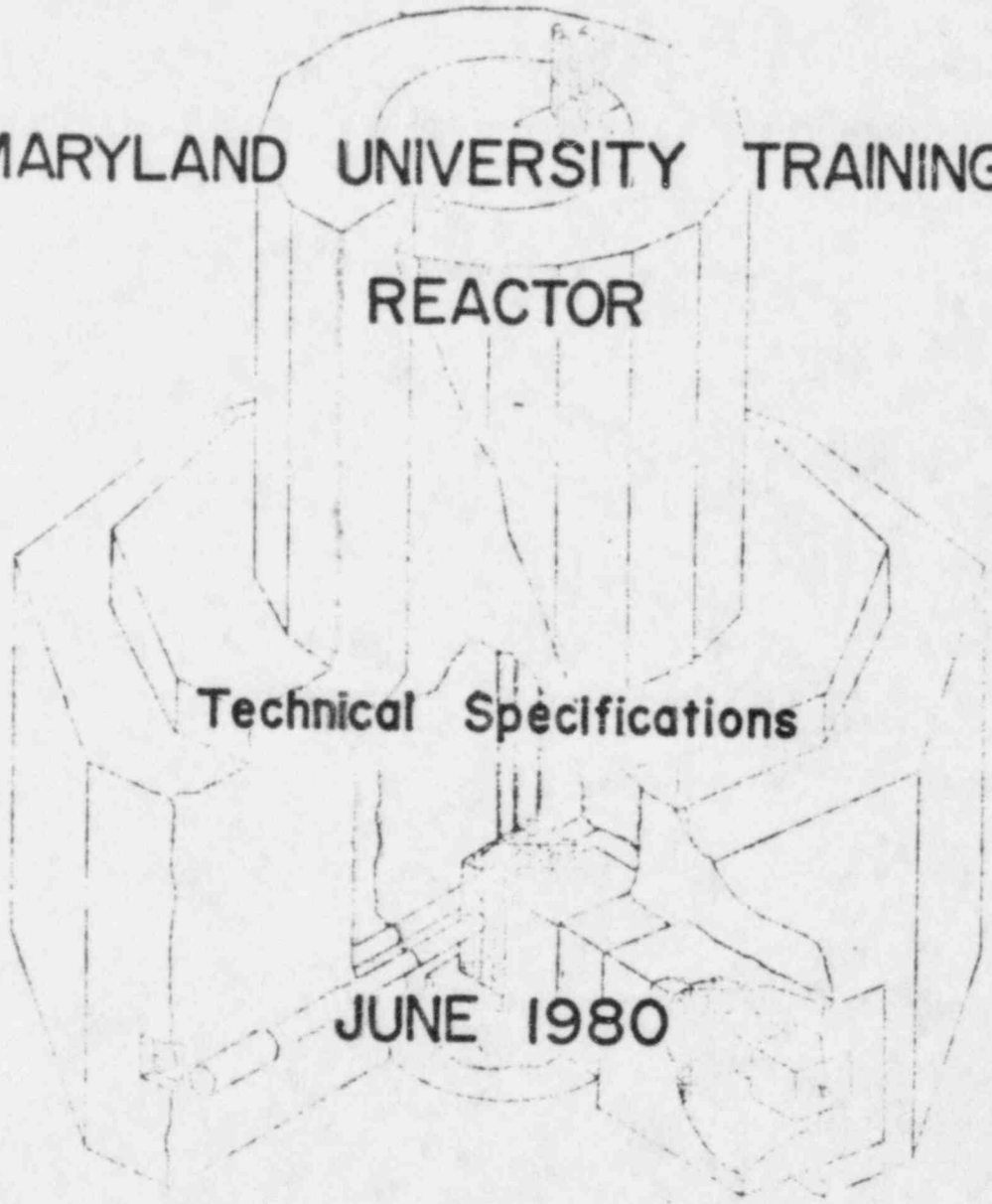
MARYLAND UNIVERSITY TRAINING REACTOR



Technical Specifications

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TECHNICAL SPECIFICATIONS

FOR THE

MARYLAND UNIVERSITY TRAINING REACTOR

April 1980

TECHNICAL SPECIFICATIONS

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TECHNICAL SPECIFICATIONS

Included in this document are the Technical Specifications and the "Basis" 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 Reactor Shutdown - The reactor is in a shutdown condition when sufficient control rods are inserted to assure that the reactor is subcritical by at least $1.00 of reactivity.

1.2 Reactor Secured - The reactor is secured when all the following conditions are satisfied:

a. The reactor is shut down,

b. Power to the control-rod magnets and actuating solenoids has been switched off and the key removed and under the control of a licensed operator or stored in a locked storage area, and

c. No work is in progress involving in-core fuel handling or refueling operations, maintenance of the reactor or its control mechanisms, or insertion or withdrawal of experiments from the core.

1.3 Reactor Operation - The reactor is in operation when it is not secured or a senior reactor operator is in charge of work in progress under 1.2.c above.

1.4 Operable - A system or component shall be considered operable when it is capable of performing its intended functions in a normal manner.

1.5 Operating - A system or component is operating when it is performing its intended functions in a normal manner.

1.6 Standard Control Rod - A standard control rod is a control rod having an electric motor drive and scram capabilities.

1.7 Flux Trap - A flux trap is any region within the core whose composition is modified to enhance the thermal neutron flux.

1.8 Cold Critical - The reactor is in the cold critical condition when it is critical with the fuel and bulk water temperatures both at ambient temperature.

1.9 Experiment - An experiment is (a) any device or material which is exposed to significant radiation from the reactor and is not a normal part of the reactor, or (b) any operation designed to measure reactor characteristics. (Normal control rod calibrations are not considered experiments.)

a. Routine Experiment - Routine experiments are those which have been previously performed in

the course of the reactor program.

b. 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.

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

1.10 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.

1.11 Reportable Occurrence - A reportable occurrence is any of the following which occurs during reactor operation:

a. Operation with any safety system setting less conservative than specified in Section 2.2., Limiting Safety System Settings;

b. Operation in violation of a Limiting Condition for Operation;

c. Malfunction of a required reactor or experiment safety system component which could render or threaten to render the system incapable of performing its intended safety function;

d. Any unanticipated or uncontrolled change in reactivity greater than [illegible].00;

e. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of a condition which could result in operation of the reactor outside the specified safety limits; and

f. Release of fission products from a fuel element.

1.12 Reactor Safety Systems - Reactor safety systems are those systems, including their associated input circuits, which are designed to initiate a reactor scram (i.e. magnet release of the control rods) for the primary purpose of protecting the reactor or to provide information which requires manual protective action to be initiated.

1.13 Experiment Safety System - 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.

1.14 Standard Thermocouple Fuel Element - A standard thermocouple fuel element is a fuel element of standard shape and of known and specified composition containing

sheathed thermocouples imbedded near the axial and radial center of the fuel element.

1.15 Measured Value - The measured value is the magnitude of that variable as it appears on the output of a measuring channel.

1.16 Measuring Channel - A measuring channel is the combination of sensor, interconnecting cables or lines, amplifiers, and output devices which are connected for the purpose of measuring the value of a variable.

1.17 Channel Check - A channel check is a qualitative verification of acceptable performance by observation of channel behavior. The verification shall include comparison of the channel with other independent channels or methods of measuring the same process variable.

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

1.19 Channel Calibration - A channel calibration consists of comparing a measured value from the measuring channel with a corresponding known value of the parameter so that the measuring channel output can be adjusted to respond, with acceptable accuracy, to known values of the measured variable.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability

This specification applies to the maximum fuel element temperature and steady state reactor power level which are safety limits for reactor operation.

Objective

The objective is to define a fuel element temperature and reactor power level that will provide confidence that no fuel element cladding damage will result.

Specifications

The Safety Limits for the MUTR are as follows:

a. Fuel element temperature - 1000°C;

b. Reactor steady state power level - 1000 kW.

Basis

A loss in the integrity of the fuel element cladding could arise from an excessive build-up of pressure between the fuel-moderator and the cladding. The pressure is caused by the presence of fission product gases and the dissociation of the hydrogen and zirconium in the fuel moderator. The magnitude of this pressure is determined by the fuel-moderator temperature. The safety limit for the stainless steel clad high hydride ($ZrH_{1.7}$) fuel element is based on data (SAR pages III-56 through III-59 for

the Illinois Advanced TRIGA) which indicates that the stress in the cladding (due to the hydrogen pressure from the dissociation of zirconium hydride) will remain below the yield stress (0.2% offset) provided the temperature of the fuel does not exceed 1000°C.

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. It has been shown by analysis and by measurements on other TRIGA reactors that a power level of 1000 kW corresponds to a peak fuel temperature of approximately 400°C. Thus, a Safety Limit on power level of 1000 kW provides an ample margin of safety for operation.

2.2 Limiting Safety System Settings

Applicability

This specification applies to the limiting trip settings for fuel element temperature and power level channels.

Objectives

The objective is to prevent the safety limits from being exceeded.

Specifications

The Limiting Safety System Settings for the MUTR are as follows:

a. Fuel element temperature channel - 400°C;

b. Reactor power level channels - 300 kW.

Basis

A Limiting Safety System Settings of 400°C provides a safety margin of 600°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 indicate that, for this case, 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 trip settings of 400°C, the true temperature at the hottest location would be no greater than 800°C providing a margin to the safety limit of at least 200°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.

Calculations and measurements for similar TRIGA reactors indicate that at 300 kW, the peak fuel temperature in the core will be less than approximately 300°C so that the limiting power level setting provides an ample safety margin to accommodate errors in power level measurement and anticipated operational transients.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity

Applicability

These specifications apply to the reactivity condition of the reactor, and the reactivity worths of control rods and experiments.

Objective

The objective is to assure that the reactor can be shut down at all times and to assure that the fuel temperature safety limit will not be exceeded.

Specifications

The reactor shall not be operated unless the following conditions exist:

a. The shutdown margin referred to the cold-critical xenon-free condition, with the highest worth rod fully withdrawn, is greater than $0.50;

b. The rate of reactivity insertion by control rod motion is not greater than $0.30 per second;

c. Any experiment with a reactivity worth greater than $1.00 is securely fastened so as to prevent unplanned removal from or insertion into the reactor;

d. The excess reactivity (i.e. the reactivity available with control rods removed from the cold critical, xenon-free condition with or without experiments in place) is less than 2.5% $\Delta k/k$;

e. When multiple experiments are being conducted, the reactivity worth of an individual experiment inserted into the reactor is not more than $1.00;

f. The drop time of each standard control rod from the fully withdrawn position to 90 percent of full reactivity insertion is less than one second.

Basis

The shutdown margin required by specification 3.1.a is necessary so that the reactor can be shut down from any operating condition and remain shut down after cooldown and xenon decay even if one control rod should remain in the fully withdrawn position.

Specification 3.1.b assures that power increases caused by rod motion will be terminated by the reactor safety system before the fuel temperature safety limit is exceeded.

It is assumed that the worst reactivity insertion accident from unrestrained motion of the control rods is initiated from a condition corresponding

to reactor startup, between 1 and 100 milliwatts, with a subcritical condition corresponding to the source level. Then 30¢/sec of reactivity is inserted continuously and it is assumed that the reactivity insertion continues until the power level scram is tripped. A further delay of 0.1 second is used until the control rods begin to remove reactivity at the rate of $3/sec, until the rods are inserted. Assuming no thermodynamic feedback occurs (making the calculation quite conservative) it is found that 1.7 MW-sec of energy is produced by excursion, raising the fuel temperature by 20°F in the peak fuel flux location of the core. This temperature rise is far below the allowable rise to the fuel damage point starting from ambient conditions at startup. Specification 3.1.c is based on Section 8.5 of the University of Utah SAR which indicates that as much as $3.00 reactivity could be inserted in a pulse from a power level of 3 MWt without violation of the fuel temperature safety limit. By restricting each unsecured experiment to a reactivity worth of one dollar, an ample margin is provided to allow for uncertainties in the information and the uncertainty in the worth of an experiment.

Specification 3.1.c through 3.1.e are intended to provide additional margins between those values of reactivity changes encountered during the course

of operation involving experiments and those values of reactivity which, if exceeded, might cause a safety limit to be exceeded.

Specification 3.1.f is intended to assure prompt shutdown of the reactor in the event a scram signal is received.

3.2 Reactor Instrumentation

Applicability

This specification applies to the information which must be available to the reactor operator during reactor operation.

Objective

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

Specification

The reactor shall not be operated unless the measuring channels described in the following table are operable.

Measuring Channel	Minimum Number Operable
Fuel Element Temperature	1
Reactor Power Level	2
Startup Count Rate	1
Area Radiation Monitor	2(a)

Basis

The fuel temperature displayed at the control console gives continuous information on the process variable which has a specified safety limit.

The neutron detectors assure that measurements of the reactor power level are adequately covered at both low and high power ranges. The specifications on the reactor power level indication are included in this section since the power level is closely related to the fuel temperature and is a process variable having a specified safety limit.

The radiation 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.

(a) For periods of time for maintenance to the radiation monitoring systems, the intent of this specification will be satisfied if the installed systems are replaced with protable gamma-sensitive instruments having their own alarms or which shall be kept under visual observation.

3.3 Reactor Safety System

Applicability

This specification applies to the reactor safety system channels.

Objective

The objective is to require the minimum number of reactor safety system channels that must be operable for safe operation.

Specification

The reactor shall not be operated unless the safety systems described in Table I are operable.

Basis

The fuel temperature scrams provide the protection to assure that, if a condition results in which the Limiting Safety System Setting is exceeded, an immediate shutdown will occur to keep the fuel temperature below the safety limit. The power level scrams are provided as redundant protection against abnormally high fuel temperature to assure that the reactor operation stays within the licensed limits. The equivalent operation with scrams at or below 120% of full power or 300 kW assures that the reactor operation will terminate well below that power level safety limit. The manual scrams allow the operator to shut down the system if an unsafe or abnormal condition occurs. In the event of failure

Table I

Safety Channel	Number Operable	Function
Fuel Element Temp.	1	Scram at 400°C
Reactor Power Level	2	Scram at 120%
Manual Scram Bar	1	Scram when actuated
Reactor Power Level High Voltage Supplies	2	Scram on 10% or greater loss of supply voltage
Console Power supply	1	Scram on loss of electrical power
Startup count rate interlock	1	Prevent control rod withdrawal when neutron count rate is less than one count per second
Control rod withdrawal interlocks	1	Prevent manual withdrawal of more than one control rod simultaneously
Pool level*	1	Alarm at 90% normal operating level

*For periods of time for maintenance to the pool level monitoring system, the intent of this specification will be satisified if the installed system is replaced by a portable level sensing device having its own alarm.

of the console power supply, the console power supply scram provides that operation will not continue without adequate instrumentation.

The interlock to prevent startup of the reactor with less than one count per second, or equivalent reactor power, indicated on the startup power level channel assures that sufficient neutrons are available for proper operation of the startup power level channel. The interlock to prevent simultaneous withdrawal of more than one control rod by manual operation assures that the addition of reactivity will be properly controlled. The pool level alarm is intended to alert the operator of any significant decrease in pool level.

3.4 Radiation Monitoring System

Applicability

This specification applies to the radiation monitoring information which 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 reactor operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless two of the

following monitoring channels are operable.

Radiation Monitoring Channel	Function
Exhaust Radiation Monitor	Monitor radiation levels at exhaust fan located on West wall. Scram reactor at 10 mrem/hour.
Bridge Radiation Monitor	Monitor radiation levels at entrance to reactor pool top. Scram at 50 mrem/hour.
Water Room Monitor	Monitor radiation level of particulate filter and deionizer in primary loop. Scram at 20 mrem/hour.

Basis

Settings for the radiation monitors are established from experience gained in full power reactor operations since 1974. The exhaust monitor usually measures direct radiation from the core and residual nitrogen-16 activity. In the event of cladding rupture the most likely isotopes would be the radioactive isotopes of Xenon and Krypton. Iodine would be captured in the pool water. A setting of 10 millirem/hour for the scram point on the exhaust monitor would ensure against prolonged operation with damaged fuel cladding. The bridge radiation monitor is so located

that it is influenced primarily by direct gamma radiation and nitrogen-16 during normal operation. Again, it would provide for a scram in the event of failed fuel due to the increase at the top of the pool due to the release of radioactive kryptons and xenons. A setting of 5.0 mr/hr is sufficient to permit operation at full licensed power and yet low enough to alarm in the event of a significant fuel leak.

The water room monitor usually indicates levels of less than one millirem/hour. During full power operation after several days without operating, the ionic content of the water is high enough to give increased levels in the demineralizer. A setting of 20 mr/hr is sufficient to permit full power operation. The levels in areas occupied by personnel are generally less than one millirem/hour.

The primary gaseous release during normal operation is nitrogen-16. Its short half-life and the height at which it is released mitigate against any dose to the general population. The monitor settings are such that any release due to failed fuel will be as low as reasonably achievable.

3.5 Limitations on Experiments

Applicability

This specification applies to experiments installed in the reactor and its experimental facilities.

Objective

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

Specifications

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

a. Non-secured experiments shall have reactivity worths less than 1 dollar.

b. The reactivity worth of any single experiment shall be less than 2 dollars.

c. Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 milligrams shall not be irradiated in the reactor of experimental facilities. Explosive materials in quantities less than 25 milligrams 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.

d. 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 atomsphere, the airborne concentration of radioactivity averaged over a year would not exceed the limit of Appendix B of 10 CFR Part 20.

e. In calculations pursuant to d. above, the following assumptions shall be used:

1) 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.

2) If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of particulates will escape; and

3) For material whose boiling point is above 130°F and where vapors formed by boiling this material can escape only through an undistributed column of water above the core, at least 10% of these vapors can escape.

f. 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 millicuries.

g. If a capsule fails and releases materials which could damage the reactor fuel or structure by corrosion or other means, removal and physical inspection shall be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective

action taken shall be reviewed by the Director or his designated alternate and determined to be satisfactory before operation of the reactor is resumed.

Basis

a. This specification is intended to provide assurance that the worth of a single unfastened 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 suddenly inserted.

b. 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 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.

c. This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials.

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

e. The 5 millicurie 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 10CFR Part 20 for an unrestricted area.

f. Operation of the reactor with the reactor fuel or structure damaged is prohibited to avoid release of fission products.

3.6 Fuel Element Integrity

Applicability

This specification applies to the dimensional and structural integrity of the fuel elements.

Objective

The objective is to assure that the reactor will not be operated with defective fuel elements installed.

Specification

The reactor shall not be operated with damaged fuel. A fuel element shall be considered damaged and must be removed from the core if:

a. In measuring the transverse bend, its sagitta exceeds 0.125 inch over the length of the cladding;

b. In measuring the elongation, its length exceeds its original length* by 0.250 inch; and

c. A clad defect exists as indicated by release of fission products.

Basis

The limit of transverse bend has been shown to result in no difficulty in disassembling the core. Analysis of the removal of heat from touching fuel elements shows that there will be no hot spots resulting in damage to the fuel caused by this touching. Experience with TRIGA reactors has shown that fuel element bowing that could result in touching has occurred without deleterious effects. The elongation limit has been specified to assure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment and to assure adequate coolant flow through the top grid plate.

4.0 Surveillance Requirements

4.1 Fuel

Applicability

This specification applies to the surveillance requirement for the fuel elements.

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* Original length shall be deemed to be the length of each respective fuel element as compared to a standard dummy element.

Objective

The objective is to assure fuel integrity at a minimum risk to the cladding during the inspection process.

Specifications

a. Four (4) fuel element clusters will be removed from a major axis of the reactor core and inspected visually for damage or deterioration at intervals not to exceed two years.

b. All fuel elements shall be measured for length and bend if the inspection in (a) shows any deterioration of fuel elements.

c. Fuel elements in the hottest assumed location as well as representative elements in each of the rows shall be measured for possible damage in the event that there is indication that fuel temperature greater than the limiting safety system setting may have been exceeded.

Basis

Visual inspection of the TRIGA fuel has been shown adequate to insure fuel element integrity through a long history of standard operation in the steady state mode.

4.2 Control Rods

Applicability

This specification applies to the surveillance requirements for the control rods.

Objective

The objective is to assure the integrity of the control rods.

Specifications

a. The reactivity worth of each control rod shall be determined annually, but at intervals not to exceed 15 months.

b. Control rod drop times shall be determined annually, but at intervals not to exceed 15 months.

c. The control rods shall be visually inspected for deterioration at intervals not to exceed 2 years.

Basis

The reactivity worth of the control rods is measured to assure that the required shutdown margin is available and to provide a means for determining the reactivity worths of experiments inserted in the core. Experience with TRIGA reactors over more than 10 years gives assurance that measurement of the reactivity worths on an annual basis is adequate to insure no significant changes in the shutdown margin. The visual inspection of the

control rods and measurements of their drop times are made to determine whether the control rods are capable of performing properly.

4.3 Reactor Instrumentation and Safety Systems

Applicability

This specification applies to the surveillance requirements for the measuring channels of the reactor instrumentation and safety systems.

Objective

The objective is to assure that the instrumentation and safety systems will remain operable and will prevent the fuel temperature safety limit from being exceeded.

Specification

a. A channel test of each of the reactor instrumentation and safety system channels shall be performed prior to each day's operation or prior to the start up of an operation extending more than one day. This test shall not be required prior to immediate restart to power after an unplanned scram is readily determined not to involve an unsafe condition.

b. A channel calibration shall be made of the power level monitoring channels by either nuclear or calorimetric methods annually, but at intervals not to exceed 15 months.

c. A channel check of the power level measuring channels shall be performed prior to each days operation.

d. A channel check of the fuel element temperature measuring channels shall be made prior to each days operation.

Basis

The channel tests and checks will assure that the instrumentation and safety channels are operable on a daily basis or prior to an extended run. A test of the console power supply scram may be made using the normal front panel ON/OFF switch. The calibrations will assure that long-term drift of the channels is corrected.

4.4 Radiation Monitoring Equipment

Applicability

This specification applies to the radiation monitoring equipment required by Section 3.3 of these specifications.

Objective

The objective is to assure that the radiation monitoring equipment is operating and to verify the appropriate alarm settings.

Specification

The alarm set points for the radiation monitoring instrumentation shall be verified prior to each day's operation or prior to the start up of an operatior extending more than one day.

Basis

This surveillance of the equipment will assure that the radiation monitoring channels are operable for each operation.

4.5 Maintenance

Applicability

This specification applies to the surveillance requirements following maintenance on the control or safety system.

Objective

The objective is to assure that a system is operable before being used for reactor operation after maintenance has been performed.

Specification

Following maintenance or modification of a control or safety system or component, it shall be verified that the system is operable prior to its return to service for reactor operation purposes.

Basis

The intent of the specification is to assure that work on the system or component has been properly carried out and that the system or component has been properly reinstalled or reconnected.

4.6 Reactor Pool Water

Applicability

This specification applies to the radioactivity content of the reactor pool water.

Objective

The objective is to assure that the reactor is not operated with leaking fuel elements installed.

Specification

The reactor pool water will be sampled and monitored for radioactivity content on a monthly basis but at intervals not to exceed 60 days. If any activity above background levels is detected, a determination of the isotopes present will be made. If these isotopes are those expected as common fission products, normal operations will be terminated. These isotopes may include radioactive cesium, radioactive strontium and radioactive iodine among others. Operations may then continue under the direction of the Radiation Safety Office to determine which elements are defective.

Basis

Operating experience at the University of Maryland has shown that the pool water is "clean" with the possible exception of some sodium contamination. The presence of fission fragment isotopes would indicate a possible leaking fuel element.

5.0 DESIGN FEATURES

5.1 Reactor Fuel

Applicability

This specification applies to the fuel elements used in the reactor core.

Objective

The objective is to assure that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their mechanical integrity.

Specification

The fuel element shall contain uranium-zirconium hydride, clad in 0.020 inch of type 304 stainless steel. It shall contain a maximum enrichment of 20 percent. There shall be 1.55 to 1.80 hydrogen atoms to 1.0 zirconium atom.

Basis

This type of fuel element has a long history of successful use in TRIGA reactors.

5.2 Reactor Building

Applicability

This specification applies to the building which houses the reactor.

Objective

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

Specification

a. The reactor shall be housed in a closed room designed to restrict leakage.

b. The minimum free volume of the reactor room shall be 55,000 cubic feet.

c. The reactor room shall be equipped with a ventilation system capable of exhausting air or other gases from the reactor room from roof vents at 24 feet above ground level.

Basis

This specification limits the release of radioactivity in an uncontrolled manner. Twenty years of operating experience shows that this building is adequate to house the MUTR.

5.3 Fuel Storage

Applicability

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

Objective

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

Specification

a. All fuel elements shall be stored in a geometrical array where the k_{eff} is less than 0.8 for all

conditions of moderation.

b. 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.

Basis

The limits of Specifications 5.3.a and 5.3.b are conservation and assure safe storage.

6.0 REACTOR POOL WATER SYSTEM

Applicability

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

Objective

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

Specifications

a. The reactor shall be cooled by natural convective water flow.

b. The pool water inlet shall have a siphon break installed. The pool water outlet will be 24 inches below normal operating height when fuel is in the core.

c. The diffuser inlet shall have a siphon break installed. The diffuser pump inlet will be 24 inches below normal operating height.

d. A pool level alarm shall indicate loss of coolant if the pool level drops approximately 2 feet below normal level.

Basis

a. This specification is based on thermal and hydraulic calculations which show that the TRIGA fuel can operate in a safe manner at power levels up to one megawatt with natural convection flow of the coolant water.

b. This prevents accidental siphoning of the water to a limit of no more than 2 feet below normal.

c. This prevents accidental siphoning of the water to a limit of no more than 2 feet below normal.

d. Loss of coolant alarm after 2 feet of loss requires corrective action. This alarm is observed in the reactor control room and outside the reactor building.

7.0 ADMINISTRATIVE CONTROLS

7.1 Organization

a. The reactor facility shall be an integral part of the Nuclear Engineering Program of the Department of Chemical and Nuclear Engineering of the University of Maryland. The reactor shall be related to the University structure as shown in Chart 1.

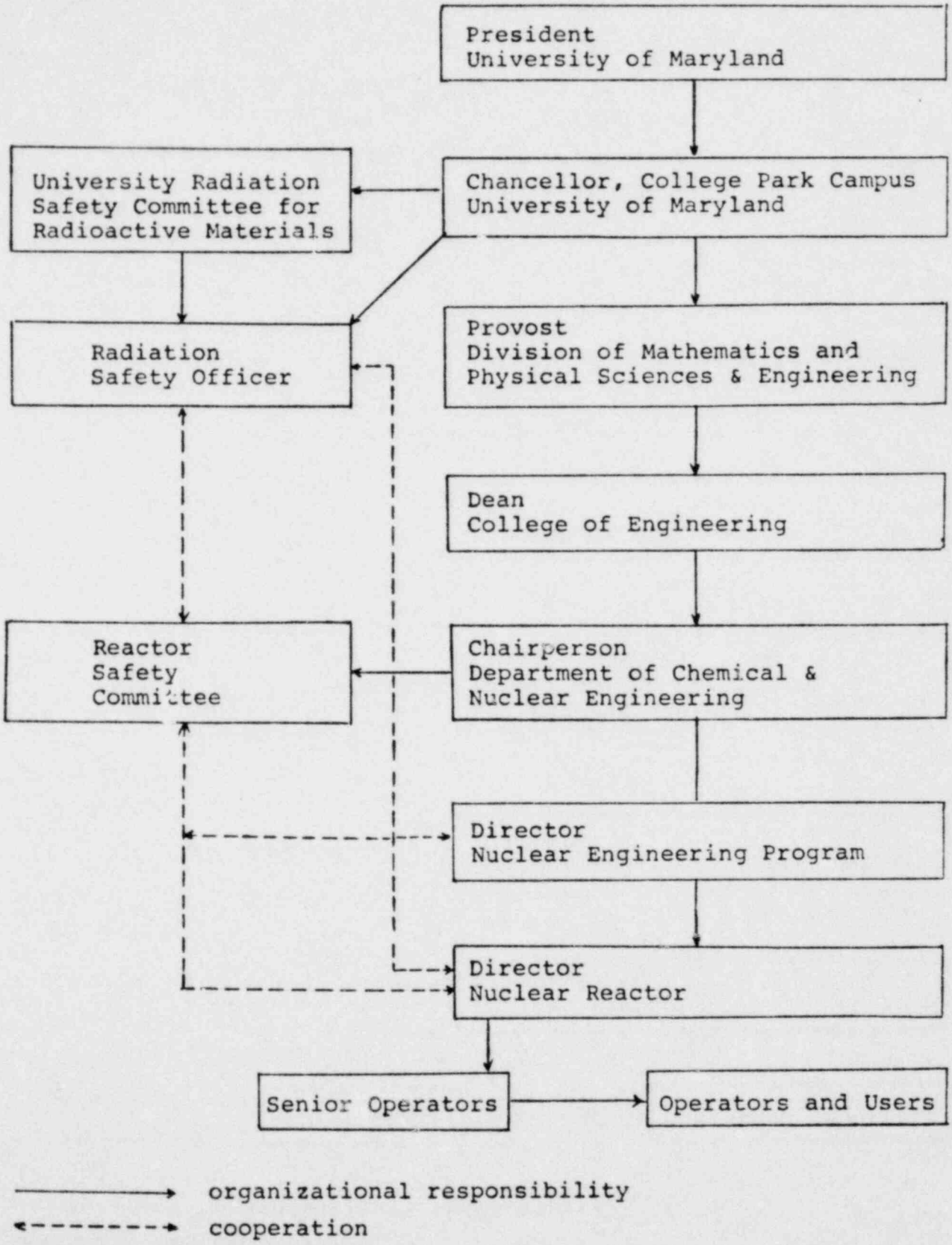

Chart 1 Administration Chart

b. The reactor facility shall be under the supervision of the Reactor Director who shall have been qualified as a licensed senior operator for the reactor. He shall be responsible to the chairman of the department for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and the provisions of the Reactor Safety Committee.

c. There shall be a Radiation Safety Officer responsible for assuring the safety of reactor operations from the standpoint of radiation protection. He shall report to the University Radiation Safety Committee which is an organization independent of the Reactor Safety organization as shown in Chart 1.

7.2 Reactor Safety Committee

The Reactor Safety Committee, including its chairman, is appointed by the Head of the Chemical and Nuclear Engineering Department. The Committee reviews experiments, procedures and safety programs that involve the Reactor Facility and its contents and makes appropriate recommendations to the Head of the Department of Chemical and Nuclear Engineering.

a. The Committee membership and minimum qualifications are:

1. The Head of the Chemical and Nuclear Engineering Department (ex officio member).

2. The Reactor Director (ex officio member).
3. At least one member of the nuclear engineering faculty.
4. At least one member of the faculty outside the Nuclear Engineering Program.
5. The Radiation Safety Officer (RSO) of the University of Maryland.

Faculty members cited in items 3 and 4 must hold PhD degrees and together have experience and competence in radiation safety, chemistry, physics and engineering.

The RSO is appointed by the Chancellor's office with University-wide responsibility for the administration of the Byproduct Materials License on the College Park and Baltimore Campuses. The RSO provides technical assistance in radiological safety matters associated with the reactor. He also has an indirect role as a member of the Reactor Safety Committee, and he cooperates with the Reactor Director in such matters as waste disposal, film badge service, and the movement of radioactive materials in and out of the Reactor Building.

The Reactor Director is responsible for administration of radiation safety procedures in the Reactor Building. This is accomplished directly and by delegation of

such duties to operators and reactor staff members. Any operator or reactor staff member charged with radiological safety responsibilities has authoritiy to shutdown any activity in the Reactor Building that is, in his view, a source of immediate hazard.

b. The minimum meeting frequency:

The Reactor Safety Committee will meet as frequently as required in the circumstances, but not less than quarterly. Attendance of other persons at a given meeting may be requested by the Chairman of the Committee.

c. The Committee shall:

1. Be responsible for review of all proposed normal and abnormal maintenance and emergency operating procedures and proposed changes thereto;
2. Review proposals for new and unique experiments;
3. Review reported violations of the Technical Specifications and make appropriate recommendations;
4. Review proposed changes or modifications to plant systems and equipment;
5. Review unusual occurences and incidents which are reported under the provisions of 10CFR 20 and 10 CFR 50;

6. Recommend approval or disapproval of (a) contemplated changes to procedures or equipment which may involve "unreviewed safety questions" or changes to the Technical Specifications pursuant to 10 CFR 50.59, and (b) implementation of changes or modifications to plant systems or equipment; and
7. Periodically audit the radiation safety program at the reactor.

If any member of the Committee feels that a hazards condition prevails or that safety records are improperly maintained, the reactor operation will cease at his request until the Committee and the Reactor Director agree that the necessary steps are taken to bring about safe operations.

The recommendations, reviews, and deliberations of the Committee will be made to the Head of the Department of Chemical and Nuclear Engineering to whom it is responsible. These actions will be recorded.

7.3 Operating Procedures

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

a. Start-up, operation, and shutdown of the reactor.

b. Installation or removal of fuel elements, control rod, experiments, and experimental facilities.

c. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary coolant system leaks, and abnormal reactivity changes.

d. Emergency conditions involving potential or actual release of radioactivity, including provisions for evacuation, re-entry, recovery and medical support.

e. Maintenance procedures which could have an effect on reactor safety.

f. Periodic surveillance of reactor instrumentation and safety systems, area monitors and continuous air monitors.

g. Civil disturbances on or near the campus.

Substantive changes to the above operating procedures shall be made with the approval of the Reactor Safety Committee. Temporary changes to the procedures that do not change their original intent may be made with the approval of the Reactor Director. All such temporary changes to procedures shall be documented

and subsequently reviewed by the Reactor Safety Committee.

7.4 Action to be Taken in the Event a Safety Limit is Exceeded

In the event the safety limit (fuel temperature) is exceeded:

a. The reactor shall be shutdown and reactor operation shall not be resumed until authorized by the NRC.

b. An immediate report of the occurrence shall be made to the Chairman of the Reactor Safety Committee, and reports shall be made to the NRC in accordance with Section 7.5 of these specifications.

c. A report shall be made which shall include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability or reoccurrence. This report shall be submitted to the Reactor Safety Committee for review, and a suitable similar report submitted to the NRC when authorization to resume operation of the reactor is sought.

7.5 Action to be Taken in the Event of an Reportable Occurrence

In the event of a reportable occurrence, as defined in Section 1.11 of the Specifications, the following

action shall be taken:

a. The Reactor Director shall be notified and corrective action taken prior to resumption of the operation involved.

b. A report shall be made which shall include an analysis of the cause of the occurrence, efficacy of corrective action and recommendations for measures to prevent or reduce the probability of reoccurrence. This report shall be submitted to the Reactor Safety Committee for review.

c. Where appropriate, a report shall be submitted to the NRC in accordance with Section 7.7 of these Specifications.

7.6 Plant Operating Records

In addition to the requirements of applicable regulations, and in no way substituting therefore, records and logs shall be prepared and retained for a period of at least 5 years for items a through e and indefinitely for items f through k;

a. Normal plant operation;

b. Principal maintenance activities;

c. Reportable occurrences;

d. Equipment and component surveillance activities;

e. Experiments performed with the reactor;

f. Gaseous and liquid radioactive effluents released to the environs;

g. Off-site environmental monitoring surveys;

h. Fuel inventories and transfers:

i. Facility radiation and contamination surveys;

j. Radiation exposures for all personnel.

7.7 Reporting Requirements

In addition to the requirements of applicable regulations, and in no way substituting therefore, reports shall be made to the NRC as follows:

a. A report within 24 hours by telephone and telegraph to the NRC Region I Regulatory Operations with a copy to the Director of Licensing.

   1. Any accidental release of radioactivity above permissible limits in unrestricted areas, whether or not the release resulted in property damage, personal injury, or exposure,

   2. Any violation of the safety limit; and

   3. Any reportable occurrences as defined in Section 1.11 of these specifications.

b. A report within 10 days in writing to the Director of Licensing, NRC, Washington, D.C., 20545, with a copy to the NRC Region I Regulatory Operations of:

   1. Any accidental release of radioactivity above permissible limits in unrestricted areas, whether

or not the release resulted in property damage, personal injury, or exposure; the written report (and, to the extent possible, the preliminary telephone or telegraph report) shall describe, analyze and evaluate safety implications, and outline the corrective measures taken or planned to prevent reoccurrence of the event.

2. Any violation of a Safety Limit; and

3. Any reportable occurrence as defined in Section 1.11 of these Specifications.

c. A report within 30 days in writing to the Director of Licensing, USNRC, Washington, D.C., 20545, with a copy to the NRC Region I Regulatory Operations of:

1. Any significant variation of measured values from a corresponding predicted or previously measured value of safety-connected operating characteristics occuring during operation of the reactor;

2. Any significant change in accident analyses as described in the Safety Analysis Report; and

3. Any changes in facility organization.

7.7.1 An annual report within 60 days following the 30th of June of each year (in writing to the Director of Licensing, USNRC Washington, D.C. 20545, with a copy to the NRC Region I Regulatory Operations) providing the following information.

a. A brief narrative summary of (1) operating experience (including experiments performed), (2) changes in facility design, performance characteristics and operating procedures related to reactor safety and occurring during the reporting period, and (3) results of surveillance tests and inspections;

b. Tabulation of the energy output (in megawatt days) or the reactor, hours reactor was critical, and the cummulative total energy output since initial criticality;

c. The number of emergency shutdown and inadvertent scrams, including reasons therefore;

d. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor, and the reasons for any corrective maintenance required;

e. A brief description including a summary of the safety evaluations of changes in the facility

or in procedures and of tests and experiments carried out pursuant to Section 50.59 of 10 CFR Part 50;

f. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge.

Liquid Waste (summarized on a monthly basis)

1. Radioactivity discharged during the reporting period.
   a. Total radioactivity released (in curies).
   b. The MPC used and the isotopic composition if greater than 1×10^{-7} microcuries/cc for fission and activation products.
   c. Total radioactivity (in curies) released by nuclide during the reporting period based on representative isotopic analysis.
   d. Average concentration at point of release (in microcuries/cc) during the reporting period.
2. Total volume (in gallons) of effluent water (including diluent) during periods of release.

Gaseous Waste (summarized on a monthly basis)

1. Radioactivity discharged during the reporting period (in curies) for;
   a. Gases
   b. Particulates with half lives greater than eight days.
2. The MPC used and the estimated activity (in curies) discharged during the reporting period, by nuclide, for all gases and particulates based on representative isotopic analysis.

Solid Waste

1. The total amount of solid waste packaged (in cubic feet).
2. The total activity involved (in curies).
3. The dates of shipment and disposition (if shipped off-site).

g. A summary of radiation exposures received by facility personnel and visitors, including dates and time of significant exposures, and a summary of the results of radiation and contamination surveys performed within the facility; and;

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

7.8 Experiment Approval Procedures

a. Routine experiments may be performed at the discretion

of the Reactor Director or his designated alternate, without the necessity of any further review or approval.

b. Prior to performing any reactor experiment which is not a routine experiment, the proposed experiment shall be evaluated by the Reactor Director or a qualified person appointed by him to be responsible for the experiment. He shall consider the experiment in terms of its effect on reactor operation and the possibility and consequences of its failure, including, where significant, consideration of chemical reactions, physical integrity, design life, proper cooling, interaction with core components, and reactivity effects.

c. Modified routine experiments may be performed at the discretion of the Reactor Director without the necessity of any further review or approval provided that the evaluation results in a determination that the hazards associated with the modified routine experiment are neither greater nor significantly different than those involved with the corresponding routine experiment which shall be referenced.

d. No special experiment shall be performed until the proposed experiment or type of experiment has been reviewed and approved by the Reactor Safety Committee.

e. Favorable evaluation of the protective measures provided for an experiment shall conclude that failure of the experiment will lead to failure of a fuel element or of other experiments or interfere with movement of a control rod.