UNIVERSITY OF
MARYLAND

Radiation Facilities
Mohamad Al-Sheikhly, Director

50-166

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

April 18, 2005

Enclosed please find the University of Maryland's response to the request for additional information as it pertains to the Technical Specifications for the Maryland University Training Reactor.

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

Executed on April 18, 2005

Alsheikhly

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

A020

DRAFT EDIT IN RESPONSE TO USNRC QUESTIONS IN REGARD TO MUTR RELICENSING DATED January 25, 2005

TECHNICAL SPECIFICATIONS

FOR THE

MARYLAND UNIVERSITY TRAINING REACTOR

License No. R-70

Docket No. 50-166

21 MAR 2000

Last Edit 4/13/2005

Formatted: Top: 0.6", Header distance from edge: 0.79", Footer distance from edge: 0.2"

TABLE OF CONTENTS

Formatted: Indent: Hanging: 0.5", Numbered + Level: 1 + Numbering Style: 1, 2, 3, ... + Start at: 1 + Alignment: Left + Aligned at: 0.25" + Tab after: 0.5" + Indent at: 0.5"

Deleted: 9
Deleted: 10
Deleted: 11
Deleted: 12
Deleted: 13
Deleted: 14
Deleted: 15
Deleted: 16
Deleted: 17
Deleted: 18
Deleted: 19
Deleted: 20
Deleted: 21
Deleted: 22

TABLE OF CONTENTS (CONT.)

Deleted: 23
Deleted: 24
Deleted: TABLE OF CONTENTS (CONT.)¶
Deleted: 25
Deleted: 26
Deleted: 27
Deleted: 28
Deleted: 29
Deleted: 30
Deleted: 31
Deleted: 32
Deleted: 33
Deleted: 34
Deleted: 35
Deleted: 36
Deleted: 37
Deleted: 38
Deleted: 39
Deleted: 40

TABLE OF CONTENTS (CONT.)

Deleted: TABLE OF CONTENTS (CONT.)¶

LIST OF TABLES

LIST OF FIGURES

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.

1.0 DEFINITIONS

1.1 ALARA (acronym for "as low as is reasonably achievable") means making every reasonable effort to maintain exposures to radiation as far below the dose limits in this part 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.

Deleted: ALARA - The ALARA program (As Low as Reasonably Achievable) is a program for maintaining occupational exposures to radiation and release of radioactive effluents to the environs as low as reasonably achievable.

1.2 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.

1. 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.

2. 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.

3. Channel Test - A channel test is the introduction of a signal into the channel to verify that it is operable.

1.3 CONFINEMENT - Confinement means a closure on the overall facility that controls the movement of air into it and out through a controlled path.

1.4 EXCESS REACTIVITY - Excess reactivity is that amount of reactivity that would exist if all reactivity control devices were moved to the maximum reactive condition from the point where the reactor is exactly critical (k_{eff}=1)

1.5 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 pool, on or in a beamport or irradiation facility, and that is not rigidly secured to a core or shield structure so as to be part of their design.

1. Routine Experiments - Routine Experiments are those which have been previously performed in the course of the reactor program.

2. Modified Routine Experiments - Modified routine experiments are those which have not been performed previously but are similar to routine experiments in that the hazards are neither greater nor significantly different than those for the corresponding routine experiments.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

3. Special experiments - Special experiments are those which are not routine or modified routine experiments.

1.6 EXPERIMENTAL FACILITIES - Experimental facilities are facilities used to perform experiments and include, for example, the beam ports, pneumatic transfer systems and any in-core facilities.

1.7 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.

1.8 Fuel Bundle - The 4-rod bundle consists of an aluminum bottom, 4 stainless steel clad fuel rods and aluminum top handle.

Formatted: Indent: Left: 0", Hanging: 0.5"

1.8 FUEL ELEMENT - A fuel element is a single TRIGA fuel rod.

1.9 FULL POWER - Full licensed power is defined as 250 kW.

1.10 INSTRUMENTED ELEMENT - An instrumented element is a special fuel element in which a sheathed chromel-alumel or equivalent thermocouple is embedded in the fuel.

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

1.12 LIMITING SAFETY SYSTEM SETTING - Limiting safety system settings (LSSS) for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions.

Deleted:

1.13 MEASURING CHANNEL - A measuring channel is the combination of sensor, interconnecting cables or lines, amplifiers, and output device, which are connected for the purpose of measuring the value of a variable.

1.14 MEASURED VALUE - The measured value is the value of a parameter as it appears on the output of a channel.

1.15 MOVEABLE EXPERIMENT - A movable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

1.16 ON CALL - A senior operator is available "on call" if the senior operator is either on the College Park campus or within 10 miles from the facility and can reach the facility within one half hour following a request.

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

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

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

1.19 REACTIVITY WORTH OF AN 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.

1.20 REACTOR CONSOLE SECURED - The reactor console is secured whenever all scrammable rods have been fully inserted and verified down and the console key has been removed from the console.

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

1.22 REACTOR OPERATOR - A reactor operator (RO) is an individual who is certified to manipulate the controls of the reactor.

1.23 REACTOR SAFETY SYSTEMS - 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. Manual protective action is considered part of the reactor safety system.

1.24 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, or

2. The following conditions exist:

   a. All control devices (3 control rods) are fully inserted, and
   b. The console key switch is in the off position and the key is removed from the lock, and
   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, and
   d. No experiments in or near the reactor are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment, or one dollar, whichever is smaller.

1.25 REACTOR SHUTDOWN - The reactor is shut down if it is subcritical by at least one dollar in the reference core condition and for all allowed ambient conditions with the reactivity worth of all installed experiments included.

1.26 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 zero (i.e., cold, clean, and critical).

Formatted: Font: 11 pt

Formatted: Font: 11 pt

Formatted: Indent: Left: 1", Hanging: 0.5", Tabs: Not at 0.2" + 0.3" + 1"

Deleted: The following conditions exist:¶
¶
a. The minimum number of neutron absorbing control devices are fully inserted or other safety devices are in shutdown position, as required by technical specifications, and¶
¶
b. The console key switch is in the off position and the key is removed from the lock, and¶
¶
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, and¶
¶
d. No experiments in or near the reactor are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment, or one dollar, whichever is smaller.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

1.27 REPORTABLE OCCURRENCE - A reportable occurrence is any of the following, which occurs during reactor operation:

1. Operation with actual safety-system settings for required systems less conservative than the Limiting Safety-System Settings specified in technical specifications 2.2.

2. Operation in violation of the Limiting Conditions for Operation established in the technical specifications.

3. A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns. (Note: Where components or systems are provided in addition to those required by the Technical Specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems specified or required performs their intended reactor safety function.)

4. An unanticipated or uncontrolled change in reactivity greater than one dollar.

5. Abnormal and significant degradation in reactor fuel, or cladding, or both, coolant boundary, or containment boundary (excluding minor leaks) where applicable which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

6. An observed inadequacy in the implementation of 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.

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

1.29 SAFETY CHANNEL - A safety channel is a measuring channel in the reactor safety system.

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

1.31 SCRAM TIME - Scram time is the elapsed time between the initiation of a scram signal by either automated or operator initiated action and the time required for the control rods to reach a fully inserted position into the core.

1.32 SECURED EXPERIMENT - A secured experiment is any experiment, experimental facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces with are normal to the operating environment of the experiment, or by forces that can arise as a result of credible malfunctions.

Deleted: number of components or systems specified or required perform

Deleted: SCRAM TIME - Scram time is the elapsed time between the initiation of a scram signal and a specified movement of a control or safety device.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

1.33 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.

1.34 SENIOR REACTOR OPERATOR - A senior reactor operator (SRO) is an individual who is certified to direct the activities of reactor operators.

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

1.36 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 value of shutdown reactivity includes the reactivity value of all installed experiments and is determined with the reactor at ambient conditions.

1.37 STANDARD CORE - A standard core is an arrangement of standard TRIGA fuel in the reactor grid plate.

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

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

1.40 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 which could adversely affect safe operation, not to include shutdowns which occur during testing or check-out operations.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTING

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 important parameter for TRIGA reactor is the UZrH fuel element temperature. This parameter is well suited as a single specification especially since it can be measured. A loss in the integrity of the fuel element cladding could arise from a build-up 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 in the cladding due to hydrogen pressure from the dissociation of ZrH_x will remain below the ultimate stress 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.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

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 provide a reactor scram to prevent the safety limit (fuel element temperature of 1000 °C) from being reached.

Specification

The limiting safety system setting shall be 175 °C as measured by the instrumented fuel element. The instrumented element may be located at any position in the core.

Deleted: 350

Basis

A Limiting Safety 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 thermocouple element 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 thermocouple element 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 thermocouple element 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, when the temperature in the thermocouple element 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 during steady state mode operation.

Deleted: 350

Deleted: 350

Deleted: 700

Deleted: 300

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

3.0 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 objectives are to ensure that the reactor can be controlled and shut down at all times and that the safety limits will not be exceeded.

Specifications

1. The excess reactivity relative to the cold critical conditions, with or without experiments in place shall not be greater than $3.50.

2. The shutdown margin shall not be less than $0.50.

3. Core configurations:

   a. The reactor shall only be operated with a standard core.

   b. No fuel may be inserted or removed from the core unless the reactor is subcritical by more than the worth of the most reactive fuel element.

   c. No control rods may be removed form the core unless a minimum of four fuel bundles are removed from the core.

Deleted: is

4. No operation with damaged fuel except to locate such fuel.

5. The reactivity coefficients for the reactor are:

Fuel:	-1.2 ¢/ °C
Moderator:	+3.0 ¢/ °C
Power:	-0.53 ¢/kW

6. The burnup 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.

Deleted: overconstrain

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.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

3. Specification 3.1.3 ensures that the operable core is similar to the core analyzed in the FSAR. 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 under the air and/or water sampling or under such case that the fuel has been exposed to temperature above 175 °C as measured on the instrumented fuel element. The criteria of the water and air sampling to determine a leaking fuel rod is considered positive if either sample is found to contain I-129 through I-135, Xe-135, Kr-85, 87 and Kr-88, Cs-135 and Cs-137, or Sr-89 through Sr-92.

5. The reactivity coefficients in Specification 3.1.5 ensure that the net reactivity feedback is negative.

6. General Atomic tests of TRIGA fuel indicate that keeping fuel element burnup below 50 % of the original ^{235}U loading will avoid damage to the fuel from fission product buildup.

3.2 REACTOR CONTROL AND SAFETY SYSTEMS

These specifications apply to reactor control and safety systems and safety-related instrumentation that must be operable when the reactor is in operation.

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 of each of the three standard 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, must be attached to a recording device sufficient for auditing of reactor operation history.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005
33

Bases

1. Specification 3.2.1 assures 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 assure 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.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005
33

Table 3.1: Reactor Safety Channels: Scram Channels

Scram Channel	Minimum Operable	Scram Setpoint
Reactor Power Level	2	Not to exceed 120 %
Fuel Element Temperature	1	Not to exceed 175 °C
Reactor Power Channel Detector Power Supply	2	Loss of power supply voltage to chamber
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

Table 3.2: Reactor Safety Channels: Interlocks

Interlock/Channel	Function
Log Power Level	Provide signal to period rate and minimum source channels
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 activated.
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.

Deleted: Countrate

Deleted: .

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

Table 3.3: Reactor Safety Channels: Scram Channel Bases

Scram Channel	Bases
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.
Reactor Power Channel Detector Power Supply	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	Assures that the reactor cannot be operated without a secure electric supply.
Rate of power change – Period	Assures 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.

Table 3.4: Reactor Safety Channels: Interlock Bases

Interlock/Channel	Bases
Log Power Level	This channel is required to provide a neutron detector input signal to the start up 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 by creating an artificial 1 cps signal with the Safety 1 trip test switch
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.

Deleted: Countrate

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005

3.3 COOLANT SYSTEMS

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.

Objective

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 reactor operations.

4. To maintain water clarity.

Specification

1. A minimum of 15 ft. of coolant shall be above the core.

2. A continuous radiation area monitor shall be mounted near the top of reactor pool tank. This monitor shall be able to scram the reactor, sound an audible alarm, and isolate the confinement building.

.

3 Conductivity of the pool water shall be no higher than 5×10^{-6} S/cm and the pH shall be between 5.0 and 7.5. Conductivity shall be measured before each reactor operation. pH shall be measured monthly, interval not to exceed six weeks.

Deleted: 3

Deleted: Gross gamma levels and isotopic analysis of the pool water shall be performed monthly, interval not to exceed six weeks.

Deleted: 4.

Bases

1. Specification 3.3.1 ensures that both sufficient cooling capability and sufficient biological shielding are available for safe reactor operation.

2. Specification 3.3.2 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.

Deleted: 3. Specification 3.3.3 combined with 3.3.2 ensures that any small leaks in a fuel element will not go unnoticed for an extended period of time.¶

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

3. A small 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 water cleanup 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 exposures during maintenance and operation.

Deleted: 4

3.4 CONFINEMENT

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.

Specifications

1. The reactor shall be housed in a closed room designed to restrict leakage. The closed room does not include the West balcony area.

2. Confinement shall be considered established when the doors leading from the reactor bay area leading into the balcony area on the top floor, and the reception area as well as the building exterior are secured.

3. Confinement must be established whenever the reactor is in an unsecured mode with the exception of the time that persons are physically entering or leaving the confinement area.

2

Deleted: . The minimum free air volume of the reactor room shall be 1.7 x 10^9 cm^3.

Bases

These specifications will dilute and delay the release of radioactive materials and ensure that the release conditions are similar to those assumed in the SAR.

3.5 VENTILATION SYSTEMS

Applicability

These specifications apply to the ventilation systems for the reactor building.

Objective

The objective of these specifications is to ensure that air exchanges between the reactor confinement building and the environment do not impact negatively on the general public.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005
33

Specifications

1. Air within the reactor building shall not be exchanged with other occupied spaces in the building.

2. All locations where ventilation systems exchange air with the environment shall have failsafe closure mechanisms.

3. Forced air ventilation shall automatically secure without operator intervention in such case that the radiation levels exceed a preset level as defined in facility procedures.

Formatted: Indent: Left: 0.5", Hanging: 0.5", Tabs: 1", List tab + Not at 0.75"

Formatted: Bullets and Numbering

Formatted: Indent: Left: 0.5", Hanging: 0.5", Tabs: 1", List tab + Not at 0.75"

Formatted: Bullets and Numbering

Bases

1. This specification ensures that radioactive releases inside the reactor building will not be transported to the remainder of the building.

2. This specification ensures that the reactor building can always be isolated from the environment.

3.6 RADIATION MONITORING SYSTEM

Applicability

This specification applies to the radiation monitoring information that must be available to the reactor operator during reactor operation.

Objective

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

Specification

1. The reactor shall not be operated unless the radiation area monitor channels listed in Table 3.5 are operable.

2. For a period of time not to exceed 48 hours for maintenance or calibration to the radiation monitor channels, the intent of specification 3.6.1 will be satisfied if they are replaced with portable gamma sensitive instruments having their own alarms or which shall be observable by the reactor operator.

3. The alarm set points shall be stated in a facility operating procedure. These set points 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.

4. The campus radiation safety organization shall maintain an environmental monitor at the MUTR site boundary.

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

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005
33

Table 3.5: Minimum Radiatiion Monitoring Channnels

Radiation Area Monitors	Function	Minimum Number Operable
Exhaust Radiation Monitor	Monitor radiation levels in Reactor Bay area at an Exhaust Fan location	A minimum of 1 of the 2 monitors shall be operable
Bridge Radiation Monitor	Monitor radiation levels in Reactor Bay area at the Reactor Bridge location	

Basis

The radiation area monitors provide 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.

The additional function of the radiation area monitor that monitors the reactor bay area is to warn personnel entering the building of high radiation levels if the pool water level should decrease to the level of inadequate biological shielding.

The intent of 3.6.3 and 3.6.5 are to ensure that facility does not lead to a dose to the general public greater than that allowed by 10 CFR Part 20.

Deleted: 5

Deleted: 6

3.7 LIMITATIONS ON EXPERIMENTS

Applicability

The 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 in-core 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, potentially explosive materials, and liquid fissionable materials shall be doubly encapsulated.
4. 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 less

Deleted: 1. Each non-secured experiment shall have a reactivity worth less than $1.00.¶
¶
2

Deleted: 3

Deleted: 4

Deleted: 5

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

than 25 mg may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be-less than the design pressure of the container. Total explosive material inventory in the reactor facility may not exceed 100 mg TNT or its equivalent.

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 activity such that if 100 % of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the airborne concentration of radioactivity within the reactor room averaged over a year would not exceed the limit of Table I of Appendix B of 10 CFR Part 20.

Deleted: 6

In calculations pursuant to 3.7.5 above, the following assumptions shall be used:

Deleted: 6

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 will escape.

b. If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99 % efficiency for 0.3 μm particles, at least 10 % of these particles can escape.

c. If an experiment fails and releases radioactive gases or aerosols to the reactor bay or atmosphere, 100 per cent of the radioactive gases or aerosols escape.

d. If an experiment fails that contains materials with a boiling point above 130° F (54° C), the vapors of at least 10 percent of the materials 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.

Deleted: 7

Bases

1. This specification is intended to provide assurance that the worth of a single unsecured 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 core temperature 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.

3. 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.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005
33

4. Double encapsulation is required to lessen the experimental hazards of some types of materials.

5. 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.

6. This specification is intended to reduce the likelihood that airborne activities in excess of the limits of Table 11 of Appendix B of 10 CFR Part 20 will be released to the atmosphere outside the facility boundary.

7. 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)

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

4.0 SURVEILLANCE REQUIREMENTS

INTRODUCTION

Surveillances shall be performed on a timely basis as defined in the individual procedures governing the performance of the surveillance. 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 until such time that the reactor is operable. In such case that any surveillance must be postponed, a written directive signed by the Facility 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.

4.1 REACTOR CORE PARAMETERS

Applicability

These specifications apply to the surveillance requirements for reactivity limits.

Objective

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

Specifications

The excess reactivity and 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.

Bases

The excess reactivity is required to determine the limits on incore experiments and the shutdown margin is calculated to ensure the reactor is capable of being shutdown under all conditions. The long-term changes in these parameters, excluding those caused by a change in the core configuration, are slow to develop and an annual schedule is sufficient to monitor them.

Deleted: shtudown

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

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

1. The reactivity worth of each standard control rod shall be determined annually, intervals not to exceed 15 months, and after each time the core fuel configuration is changed or a control rod is changed.

2. The control rod withdrawal and insertion speeds shall be determined annually, 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; intervals not to exceed 15 months, or whenever maintenance or repairs are made that could affect their drop time.

Deleted: annually,

4. All scram channels and power measuring channels shall have a channel test, including trip actions with safety rod release and specified interlocks 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 shall be calibrated annually, intervals not to exceed 15 months.

Deleted: an

Deleted: following any shutdown greater than 24 hours,

5. Operability 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, intervals not to exceed 15 months.

7. A visual inspection of the control rod poison sections shall be made biennially, intervals not to exceed 28 months.

8. A visual inspection of the control rod drive and scram mechanisms shall be made annually, 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 is 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 overall functional capability is maintained.

5. The surveillance test performed after maintenance or repairs to the reactor safety system as required by specification 4.2.5 ensures that the affected channel will perform as intended.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

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 COOLANT SYSTEMS

Applicability

These specifications apply to the surveillance requirements of the reactor coolant systems.

Objective

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

Specifications

1. Pool water gross gamma activity shall be determined monthly, at intervals not to exceed six weeks.

2. Pool water conductivity and pH shall be determined monthly, at intervals not to exceed six weeks.

3. The primary coolant level shall be verified before each reactor startup or daily during operations exceeding 24 hours.

Bases

1. Gross gamma activity measurements are conducted to detect fission product releases from damaged fuel element cladding.

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. Specification 4.3.3 ensures that sufficient water exists above the core to provide both sufficient cooling capacity and an adequate biological shield.

4.4 CONFINEMENT

Applicability

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

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

Objective

The objective of these specifications is to ensure that radioactive releases from the confinement can be limited.

Specifications

Prior to a reactor startup the isolation of the confinement building shall be visually verified.

Bases

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.

4.5 VENTILATION SYSTEM

Applicability

This specification applies to the reactor ventilation system.

Objective

The objective is to assure that provisions are made to restrict the amount of radioactivity released to the environment.

Specification

1.

1. The ability to secure the ventilation system shall be verified before each reactor startup.

Bases

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 (10 CFR Part 20) is exceeded.

Deleted: The reactor room air conditioning system shall be contained and can circulate air within the confines of the reactor room. Air and exhaust gases from the reactor room shall be released to the environment only through the ventilation exhaust system (Ventilation fan) or as a result of leakage around exit doors.

Deleted: 2

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

4.6 RADIATION MONITORING SYSTEMS AND EFFLUENTS

4.6.1 Monitoring Systems

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 each radiation area monitoring channel as required by Section 3.4 and to ensure that releases to the environment are kept below allowable limits.

Specifications

1. A channel calibration shall be made for each channel listed in Table 3.2 annually but at intervals not to exceed 15 months or whenever maintenance or repairs are made that could affect their calibration.

2. A channel test shall be made for each channel listed in Table 3.2 prior to starting up the reactor.

Bases

Specifications 4.6.1.1 and 4.6.1.2 ensure that the various radiation area monitors are checked and calibrated on a routine basis, in order to assure compliance with 10 CFR Part 20.

4.6.2 Effluents

Applicability

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

Objective

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

Specifications

1. Reactor building air samples shall be counted for gross gamma activity monthly, 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.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005
33

Bases

Specifications 4.6.2.1 and 4.6.2.2 ensure that the facility effluents comply with 10 CFR Part 20.

4.7 EXPERIMENTS

Applicability

This specification applies to experiments that operate with emergency systems or with connections to the reactor protection systems.

Objective

The objective of this specification is to ensure the operability of the reactor protection and emergency systems at all times.

Specifications

Any experiment which operates with emergency systems or with connections to the reactor protection systems shall have a channel check performed on those systems both daily and before any reactor startup when the experiment is being performed.

Basis

The specification in this part ensures that the reactor protection systems will operate as intended during experiments involving those systems.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

5.0 DESIGN FEATURES

5.1 SITE CHARACTERISTICS

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 site boundary shall consist of the outer walls of the reactor building and the area enclosed by the loading dock fence.

2. The restricted area shall consist of all areas interior to the reactor building including the west balcony and lower entryway.

3. The controlled area shall consist of the free air volume containing the reactor pool tank, the hot room, and the water room.

Bases

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

5.2 REACTOR COOLANT SYSTEM

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 connective water flow.

2. The pool water inlet pipe is equipped with a siphon break at the surface of the pool.

3. The pool water return (outlet) pipe shall not extend more than 50.8 cm (20 in) below the overflow outlet pipe when fuel in the core.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

Bases

Specification 5.2.1 is based on thermal and hydraulic calculations and operation of other TRIGA reactors that show that a core can operate in a safe manner at power levels up to 1500 kW with natural convection flow of the coolant.

Specification 5.2.2 and 5.2.3 ensures that the pool water level can normally decrease only by 50.8 cm (20 in) if the coolant piping where to rupture and siphon water from the reactor tank. Thus, the core will be covered by at least 4.57 m (15 ft) of water.

5.3 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 w/o uranium enriched to less than 20 % ^{235}U
2. Zirconium hydride atom ratio: nominal 1.5 - 1.8 hydrogen-to-zirconium, ZrH_x
3. Cladding: 304 stainless steel, nominal thickness of 0.508 mm (.020 in)
4. The standard TRIGA core shall consist of 93 standard TRIGA fuel elements and 3 control rods

Basis

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

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

5.4 FISSIONABLE MATERIAL STORAGE

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

<u>Objective</u>

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

<u>Specifications</u>

1. All fuel elements shall be stored either in a geometrical array where the k-effective is less than 0.8 for all conditions of moderation or stored in an approved fuel shipping container.

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. <u>When fuel is in storage in any area other than the grid plate, that area must be equipped with monitoring devices that both measure and record the radiation levels and temperature of the region surrounding the fuel.</u>

<u>Basis</u>

The limits imposed by Specifications 5.4.1 and 5.4.2 are conservative and assure safe storage.

Deleted: 2.

Formatted: Indent: Left: 0.5", Hanging: 0.5", No bullets or numbering

Formatted: Indent: Left: 0.5", Hanging: 0.5"

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

6.0 ADMINISTRATION

6.1 ORGANIZATION

The Maryland University Training Reactor (MUTR) is owned and operated by the University of Maryland, College Park. Its position in the university's structure is shown in Figure 6.1

The university will provide whatever resources are required to maintain the facility in a condition that poses no hazard to the general public or to the environment.

6.1.1 Structure

Figure 6.2 shows the MUTR organizational structure.

6.1.2 Responsibility

Responsibility for the safe operation of the reactor facility and radiological safety will rest in the Facility Director. The members of the organization chart shown in Figure 6.2 shall be responsible for safeguarding the public and facility personnel from undue radiation exposure and for adhering to all requirements of the operating license.

6.1.3 Facility Staff Requirements

1. The minimum staffing when the reactor is not in a secured condition 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 must be present in the facility or in the Chemical and Nuclear Engineering Building when the reactor is operating: 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 must be present or readily available on call. "Readily Available on Call" means an individual who (1) has been specifically designated and the designation known to the operator on duty, (2) keeps the operator on duty informed of where they may be rapidly contacted and the method of contact, and (3) is capable of arriving at the reactor facility within a reasonable amount of time under normal conditions. At no time 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

   b. Radiation safety personnel

   c. Licensed operators

Deleted: operating

Deleted: whenever the reactor is operating.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005
33

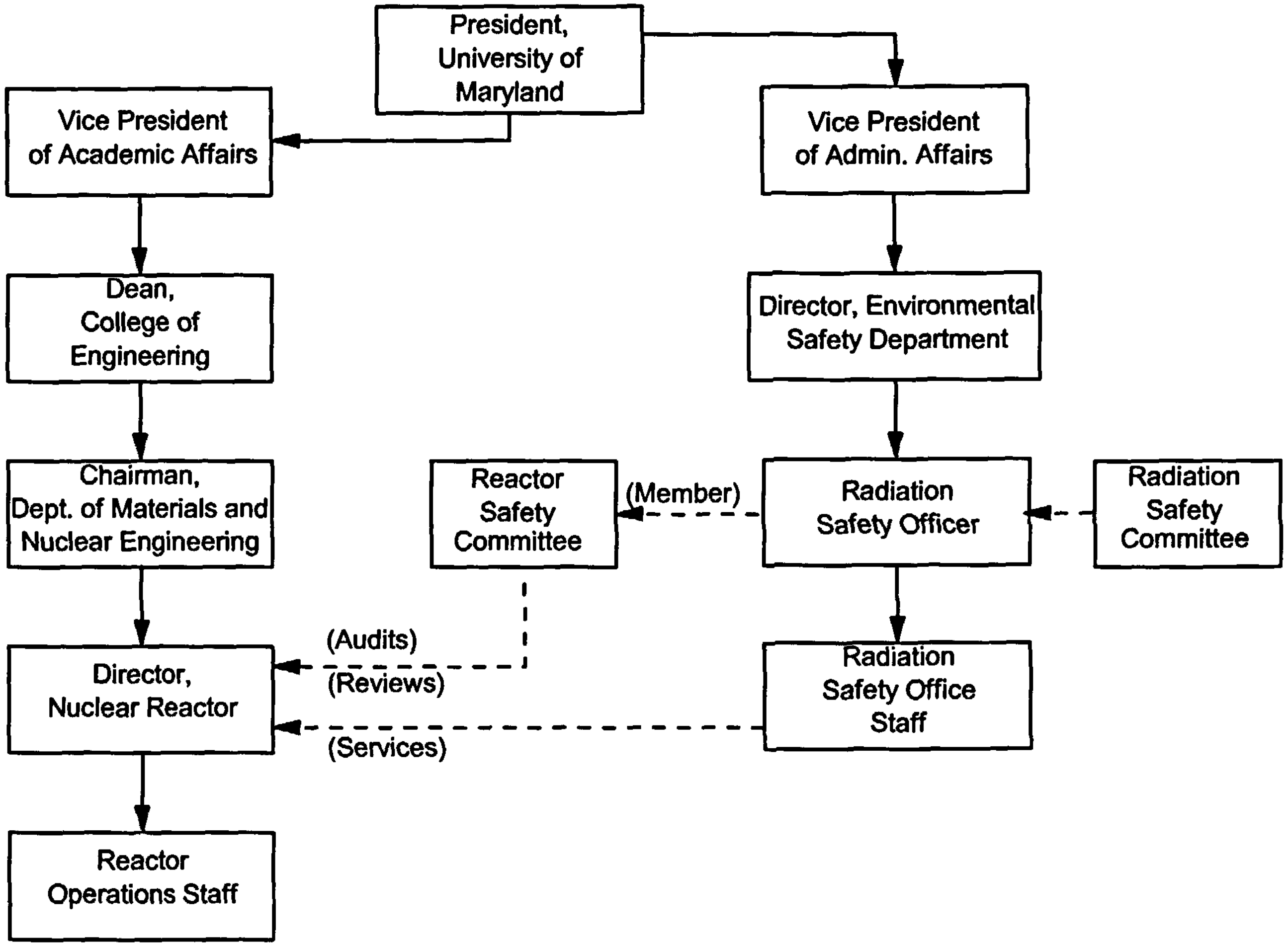


Figure 6.1: MUTR Position In University Of Maryland Structure

Formatted: Page Number, Font: 8 pt
Deleted: January 25, 2005¶

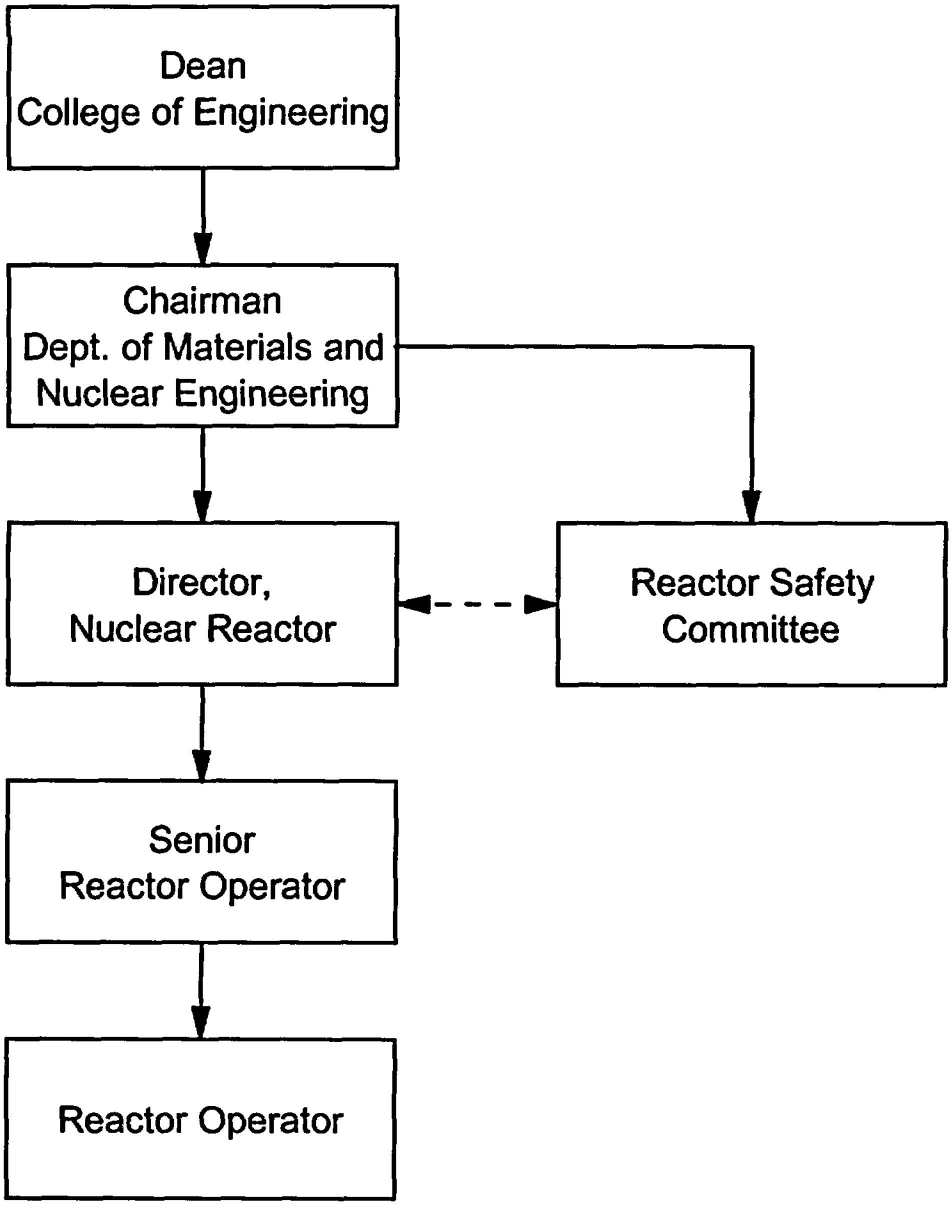


Figure 6.2: MUTR Organizational Structure

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

3. The following operations must be supervised by a senior reactor operator:

   a. Initial startup and approach to power following new fuel loading or fuel rearrangement

   b. When experiments are being manipulated in the core that have an estimated worth greater than $0.80

   c. Removal of control rods or fuel manipulations in the core

   d. Resumption of operation following an unplanned or unscheduled shutdown or any unplanned or unexpected significant reduction in power.

Deleted: (This requirement is waived if the shutdown is initiated by an interruption of electrical power to the plant.)

6.1.4 Selection and Training of Personnel

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

1. Responsibility - The Facility Director or his designated alternate is responsible for the training and requalification of the facility reactor operators and senior reactor operators. This selection shall be in conjunction with the guidelines set forth in ANSI/ANS 15.1 and 15.4.

Deleted: 2. Requalification Program¶
¶
a. Purpose - To insure that all operating personnel maintain proficiency at a level equal to or greater than that required for initial licensing¶
¶
b. Scope - Lectures, written examinations, and evaluated console manipulations will be used to insure operator proficiency is maintained¶

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 is appointed by and reports to the Chairperson of the Materials and Nuclear Engineering Department. 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 are the Facility Director and the Campus Radiation Safety Officer or that office's 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 must be from outside the Department of Materials and Nuclear Engineering.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

6.2.2 Reactor Safety Committee Charter And Rules

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

2. A quorum of the RSC must have at least three members and the Campus Radiation Safety Officer (or designated alternate). No more than two alternates may be used to make a quorum. MUTR staff members may 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.

6.2.3 Reactor Safety Committee Review Function

The RSC shall review the following:

1) Determinations that proposed changes in equipment, systems, test, experiments, or procedures do not involve an unreviewed safety question.
2) All new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance.
3) All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.
4) Proposed changes in technical specifications, license, or charter.
5) Violations of technical specifications, license, or charter. Violations of internal procedures or instructions having safety significance.
6) Operating abnormalities having safety significance.
7) Reportable occurrences listed under section 6.7.2.
8) Audit reports.

Deleted: ¶
1. Experiments referred to it by the Facility Director because of the degree of hazard involved or the unusual nature of the experiment¶
¶
<#>Reportable occurrences (see Section 6.6)¶
¶
<#>Violations of technical specifications or license¶
¶
<#>Proposed changes to the facility license, Emergency Plan, Technical Specifications, and experiments or changes made pursuant to 10 CFR Part 50.59¶
¶
<#>Operating procedures¶
¶
<#>Audit reports and inspection reports¶
¶
<#>Operating abnormalities having safety significance¶
¶
<#>Results of emergency drills¶

Formatted: Bullets and Numbering

6.2.4 Reactor Safety Committee Audit Function

1. An annual audit and review of the reactor operations will be performed by an outside individual or group familiar with research reactor operations. They shall submit a report to the Facility Director and the Reactor Safety Committee.

2. The following shall be reviewed:

   a. Reactor operators and operational records for compliance with internal rules, procedures, and regulations, and with license provisions

   b. Existing operating procedures for adequacy and accuracy

   c. Plant equipment performance and its surveillance requirements

   d. Records of releases of radioactive effluents to the environment

   e. Operator training and requalification

6.2.5 Audit Of ALARA Program

The Facility Director or his designated alternate shall conduct an audit of the reactor facility ALARA Program at least once per calendar year (not to exceed fifteen months). The results of the audit shall be presented to the RSC at the next scheduled meeting. This audit may occur as part of a review of the overall campus ALARA program.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

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 facility director will ensure that ALARA principles are followed during all facility activities.

6.4 OPERATING PROCEDURES

Written procedures, reviewed and approved by the Reactor Safety Committee, shall be in effect and followed for the following items prior to performance of the activity. The procedures shall be adequate to assure the safety of the reactor, but should not preclude the use of independent judgment and action should the situation require such:

1. Start-up, operation, and shutdown of the reactor
2. Installation or removal of fuel elements, control rods, experiments, experiment approval, and experimental facilities
3. Maintenance procedures that could have an effect on reactor safety
4. Periodic surveillance of reactor instrumentation and safety systems and area monitors as required by these Technical Specifications
5. 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.
6. Implementation, maintenance, and modification to the Emergency Plan.
7. Implementation, maintenance, and modification to the Security Plan.
8. Implementation, maintenance, and modification to the Radiation Protection Plan. The Protection Plan shall include an ALARA plan as defined in ANSI/ANS-15.11.

Substantive changes to the above procedures may be made with the approval of the Facility Director and the Reactor Safety Committee. This approval must be granted before the changes may be considered in effect. The only exception to this clause is in such a case where the delay in implementation would cause a credible risk to the public or the facility. If such a case exists as determined by the Facility Director, temporary approval may be granted by the Director but must be approved by the Reactor Safety Committee within thirty days. Temporary or minor changes to procedures shall be documented and subsequently reviewed by the Reactor Safety Committee at the next scheduled meeting. The Reactor Director shall have the power to approve minor changes such as phone number changes, typographical error correction or any other change that does not change the effectiveness or the intent of the procedure. It shall be considered sufficient approval and documentation when the Director forwards by electronic means to both the Radiation Safety

Formatted: Indent: Left: 0.5", Hanging: 0.5"

Formatted: Indent: Left: 0.39"

Formatted: Indent: Left: 0.5", Hanging: 0.5", Outline numbered + Level: 1 + Numbering Style: 1, 2, 3, ... + Start at: 5 + Alignment: Left + Aligned at: 0" + Tab after: 0.25" + Indent at: 0.25", Tabs: Not at 0.25"

Formatted: Font: 11 pt

Formatted: Indent: Left: 0.2"

Formatted: Indent: Left: 0.2", First line: 0.3"

Formatted: Indent: Left: 0.5", Hanging: 0.5"

Deleted: All such t

Deleted: .

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

Officer and the Chair of the Reactor Safety Committee. A copy of the transmission shall be filed with the appropriate procedure.

6.5 EXPERIMENT REVIEW AND APPROVAL

1. Routine experiments may be performed at the discretion of the duty senior reactor operator without the necessity of further review or approval.

2. Modified routine experiments shall be reviewed and approved in writing by the Facility Director, or designated alternate.

3. Special experiments shall be reviewed by the RSC and approved by the RSC and the Facility Director or desired alternate prior to initiation.

4. The review of an experiment listed in subsections 6.5.2 and 6.5.3 above, shall consider its effect on reactor operation and the possibility and consequences of its failure, including, where significant, chemical reactions, physical integrity, design life, proper cooling, interaction with core components, and any reactivity effects.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

6.6 REQUIRED ACTIONS

6.6.1 Action To Be Taken In Case Of Safety Limit Violation

In the event a safety limit is exceeded:

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 Reactor 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, Reactor Safety Committee, and reports shall be made to the NRC in accordance with Section 6.7.2 of these specifications, and

4. A report shall be prepared which shall include an analysis of the cause and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safety Committee for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

Formatted: Indent: Left: 0.5", Hanging: 0.5"

Deleted: 3

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

In the event of a reportable occurrence, as defined in section 1.27 of these Technical Specifications, the following actions will be taken:

1. Immediate action will 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 Reactor Safety Committee.

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

4. The Reactor Safety Committee will investigate the causes of the occurrence. The Reactor Safety Committee will report its findings to the NRC and Dean, College of Engineering. 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.

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

6.7 REPORTS

6.7.1 Annual Operating Report

A report summarizing facility operations will be prepared annually for the reporting period ending June 30. This report shall be submitted by September 30 of each year to the Director, Office of Nuclear Reactor Regulation, NRC, with a copy to the NRC Document Control Desk. The report shall include the following:

1. A brief narrative summary of results of surveillance tests and inspections required in section 4.0 of these Technical Specifications
2. A tabulation showing the energy generated in MW hr^{-1} 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 percent 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 to the NRC Operations Center, followed by a written report faxed within 14 days in the event of the following:

1. A reportable occurrence, as defined in Section 1.27 of this document
2. Release of radioactivity from the site above allowed limits
3. Exceeding the Safety Limit

The written report shall be sent to the NRC document control desk. The written report and, to the extent possible, the preliminary telephone or facsimilie notification shall:

Deleted: ·
Formatted: Superscript
Deleted: or telegraph
Deleted: Regional Administrator, Region 1,
Deleted: to the
Deleted: 0
Deleted: Director, Office of Nuclear Reactor Regulation,
Deleted: , with a copy to the Regional Administrator, Region I, NRC.
Deleted: telegraph
Formatted: Page Number, Font: 8 pt
Deleted: January 25, 2005¶
33

1. Describe, analyze, and evaluate safety implications

2. Outline the measures taken to ensure that the cause of the condition is determine

3. Indicate the corrective action taken to prevent repetition of the occurrence including chances to procedures

4. Evaluate the safety implications of the incident in light of the cumulative experience obtained from the report of previous failure and malfunction of similar systems and components

6.7.3 Unusual Event Report

A written report shall be forwarded within 30 days to the Director, Office of Nuclear Reactor Regulation, NRC, with a copy to the Regional Administrator, Region I, NRC, in the event of:

1. Discovery of any substantial errors in the transient or accident analysis or in the methods used for such analysis as described in the Safety Analysis Report or in the bases for the Technical Specifications

2. Discovery of any substantial variance from performance specifications contained in the Technical Specifications or Safety Analysis Report

3. Discovery of any condition involving a possible single failure which, for a system designed against assumed failure, could result in a loss of the capability of the system to perform its safety function

4. A permanent change in the position of Department Chair or Facility Director

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005¶
33

6.8 RECORDS

1. The following records shall be retained for a period of at least five years:

   a. Normal reactor facility operation and maintenance

   b. Reportable occurrences

   c. Surveillance activities required by Technical Specifications

   d. Facility radiation and contamination surveys

   e. Experiments performed with the reactor

   f. Reactor fuel inventories, receipts, and shipments

   g. Approved changes in procedures required by these Technical Specifications

   h. Minutes of the Reactor Safety Committee meetings

   i. Results of External Audits

2. Retraining and requalification records of current licensed operators shall be retained for at least one training cycle.

3. The following records shall be retained for the lifetime of the facility:

   a. Liquid radioactive effluents released to the environs

   b. Gaseous radioactive effluents released to the environs

   c. Radiation exposure for all facility personnel

   d. Radiation exposures monitored at site boundary

   e. As-built facility drawing

   f. Violation of the Safety Limit

   g. Violation of any Limited Safety System Setting (LSSS)

   h. Violation of any Limiting Condition of Operation (LCO)

4. Requirement 6.8.1 (a) above does not include supporting documents such as checklists, logsheets and recorder charts, which shall be maintained for a period of at least one year.

5. Applicable annual reports, if they contain any of the required information may be used as records in subsection 6.8.3 above.

Deleted: b

Deleted: c

Formatted: Page Number, Font: 8 pt

Deleted: January 25, 2005

TECHNICAL SPECIFICATIONS

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

Response:

TS 1.1 Presently reads:

" 1.1 ALARA - The ALARA program (As Low as Reasonably Achievable) is a program for maintaining occupational exposures to radiation and release of radioactive effluents to the environs as low as reasonably achievable."

It shall be amended to read:

" 1.1 ALARA (acronym for ``as low as is reasonably achievable'') means making every reasonable effort to maintain exposures to radiation as far below the dose limits in this part 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.

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).

Response:

TS 1.24.2 Presently reads:

2. The following conditions exist:

   a. The minimum number of neutron absorbing control devices are fully inserted or other safety devices are in shutdown position, as required by technical specifications, and

   b. The console key switch is in the off position and the key is removed from the lock, and

   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, and

   d. No experiments in or near the reactor are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment, or one dollar, whichever is smaller.

It shall be amended to state:

2. The following conditions exist:

   a. All control devices (3 control rods) are fully inserted, and

b. The console key switch is in the off position and the key is removed from the lock, and

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, and

d. No experiments in or near the reactor are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment, or one dollar, whichever is smaller.

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).

TS 1.31 Presently reads:

1.31 SCRAM TIME - Scram time is the elapsed time between the initiation of a scram signal and a specified movement of a control or safety device.

It shall be amended to state:

1.31 SCRAM TIME - Scram time is the elapsed time between the initiation of a scram signal by either automated or operator initiated action and the time required for the control rods to reach a fully inserted position into the core.

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.
There is no attempt to assure that the fuel is maintained below 500°C. The limit described in TS 2.1 is 1000°C.

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?
Assuming that the four bundles are not removed from the core, the MUTR is required to have a minimum of $0.50 shutdown margin. This margin is calculated with the most reactive rod stuck in the fully

withdrawn position and the maximum permissible experimental reactivity inserted. Therefore, the MUTR cannot achieve criticality or maintain criticality even with the four bundles in the core. By removing the four referenced bundles, regardless of their position, the reactor will remain Subcritical.

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.
There is no formal attempt to perform trend analysis on this data.

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.
See attachment 1:

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.
The beam ports and through tube interlocks are designed to ensure that the operator is aware of the condition of the plugs and ports. In such an event that the either of these tubes must be opened, the Director must give permission for operations that include open ports. The typical precautions include the installation of a mobile concrete shield, the area being roped off to deny access, and the installation or portable radiation monitoring as may be required.

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.
Added period scram as a TS required scram, modified table 3.1 and table 3.3.

87. TS Table 3.2, REACTOR SAFETY CHANNELS: INTERLOCKS. It is not clear what interlock is provided by the log power channel. Please clarify. TS's 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? Presently there is no limitation on the bulk temperature.

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? The procedure for fuel movement requires that the orientation and position of the fuel bundles be recorded for both bundle removal and replacement. This ensures that the core is reassembled in the same configuration after each disassembly/assembly cycle.

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. TS modified, please review.

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? SURVEILLANCE PROCEDURE 201; CONTROL ROD POISON SECTION

INSPECTION and SURVEILLANCE PROCEDURE 212; CONTROL ROD DRIVE MECHANISM INSPECTION define these surveillance requirements.

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?
See attachment 2; Sp210r12-AirSample

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.
Following the example of ANSI/ANS 15.1, the dotted lines signify communication lines where the solid lines indicate reporting lines.

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.

SAR 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.

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. See TS 6.4.5

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.

Please review question, the following statement does appear in TS 6.4: Substantive changes to the above procedures may be made with the approval of the Facility Director. All such temporary changes to procedures shall be documented and subsequently reviewed by the Reactor Safety Committee.

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 LCO' to be retained for the life of the facility. Please modify your TSs or justify not making these changes.

ATTACHMENT 1: RESPONSE TO QUESTION 84

Four scenarios for $0.30/s insertion will be considered for reactor response with absolutely no human intervention:

Low initial power, period scram at ≤5s period,
Low initial power, no period scram at ≤5s period,
High initial power, power scram at ≥120%,
And High initial power, no power scram at ≥120%.

The first scenario considered is with a low initial power (1mW is first considered) and with the period safety system is in working order, so that any power fluctuations with a period shorter than 5s are scrammed. This is a realistic scenario, as the period safety system is robust, and has been consistent in its operation. Assuming an initial power of 1mW, a $0.30/s reactivity insertion would finish inserting reactivity after 3.7 seconds (Appendix A), but would result in a period scram within 1.2 seconds (Appendix B). Assuming the maximum allowed 1s lapse in time from the scram signal occurs before the Control Rods are inserted, and assuming the full negative reactivity of the Control Rods is inserted instantly, the power would rise to maximum of 3.04W before dropping down to 1.54W in a prompt drop and then exponentially decaying with a negative 80s period.

The second scenario considered is with a low initial power and with an inoperative period safety system, so that any power fluctuations with a period shorter than 5s are not scrammed. Prompt fuel temperature increases as a result of a sharp increase of power would add controllability, and would lessen the increase in power. Assuming an initial power of 1mW, a $0.30/s reactivity insertion would finish inserting reactivity after 3.7 seconds (Appendix A), giving the reactor a minimum period of 0.00562s at 2.23s (Appendix B). At this point, the prompt temperature increases have effectively started to remove excess reactivity at a rate of $0.30/s. The power would continue to rise, which would continue to raise the fuel temperature, until the fuel temperature took away all of the excess reactivity. Eventually, the reactor power would level off. However, there are two independent safety systems, Safety 1 and Safety 2, which would scram the reactor before it got to 120% of full power (300kW). The power would reach 300kW at approximately 14.5s (Appendix C). Assuming the maximum allowed 1s lapse in time from the scram signal occurs before the Control Rods are inserted, and assuming the full negative reactivity of the Control Rods is inserted instantly, the power would rise to maximum of 303.15kW before dropping down to 153.6kW in a prompt drop and then exponentially decaying with a negative 80s period.

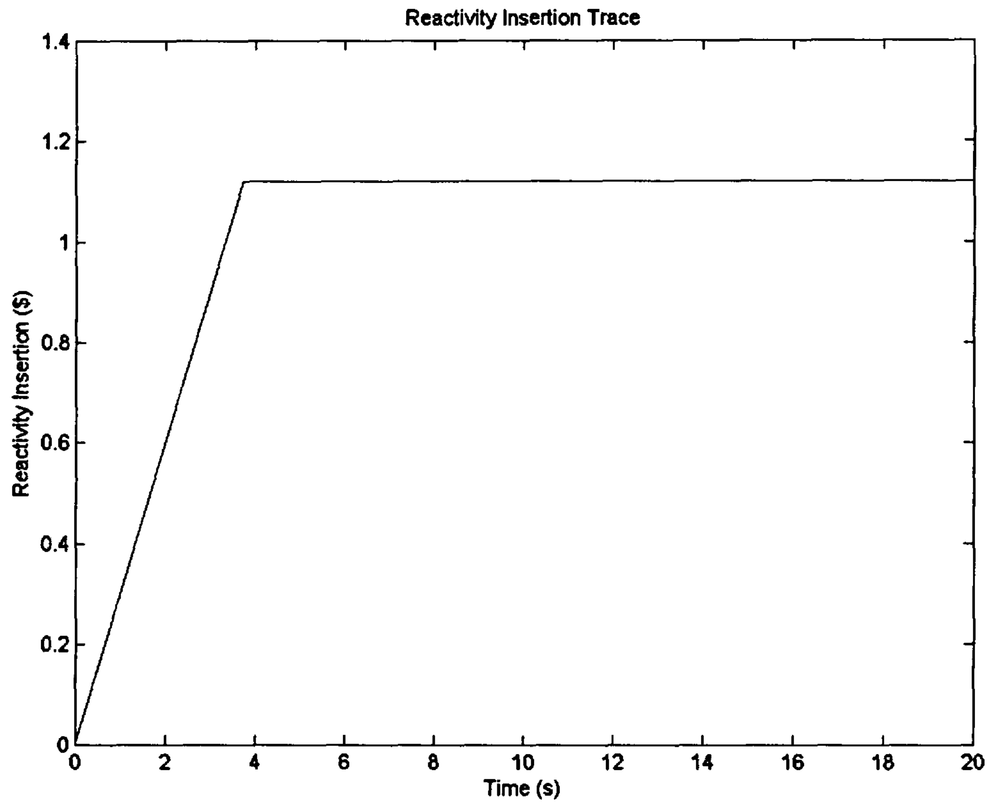
The first and second scenarios were considered with an initial power of 1mW. Simulations were done for several initial powers. According to the results, regardless of the initial power, as long as it is below the point of adding heat, the powers will converge to a power trace that levels off and then slowly increases as the moderator is heated up, adding excess reactivity (Appendix D).

The third scenario considered is with a high initial power, and with either of the Safety 1 or Safety 2 safety systems in working order, scramming any power that is greater that or equal to 120% of full power (300kW). This is also a realistic scenario, as both independent systems are robust and have been consistent in their operation. As the initial power increases, the initial fuel temperature also increases, which lowers the excess reactivity. Therefore, this scenario is more realistic than the second, as there is insufficient reactivity to obtain a period shorter than 5s. The period scram setpoint does not come into play in this scenario. The power simply converges to the low initial power trace, and then scrams at 120% of full power. The scram time will change depending on what the initial power is.

The fourth scenario is with a high initial power and with both Safety 1 and Safety 2 scram channels inoperable. Like the third scenario, there is insufficient excess reactivity to obtain a period shorter than 5s. In this scenario, the power is finally able to level off at the stable power of approximately 340kW, without being scrammed.

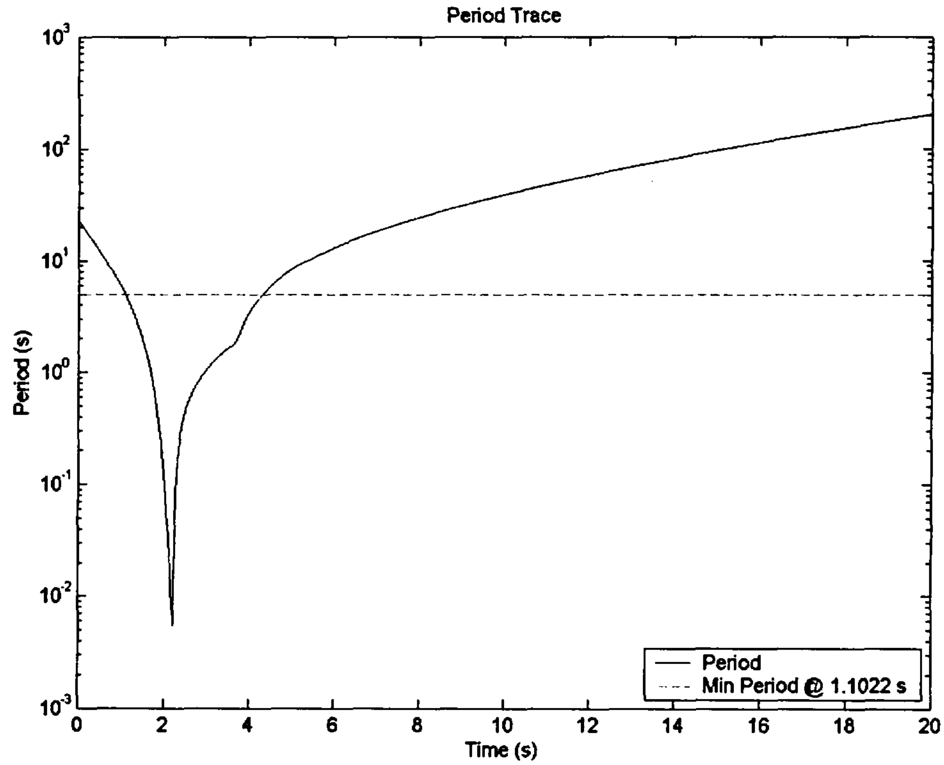
Only in this scenario is the fuel temperature relevant. The fuel temperature mirrors the power level at powers above the point of adding heat. In fact, without units or scales, it would be difficult to tell the difference between their graphs. At this point, the water surrounding the core is heated up, and reactivity is effectively inserted through the positive moderator reactivity coefficient. The reactor slowly rises in power until there is a fuel temperature scram at 175°C. This scram occurs at approximately 3130s after the beginning of the reactivity insertion (Appendix E).

Appendix A



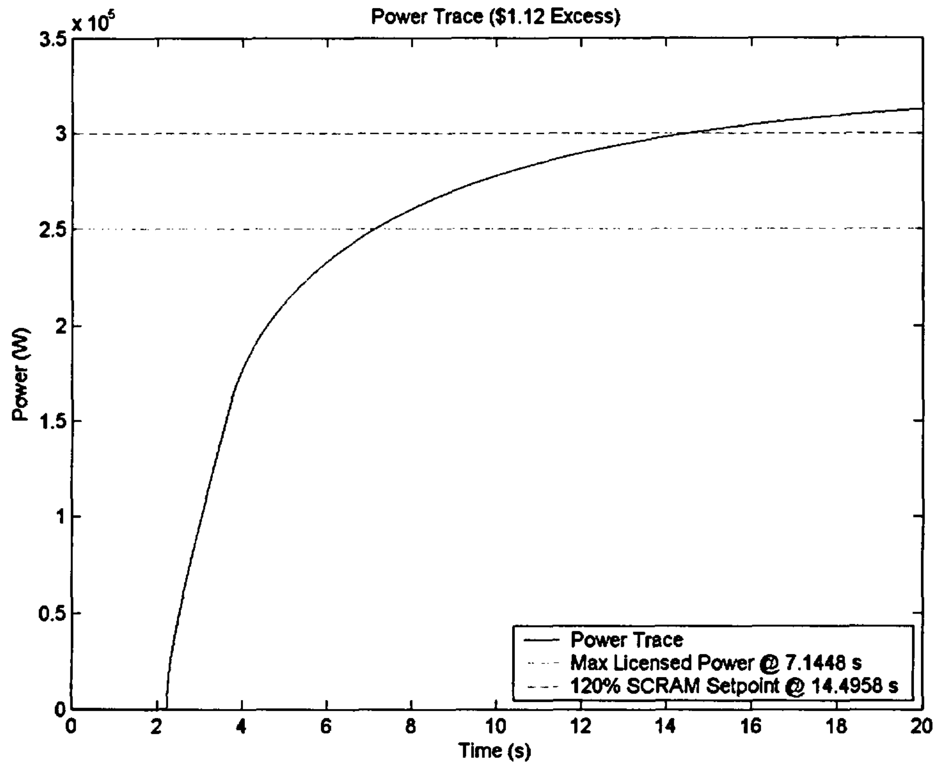
Reactivity Insertion Trace
Reactivity Insertion ($)
Time (s)
1.4
1.2
1
0.8
0.6
0.4
0.2
0
0
2
4
6
8
10
12
14
16
18
20

Appendix B

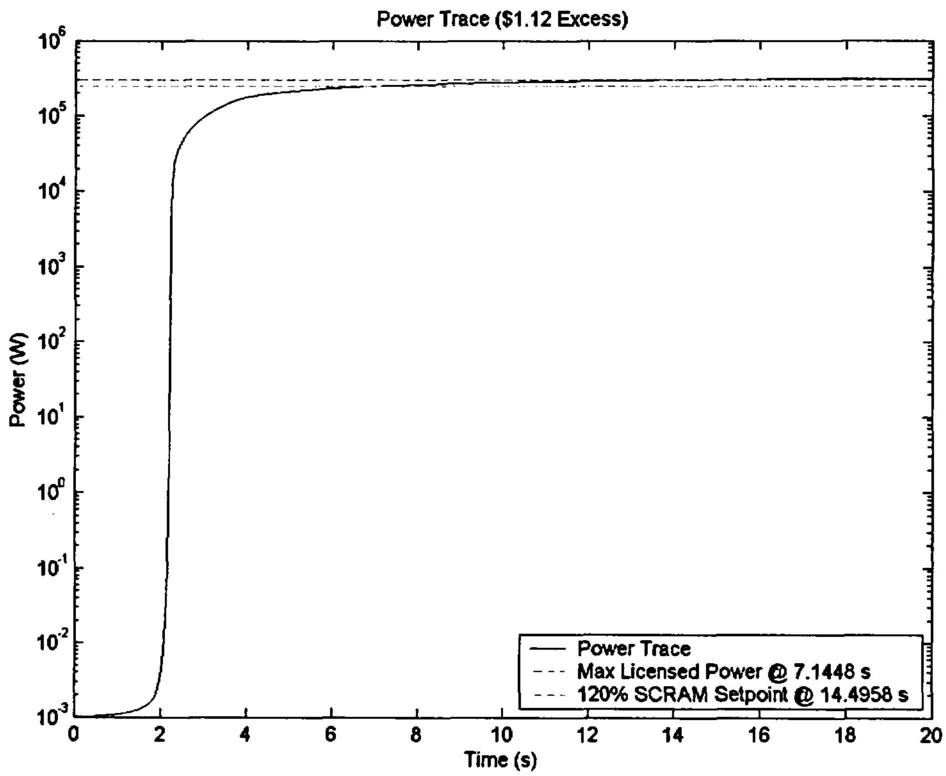


Period Trace
10^3
10^2
10^1
10^0
10^-1
10^-2
10^-3
Period (s)
0
2
4
6
8
10
12
14
16
18
20
Time (s)
Period
Min Period @ 1.1022 s

Appendix C

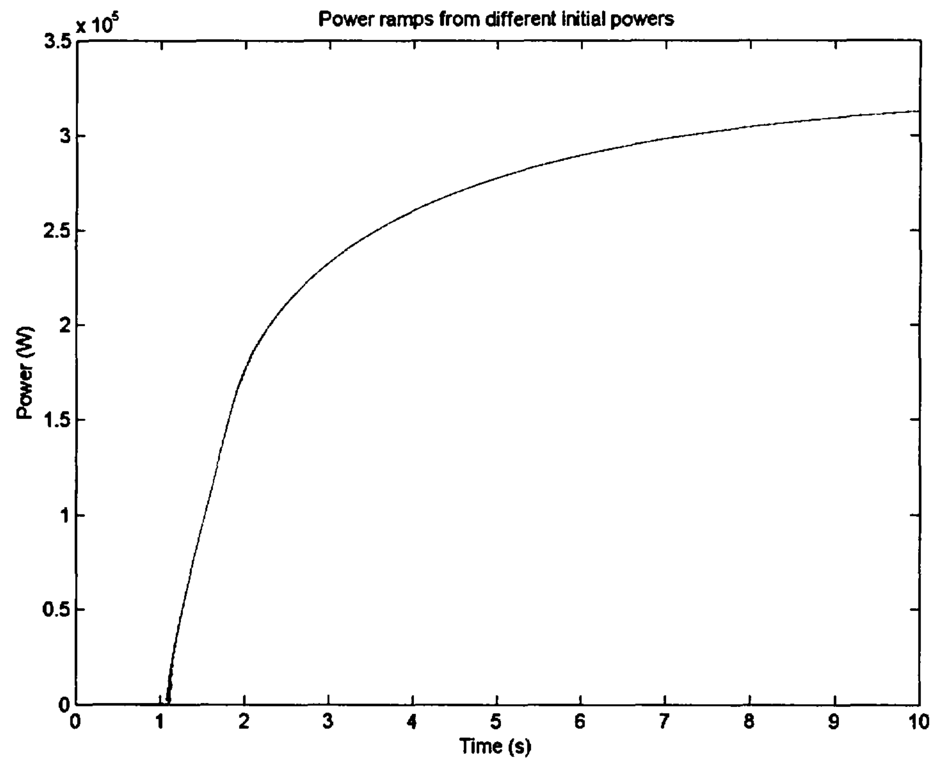


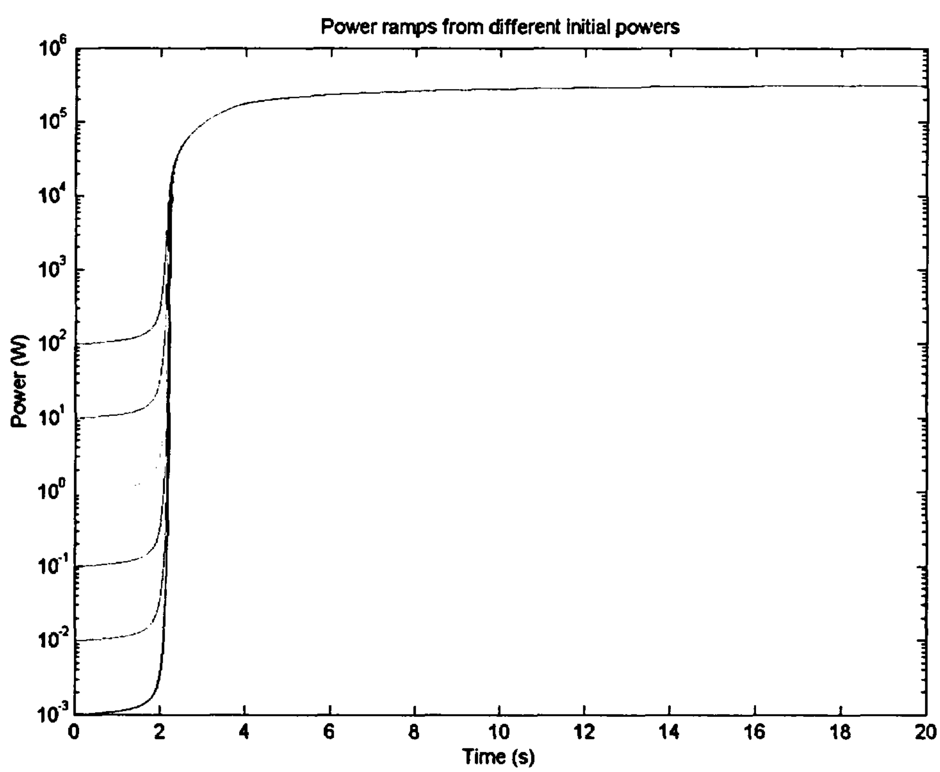
Power Trace ($1.12 Excess)
x 10^5
3.5
3
2.5
2
1.5
1
0.5
0
Power (W)
0
2
4
6
8
10
12
14
16
18
20
Time (s)
Power Trace
Max Licensed Power @ 7.1448 s
120% SCRAM Setpoint @ 14.4958 s



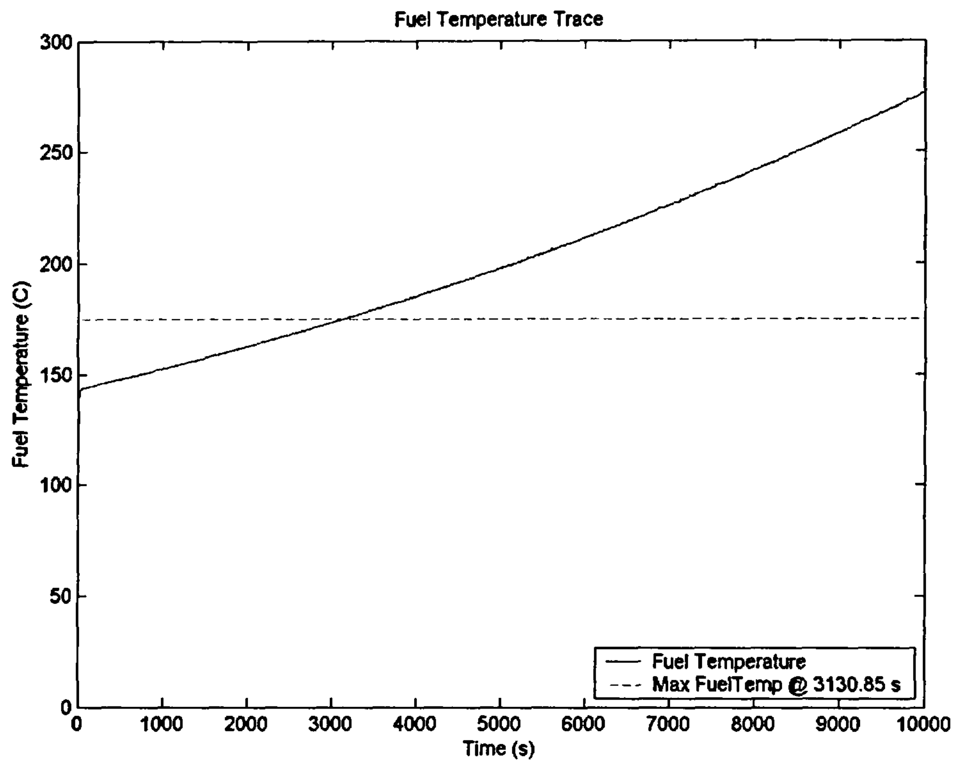
Power Trace ($1.12 Excess)
Power (W)
Time (s)
Power Trace
Max Licensed Power @ 7.1448 s
120% SCRAM Setpoint @ 14.4958 s

Appendix D

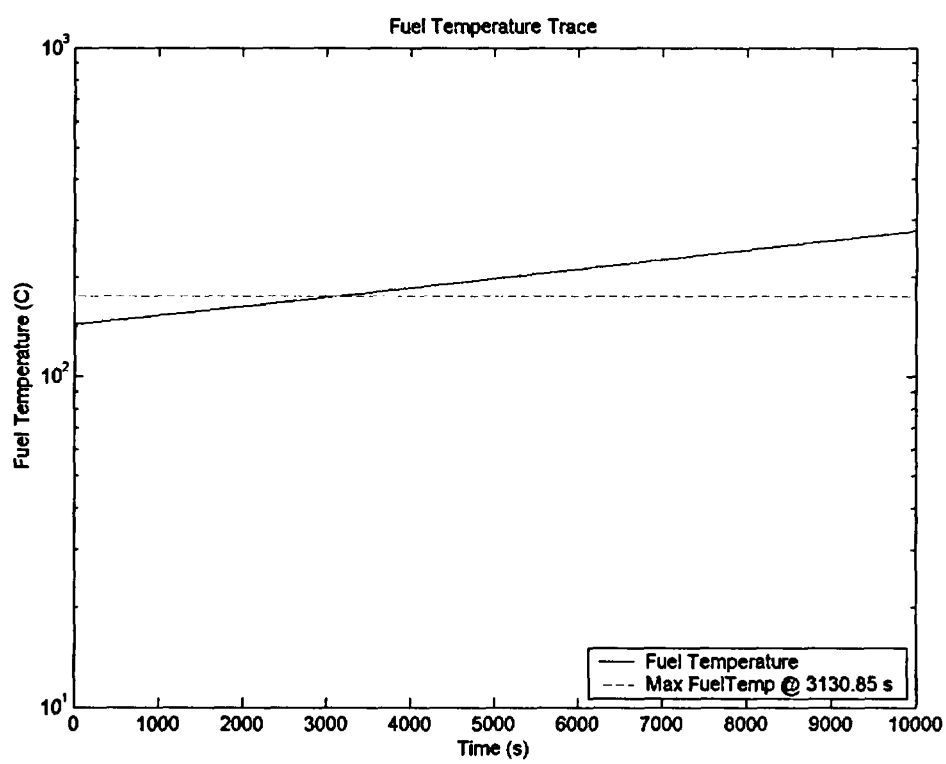

Power ramps from different initial powers
x 10^5
3.5
3
2.5
2
1.5
1
0.5
0
Power (W)
0
1
2
3
4
5
6
7
8
9
10
Time (s)


Power ramps from different initial powers
10^6
10^5
10^4
10^3
10^2
10^1
10^0
10^-1
10^-2
10^-3
Power (W)
0
2
4
6
8
10
12
14
16
18
20
Time (s)

Appendix E



Fuel Temperature Trace
Fuel Temperature (C)
Time (s)
Fuel Temperature
Max FuelTemp @ 3130.85 s
0
50
100
150
200
250
300
0
1000
2000
3000
4000
5000
6000
7000
8000
9000
10000



Fuel Temperature Trace
Fuel Temperature (C)
Time (s)
Fuel Temperature
Max FuelTemp @ 3130.85 s
10^1
10^2
10^3
0
1000
2000
3000
4000
5000
6000
7000
8000
9000
10000

ATTACHMENT 2: RESPONSE TO QUESTION 101

MARYLAND UNIVERSITY TRAINING REACTOR

SURVEILLANCE PROCEDURE 210

REACTOR ROOM AIR SAMPLE GAMMA RAY ANALYSIS

Revision 12

Nuclear Reactor Director Recommends Approval

______________________________ Date

Reactor Safety Committee Approval

______________________________ Chairperson ______________ Date

---

REACTOR ROOM AIR SAMPLE GAMMA RAY ANALYSIS

1.0 PURPOSE

The purpose of this procedure is to provide instructions for the taking of the air sample and the analysis.

2.0 PROCEDURE

2.1 Contact Radiation Safety for taking the air sample in the reactor room.

2.2 Record in the attached data sheet (Attachment SP 210-1) the required data when the sample is taken.

2.3 Submit the sample for analysis to Radiation Safety.

2.4 Record the results of the analysis on Attachment SP 210-1 or submit the printout of the results for the Reactor Director's approval.

3.0 RESULTS

3.1 If the results indicate normal background data then forward a signed copy of Attachment 1 to Radiation Safety and file the original in the reactor files.

3.2 If any of the following isotopes (fission product gasses) appear in the analysis:

I-129 through I-135
Xe-135
Kr-85, 87 and Kr-88

Notify the Reactor Director **immediately** as there may be a cladding failure on one or more fuel rods.

Attachment SP 210-1

REACTOR AIR SAMPLE ANALYSIS DATA SHEET

Date Sample Taken: ____________________

Date Analysis Performed: ____________________

Length of Time at Power Before Sample Taken: ____________

Date of Last Reactor Operation: ____________

Reactor Power Level of Last Operation: ____________

RESULTS: (or attach computer printout)

Isotopic Analysis

Radioisotope	Peak Found (KeV)	pCi/L	General Public MPC Level (pCi/l)	Remarks

Performed by : ______________________________

Reviewed by : ____

Reactor Director Date