James Clark School of Engineering and the Chair of the Department of Materials Science and Engineering shall be responsible for adhering to all requirements of the OPERATING license. The Director of MUTR shall be responsible for the administration and safe operation of the facility. The Reactor Safety Committee shall advise the director of MUTR on matters pertaining to safety. The Radiation Safety

Officer shall provide onsite advice concerning personnel radiological safety and provides assistance and review in the area of radiation protection.

6.1.1 Structure

1. A line management organizational structure, Figure 6.2, shall provide for personnel who will administrate and operate the reactor facility. The members listed in the organization chart shall be responsible for safeguarding the public and facility personnel from undue radiation exposure and for adhering to all requirements of the OPERATING license.

2. Management levels:

    (a) Level 1: Dean, A. James Clark School of Engineering and Chair, Department of Materials Science & Engineering: Shall be responsible for MUTR facility license.

    (b) Level 2: Director, MUTR: Shall be responsible for reactor facility operation, adherence to the regulations, facility license, the technical specifications, and shall report to Level 1.

    (c) Level 3: Supervisory SENIOR REACTOR OPERATORS: Shall be responsible for the day-to-day operation of the MUTR including shift operation and shall report to Level 2.

    (d) Level 4: Operating Staff: Licensed REACTOR OPERATORS, SENIOR REACTOR OPERATORS, and trainees. These individuals shall report to Level 3.

6.1.2 Responsibility

Responsibility for the safe operation of the reactor facility shall be in accordance with the line organization established in Figure 6.2. In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

6.1.3 Staffing

1. The minimum staffing when the reactor is not secured shall be:

    (a) A licensed REACTOR OPERATOR (RO) or a licensed SENIOR REACTOR OPERATOR (SRO) shall be present in the control room.

    (b) A minimum of two persons shall be present in the facility or in the Chemical & Nuclear Engineering Building while the reactor is not secured: the operator in the control room and a second person who can be reached from the control room who is able to carry out prescribed written instructions which may involve activating elements of the Emergency Plan, including evacuation and initial notification procedures.

    (c) A licensed SRO shall be present or readily available ON CALL. "Readily available ON CALL" means an individual who

        i. has been specifically designated and the designation known to the operator on duty,

        ii. keeps the operator on duty informed of where he/she may be rapidly contacted and the method of contact, and

        iii. is capable of arriving at the reactor facility within a reasonable amount of time under normal conditions. At no time while the reactor is not secured shall the designated SRO be more than thirty minutes or ten miles from the facility.

2. A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator. The list shall include:

    (a) management personnel,

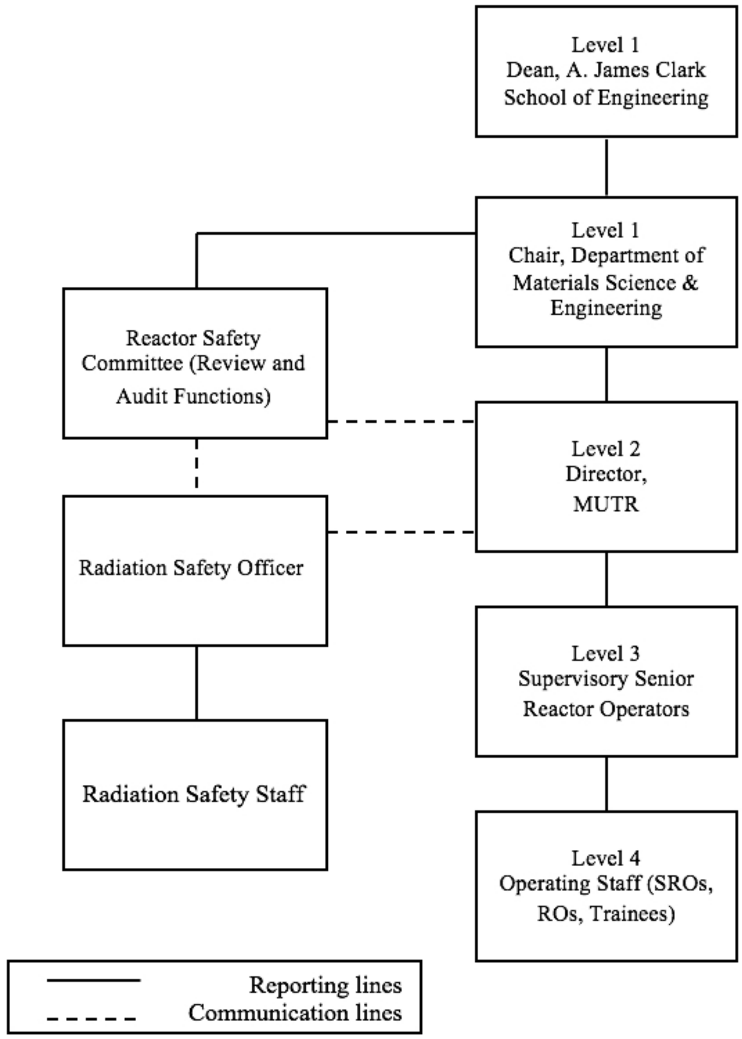

Figure 6.2: Organizational chart

(b) radiation safety personnel,

(c) other operations personnel.

3. The following operations shall be supervised by a SENIOR REACTOR OPERATOR:

   (a) Initial startup and approach to power for a startup requiring a startup checklist

   (b) When EXPERIMENTS are being manipulated in or near the core that have an estimated worth greater than $0.80

   (c) Removal of CONTROL RODS or fuel manipulations in the core

   (d) Recovery from unplanned or UNSCHEDULED SHUTDOWN or unplanned significant power reduction (greater than 10%).

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall be in accordance with the following:

1. Responsibility: The Director or designated alternate shall be responsible for the selection, training, and requalification of the facility REACTOR OPERATORS and SENIOR REACTOR OPERATORS.

2. Selection: The selection of operations personnel shall be consistent with the standards related to selection in ANSI/ANS-15.4-2007.

3. Training Program: The Training Program shall be consistent with the standards related to training in ANSI/ANS-15.4-2007.

4. Requalification Program: The Requalification Program shall be the Requalification/Training Program for the Maryland University Training Reactor.

6.2 Review and Audit

6.2.1 Reactor Safety Committee

A Reactor Safety Committee (RSC) shall exist for the purpose of reviewing matters relating to the health and safety of the public and facility staff and the safe operation of the facility. It shall be appointed by and reports to the Chairperson of the Department of Materials Science and Engineering. The RSC shall consist of a minimum of five persons with expertise in the physical sciences and preferably some nuclear experience. Permanent members of the committee shall be the Director and the Campus Radiation Safety Officer or designated alternate, neither may serve as the committee's chairperson. Qualified alternates may serve on the committee. Alternates may be appointed by the Chairperson of the RSC to serve on a temporary basis. At least one committee member shall be from outside the Department of Materials Science and Engineering.

6.2.1.1 RSC Charter and Rules

1. The RSC shall meet at least twice per year, and more often as required.

2. A quorum of the RSC shall be not less than half of the committee members, one of whom shall be the Campus Radiation Safety Officer (or designated alternate). No more than two alternates shall be used to make a quorum. MUTR staff members shall not constitute the majority of a voting quorum.

3. Minutes of all meetings will be retained in a file and distributed to all RSC members within 3 months.

6.2.1.2 RSC Review Function

The review responsibilities of the RSC shall include, but are not limited to the following:

1. Review and evaluation of determinations of whether proposed changes to the facility, procedures, and tests or experiments shall be made under 10 CFR 50.59 or would require a change in Technical Specifications or license conditions;

2. All new procedures, major revisions of procedures, and proposed changes in reactor facility equipment or systems which have significant safety impact to reactor operations;

3. All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity;

4. Proposed changes to the technical specifications, or license;

5. Violations of technical specifications or license. Violations of internal procedures or instructions having safety significance;

6. OPERATING abnormalities having safety significance;
7. REPORTABLE OCCURRENCES listed in Section 6.7.2;
8. Audit reports.

A written report of the findings and recommendations of the RSC shall be submitted to Level 1 management, the Director, and the RSC members within 3 months after the review has been completed.

6.2.1.3 RSC Audit Function

1. An annual audit of the reactor operations shall be performed by an individual or group familiar with research reactor operations. The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Discussion with cognizant personnel and observation of operations should be used also as appropriate. In no case shall the individual responsible for the area perform an audit in that area. They shall submit a report to the Director and the Reactor Safety Committee.

2. The following shall be audited:

   (a) Facility operations for conformance to the technical specifications, applicable license conditions, and standard OPERATING procedures: at least once per calendar year (interval between audits not to exceed 15 months);

   (b) The results of action taken to correct those deficiencies that may occur in the reactor facility equipment systems, structures, or methods of operations that affect reactor safety: at least once per calendar year (interval between audits not to exceed 15 months);

   (c) The retraining and requalification program for the operating staff: at least once every other calendar year (interval between audits not to exceed 30 months);

   (d) The reactor facility emergency plan and implementing procedures: at least once every other calendar year (interval between audits not to exceed 30 months); and

   Deficiencies uncovered that affect reactor safety shall immediately be reported to Level 1 management and the Director. A written report of the findings of the audit shall be submitted to Level 1 management, the Director, and the RSC members within 3 months after the audit has been completed.

6.3 Radiation Safety

A radiation safety program following the requirements established in 10 CFR Part 20 will be undertaken by the Radiation Safety Office. The Director will ensure that ALARA principles are followed during all facility activities. The program shall follow the guidelines of "Radiation Protection at Research Reactor Facilities," ANSI/ANS-15.11-1993 (2004).

6.4 Operating Procedures

Written operating procedures shall be prepared, reviewed, and approved before initiating any of the activities listed in this section. The procedures shall be reviewed and approved by the Director or a designated alternate, the Reactor Safety Committee, and shall be documented in a timely manner. The procedures shall be adequate to assure the safety of the reactor, but shall not preclude the use of independent judgment and action should the situation require such. OPERATING procedures shall be used for the following items:

1. Start-up, operation, and shutdown of the reactor;
2. Installation or removal of FUEL ELEMENTS, CONTROL RODS, EXPERIMENTS, and EXPERIMENTAL FACILITIES ;

3. Maintenance procedures that could have an effect on reactor safety;
4. Periodic surveillance checks, calibrations, and inspections required by the Technical Specifications or those that may have an effect on reactor safety;
5. Administrative controls for operations and maintenance and for the conduct of irradiations and EXPERIMENTS that could affect reactor safety or core reactivity;
6. For any activity pertaining to shipping, possession, and transfer of radioactive material, these procedures shall be written in conjunction with the Radiation Safety Office and the Radiation Safety Officer who shall inform the Reactor Director of any changes in regulations or laws that may require modification of these procedures. All shipping and receiving of radioactive material shall be performed in conjunction with, and with the approval of the Radiation Safety Office.
7. Implementation, maintenance, and modification to the Emergency Plan;
8. Implementation, maintenance, and modification to the Security Plan;
9. Implementation, maintenance, and modification to the Radiation Protection Plan. The Radiation Protection Plan shall include an ALARA plan as defined in ANSI/ANS-15.11-1993 (R2004);

Substantive changes to the previous procedures shall be made effective only after documented review by the review group and approval by the Director or designated alternate. Minor modifications to the original procedures that do not change their original intent may be made by Level 3 or higher, but the modifications must be approved by the Director or designated alternate. Temporary deviations from the procedures may be made by the responsible SENIOR REACTOR OPERATOR or higher individual present, in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported within 24 hours or the next working day to the Director or designated alternate.

6.5 Experiment Review and Approval

Approved EXPERIMENTS shall be carried out in accordance with established and approved procedures.

1. All new experiments or class of experiments shall be reviewed by the RSC as required by TS 6.2.3 and implementation approved in writing by the Director or a designated alternate.
2. Substantive changes to previously approved EXPERIMENTS shall be made only after review by the RSC and implementation approved in writing by the Director or a designated alternate. Minor changes that do not significantly alter the EXPERIMENT may be approved by Level 3 or higher. Changes to experiments shall meet the requirements in accordance with 10 CFR 50.59.

6.6 Required Actions

6.6.1 Action to be Taken in the Event of a Safety Limit Violation

In the event a SAFETY LIMIT is violated:

1. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC.
2. The event shall be reported to the Director who will report to the NRC as required in section 6.7.2.
3. An immediate report of the occurrence shall be made to the Chairman of the Reactor Safety Committee, and
4. A report, and any follow-up report, shall be prepared. The report shall describe the following:

(a) Applicable circumstances leading to the violation, including when known, the cause, and contributing factors;
(b) Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public; and
(c) Corrective action to be taken to prevent recurrence.

The report shall be reviewed by the Reactor Safety Committee and submitted to the NRC when authorization is sought to resume operation of the reactor.

6.6.2 Actions to Be Taken In The Event Of a Reportable Occurrence

In the event of a REPORTABLE OCCURRENCE, the following actions shall be taken:

1. Immediate action shall be taken to correct the situation and to mitigate the consequences of the occurrence.

2. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the Director.

3. The event shall be reported to the Director who will report to the NRC as required in section 6.7.2.

4. The Reactor Safety Committee shall investigate the causes of the occurrence at its next meeting. The report shall include an analysis of the causes of the occurrence, the effectiveness of corrective actions taken, and recommendations of measures to prevent or reduce the probability or consequences of recurrence.

6.7 Reports

6.7.1 Annual Operating Report

An annual report covering the operation of the reactor facility during the previous calendar year shall be submitted to the NRC before March 31 of each year providing the following information:

1. A brief narrative summary of results of reactor operations and surveillance tests and inspections required in section 4.0 of these Technical Specifications

2. A tabulation showing the energy generated in MW-hr for the year

3. A list of UNSCHEDULED SHUTDOWNS including the reasons therefore and corrective action taken, if any

4. A tabulation of the major maintenance operations performed during the period, including the effects, if any, on safe operation of the reactor, and the reason for any corrective maintenance required

5. A brief description of

   (a) Each change to the facility to the extent that it changes a description of the facility in the Final Safety Analysis Report
   (b) Review of changes, tests, and experiments made pursuant to 10 CFR Part 50.59

6. A summary of the nature and amount of radioactive effluents released or discharged to the environment

7. A description of any environmental surveys performed outside of the facility

8. A summary of exposure received by facility personnel and visitors where such exposures are greater than 25% of limits allowed by 10 CFR Part 20

9. Changes in facility organization

6.7.2 Special Reports

Notification shall be made within 24 hours by telephone and confirmed in writing by fax or similar conveyance to the NRC Headquarters Operation Center, and followed by a written report that describes the circumstances of the event and sent within 14 days to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, of any of the following:

1. A REPORTABLE OCCURRENCE,
2. Release of radioactivity from the site above allowed limits,
3. Exceeding the SAFETY LIMIT.

A written report within 30 days to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, of:

1. permanent changes in the facility organization involving Level 1 and Level 2 personnel; and
2. significant changes in the accident analysis as described in the Safety Analysis Report.

6.8 Records

Records of facility operations in the form of logs, data sheets, or other suitable forms shall be retained for the period indicated as follows:

6.8.1 Records to be Retained for a Period of at Least Five Years or for the Life of the Component Involved

1. Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc. which shall be maintained for a period of at least one year),
2. REPORTABLE OCCURRENCES,
3. Principal maintenance operations,
4. Surveillance activities required by the technical specifications,
5. Reactor facility radiation and contamination surveys where required by applicable regulations,
6. EXPERIMENTS performed with the reactor,
7. Fuel inventories, receipts, and shipments,
8. Approved changes in OPERATING procedures, and
9. Records of meeting and audit reports of the RSC.

6.8.2 Records to be Retained for at Least One License Cycle

Records of retraining and requalification of licensed operations personnel shall be maintained at all times the individual is employed or until the license is renewed.

6.8.3 Records to be Retained for the Lifetime of the Reactor Facility

1. Gaseous and liquid radioactive effluents released to the environs,
2. Off-site environmental monitoring surveys required by the technical specifications,
3. Radiation exposure for all personnel monitored,
4. Drawings of the reactor facility, and
5. Reviews and reports pertaining to a violation of the SAFETY LIMIT, the limiting safety system setting, or a limiting condition of operation.