UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

UNIVERSITY OF MARYLAND

DOCKET NO. 50-166

RENEWAL OF THE FACILITY OPERATING LICENSE

Amendment No. 7
License No. R-70

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

   A. The application for facility operating license renewal by the University of Maryland (the licensee) dated May 23, 1980, as supplemented, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR, Chapter I;

   B. Construction of the facility was completed in substantial conformity with Construction Permit No. CPRR-53, dated June 29, 1960, and modified in conformity with CPRR-108 dated March 25, 1970, the provisions of the Act, and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the amended license, the provisions of the Act, and the rules and regulations of the Commission;

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

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

   F. The licensee is a nonprofit educational institution and will use the facility for the conduct of educational activities, and has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   G. The issuance of this amended license will not be inimical to the common defense and security or to the health and safety of the public;

   H. The issuance of this amended license is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

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I. The receipt, possession and use of the byproduct and special nuclear materials as authorized by this amended license will be in accordance with the Commission's regulations in 10 CFR Parts 30 and 70, including Sections 30.33, 70.23 and 70.31.

2. Facility License No. R-70, as previously amended, is hereby amended in its entirety to read as follows:

A. This amended license applies to the Maryland University Training Reactor (herein MUTR or the reactor), owned by the University of Maryland (the licensee), located at College Park, Maryland, and described in the application dated May 23, 1980, as supplemented.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the University of Maryland:

(1) Pursuant to Section 104c of the Act and Title 10 CFR Chapter I, Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the reactor in accordance with the procedures and limitations described in the application and in this license;

(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess, and use up to 3,441 grams of uranium-235 contained in enriched uranium, and 80 grams of plutonium contained in encapsulated plutonium-beryllium sources for use in connection with operation of the reactor; and

(3) Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," to possess, but not to separate, such byproduct materials as may be produced by operation of the facility.

C. This amended license shall be deemed to contain and be subject to the conditions specified in Parts 20, 30, 50, 51, 55, 70 and 73 of 10 CFR, Chapter I, to all applicable provisions of the Act, and to the rules, regulations and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified below:

(1) <u>Maximum Power Level</u>

The licensee may operate the reactor at steady state power levels up to and including 250 kilowatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment 7, are hereby incorporated into the amended license. The licensee shall operate the facility in accordance with these Technical Specifications.

(3) Physical Security Plan

The licensee shall maintain and fully implement all provisions of the Commission-approved physical security plan entitled "Physical Security Plan for Maryland University Reactor Facility," dated June 1980 submitted by letter dated May 23, 1980, as revised July 21, 1980, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved security plan consists of documents withheld from public disclosure pursuant to 10 CFR 2.790(d).

3. This amended license is effective as of the date of issuance and shall expire at midnight on June 29, 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
Appendix A Technical
Specifications

Date of Issuance: August 7, 1984

TECHNICAL SPECIFICATIONS FOR THE MARYLAND UNIVERSITY TRAINING REACTOR

License No. R-70

Docket No. 50-166

22 June 1984

TECHNICAL SPECIFICATION
MARYLAND UNIVERSITY TRAINING REACTOR

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TECHNICAL SPECIFICATIONS
for the
University of Maryland Reactor

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 - 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.

1.2.1 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.

1.2.2 Channel Check - A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

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

1.3 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.)

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

1.3.2 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.

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

1.4 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.

1.5 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.

1.6 Fuel Element - A fuel element is a single TRIGA fuel rod.

1.7 Full Power - Full licensed power is defined as 250 kilowatts.

1.8 Instrumented Element - An instrumented element is a special fuel element in which a sheathed chromel-alumel or equivalent thermocouple is embedded in the fuel.

1.9 Limiting Conditions for Operation - These are administratively established constraints on equipment and operational characteristics which shall be adhered to during operation of the facility.

1.10 Limiting Safety System Setting - The limiting safety system (LSSS) setting is the setting for automatic protective devices related to those variables having significant safety functions.

1.11 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.

1.12 Measured Value - The measured value is the value of a parameter as it appears on the output of a channel.

1.13 On Call - A senior operator will be available "on call" 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.

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

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

1.16 Reactivity Worth of an Experiment - The reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter experiment position or configuration.

1.17 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.

1.18 Reactor Operating - The reactor is operating whenever it is not secured or shutdown.

1.19 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.

1.20 Reactor Secured - The reactor is secured when:

a. It contains insufficient fissile material or moderator present in the reactor and adjacent experiments to attain criticality under optimum available conditions of moderation and reflection, or

b. The reactor console is secured, and

1. 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

2. No experiments in or near the reactor are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value of one dollar.

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

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

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

1.21.3 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 sytems 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 perform their intended reactor safety function.)

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

1.21.5 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.

1.21.6 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.

1.22 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, with the fuel and moderator at ambient temperature.

1.23 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.

1.24 Safety Channel - A safety channel is a measuring channel in the reactor safety system.

1.25 Safety Limit - Safety limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of certain physical barriers which guard against the uncontrolled release of radioactivity.

1.26 Scram Time - Scram time is the time measured from the instant a simulated signal reaches the value of the LSSS to the instant that the slowest scrammable control rod reaches its fully inserted position.

1.27 Shutdown Margin - Shutdown margin shall mean 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 operating condition, if the most reactive rod is stuck in its most reactive position, and that the reactor will remain subcritical without further operator action.

1.28 Standard Core - A standard core is an arrangement of standard TRIGA fuel in the reactor grid plate.

1.29 Steady State Mode - Steady state mode operation shall mean operation of the reactor with the mode selector switch in the steady state position.

1.30 Uscheduled 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.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT

APPLICABILITY

This Specification applies to the temperature of the reactor fuel.

OBJECTIVE

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

SPECIFICATIONS

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 provided that 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. It has been shown by analysis and measurements on other TRIGA reactors that a power level of 1000 kW corresponds to a peak fuel temperature of approximately 400°C.

2.2 LIMITING SAFETY SYSTEM SETTINGS

APPLICABILITY

This specification applies to the reactor scram setting which 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 400°C as measured by the instrumented fuel element. The instrumented element may be located at any position in the core.

BASIS

A Limiting Safety Setting 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 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 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.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTIVITY LIMITS

APPLICABILITY

These specifications shall apply to the reactor at all times that it is in operation.

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 shutdown margin shall not be less than \$0.50.

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

BASES

(1) The value of the shutdown margin as required by specification 3.1 (1) 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.

(2) While specification 3.1 (2), in conjunction with specification 3.1 (1) tends to overconstrain the excess reactivity, it helps insure that the operable core is similar to the core analyzed in the FSAR.

3.2 REACTOR CONTROL AND SAFETY SYSTEM

APPLICABILITY

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 standard control rod from the fully withdrawn position to the fully inserted position shall not exceed one second.

(2) The reactor safety channels shall be operable in accordance with Table 3-1a, including the minimum number of channels and the indicated maximum or minimum set points, for the scram channels.

(3) Maximum positive reactivity insertion rate by control rod motion shall not exceed $0.30 per second.

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) provides protection against the reactor operating outside of the safety limits. Table 3-1b describes the basis for each of the reactor safety channels.

(3) Specification 3.2 (3) 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)

Table 3-1a
REACTOR SAFETY CHANNELS
Scram Channels

Scram Channel	Minimum Operable	Scram Setpoint
Reactor Power Level	2	Not to exceed 120% of full licensed power
Fuel Element Temperature	1	Not to exceed 400°C
Reactor Power Channel DetectorPower 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

INTERLOCKS

Interlock/Channel	Function
Log Power Level	Provide signal to period rate and minimum source channels
Startup Countrate	Prevent control rod withdrawal when neutron count rate is less than 1 cps.
Rod Drive Control	Prevent simultaneous manual withdrawal of two or more control rods in the steady state mode of operation.

Table 3-1b
REACTOR SAFETY CHANNELS
Scram Channels

Scram Channel	BASES
Reactor Power Level Fuel Element Temp.	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 which 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 can not be operated without a secure electric supply.

Interlocks

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 Countrate	Assures that sufficient amount of startup neutrons are available to achieve a controlled approach to criticality.
Rod Drive Control	Limits the maximum positive reactivity insertion rate available for steady state operation.

3.3 PRIMARY COOLANT CONDITION

APPLICABILITY: This specification applies to the quality 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

SPECIFICATION

Conductivity of the pool water shall be no higher than 5 x 10^{-6} mhos/cm.

BASES:

(1) 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.

(2) 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.

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 operator to assure safe operation of the reactor.

SPECIFICATIONS

(1) The reactor shall not be operated unless the radiation area monitor channels listed in Tables 3-2 are OPERABLE

(2) For a period of time not to exceed 48hours for maintenance or calibration to the radiation monitor channels, the intent of specification 3.4 (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.

Table 3-2
MINIMUM RADIATION MONITORING CHANNELS

Radiation Area Monitors	Function	Minimum Number Operable
Exhaust Radiation Monitor	Monitor Radiation levels in Reactor Bay area	1 out of 2
Bay Radiation Monitor	Monitor Radiation levels in Reactor Bay area	1 out of 2

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.

3.5 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. Each non-secured experiment shall have a reactivity worth less than one dollar.

2. The reactivity worth of any single experiment shall be less than 1.00 dollar.

3. The total reactivity worth of in-core experiments shall not exceed 3.00 dollars, including the potential reactivity which might result from experimental malfunction and experiment flooding or voiding.

4. Experiments containing materials corrosive to reactor components, compounds highly reactive with water, potentially explosive materials, and liquid fissionable materials shall be doubly encapsulated.

5. Explosive materials, such as gunpowder, TNT, PETN, or nitroglycerin, in quantities greater than 25 milligrams shall not be irradiated in the reactor or 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.

6. 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.

In calculations pursuant to 6. above, the following assumptions shall be used:

(a) If the effluent from an experimental facility exhausts through a holdup tank which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape.

(b) 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 these particles can escape.

7. 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.

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 inadvertant 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 inadvertantly.

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 II of Appendix B of 10 CFR Part 20 will be released to the atmosphere outside the facility boundary.

7. 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. (see FSAR)

4.0 SURVEILLANCE REQUIREMENTS

4.1 REACTIVITY LIMITS

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 reactivity worth of each standard control rod and the reactor shutdown margin shall be determined biennially, at intervals not to exceed 28 months, and after each time the core fuel configuration is changed.

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 worth of experiments inserted in the core. Long term effects of TRIGA reactor operation are such that measurements of the reactivity worths on a biennual basis is adequate to insure no significant changes in shutdown margin have occurred.

4.2 REACTOR CONTROL AND SAFETY SYSTEM

APPLICABILITY

These specifications apply to the surveillance requirements of the reactor safety system.

OBJECTIVE

The objective of these specifications is to ensure the operability of the reactor safety system as described in Section 3.2.

SPECIFICATIONS

(1) Control rod drop times shall be measured annually, at intervals not to exceed 15 months, or whenever maintenance or repairs are made that could affect their drop time.

(2) A channel calibration shall be made of the linear power level monitoring channels annually, at intervals not to exceed 15 months.

(3) A channel test shall be performed for each scram channel listed on TABLE 3-1a and the log power channel prior to the first operation of each day or prior to the startup of an operation extending more than one day.

(4) All reactor safety scram channels listed on table 3-1a and the log power channel after maintenance or repair shall be verified operable prior to being returned to service.

(5) The control rods shall be visually inspected for deterioration biennially, at intervals not to exceed 28 months.

BASES

(1) Measurement of the control rod drop time, specification 4.2 (1), ensures that the rods can perform their safety function properly.

(2) The linear power level channel calibration will assure that the reactor will be operated at the licensed power levels.

(3) The surveillance requirement specified in specification 4.2 (2) for the reactor safety scram channels ensures that the overall functional capability is maintained.

(4) The surveillance test performed after maintenance or repairs to the reactor safety system as required by specification 4.2 (3) ensures that the affected channel will perform as intended.

(5) Specification 4.2 (5) assures that a visual inspection of control rods is made to evaluate corrosion and wear characteristics and any damage caused by operation in the reactor.

4.3 REACTOR COOLANT SYSTEM

APPLICABILITY

These specifications apply to the surveillance requirements for the pool water.

OBJECTIVE

The objective of these specifications is to ensure that the specifications of Section 3.3 are satisfied for long intervals without operation.

SPECIFICATIONS

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

BASES

Specification 4.3 (1) 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.

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

4.4 RADIATION MONITORING SYSTEM

APPLICABILITY

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

OBJECTIVE

The objective of these specifications is to ensure the operability of each radiation area monitoring channel as required by Section 3.4.

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 4.4 (1) and (2) ensure that the various radiation area monitors are checked and calibrated on a routine basis, in order to assure compliance with 10CFR20.

5.0 DESIGN FEATURES

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 licence.

SPECIFICATIONS

Standard TRIGA fuel

The individual unirradiated standard TRIGA fuel elements shall have the following characteristics:

(1) Uranium content: a maximum of 9.0 Wt - % uranium enriched to less than 20% Uranium -235.

(2) Zirconium hydride atom ratio: nominal 1.5 - 1.8 hydrogen-to-zirconium, ZrH_x.

(3) Cladding: 304 stainless steel, nominal .020 inches thick.

BASIS

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

5.1 REACTOR COOLANT SYSTEM

APPLICABILITY

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

OBJECTIVE

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

SPECIFICATIONS

1. The reactor core shall be cooled by natural convective water flow.

2. 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 20 inches below the overflow outlet pipe when fuel is in the core.

BASES

Specification 5.1 is based on thermal and hydraulic calculations and operation of other TRIGA reactors which 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 5.1 (2 & 3) ensures that the pool water level can normally decrease only by 20 inches if the coolant piping where to rupture and siphon water from the reactor tank. Thus, the core will be covered by at least 15 feet of water.

5.2 VENTILATION SYSTEM

APPLICABILITY

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

OBJECTIVE

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

SPECIFICATION

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. The minimum free air volume of the reactor room shall be $1.7 \times 10^9 cm^3$.

3. 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.

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 (10CFR.20) is exceeded.

5.3 FUEL STORAGE

APPLICABILITY

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

OBJECTIVE

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

SPECIFICATIONS

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

2. Irradiated fuel elements and fueled devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the fuel element or fueled device temperature will not exceed design values.

BASIS

The limits imposed by Specifications 5.3(1) and 5.3(2) are conservative and assure safe storage.

6.0 ADMINISTRATION

6.1 ORGANIZATION

The Maryland University Training Reactor (MUTR) is owned and operated by the University of Maryland. Its position in the university 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 Organization 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.

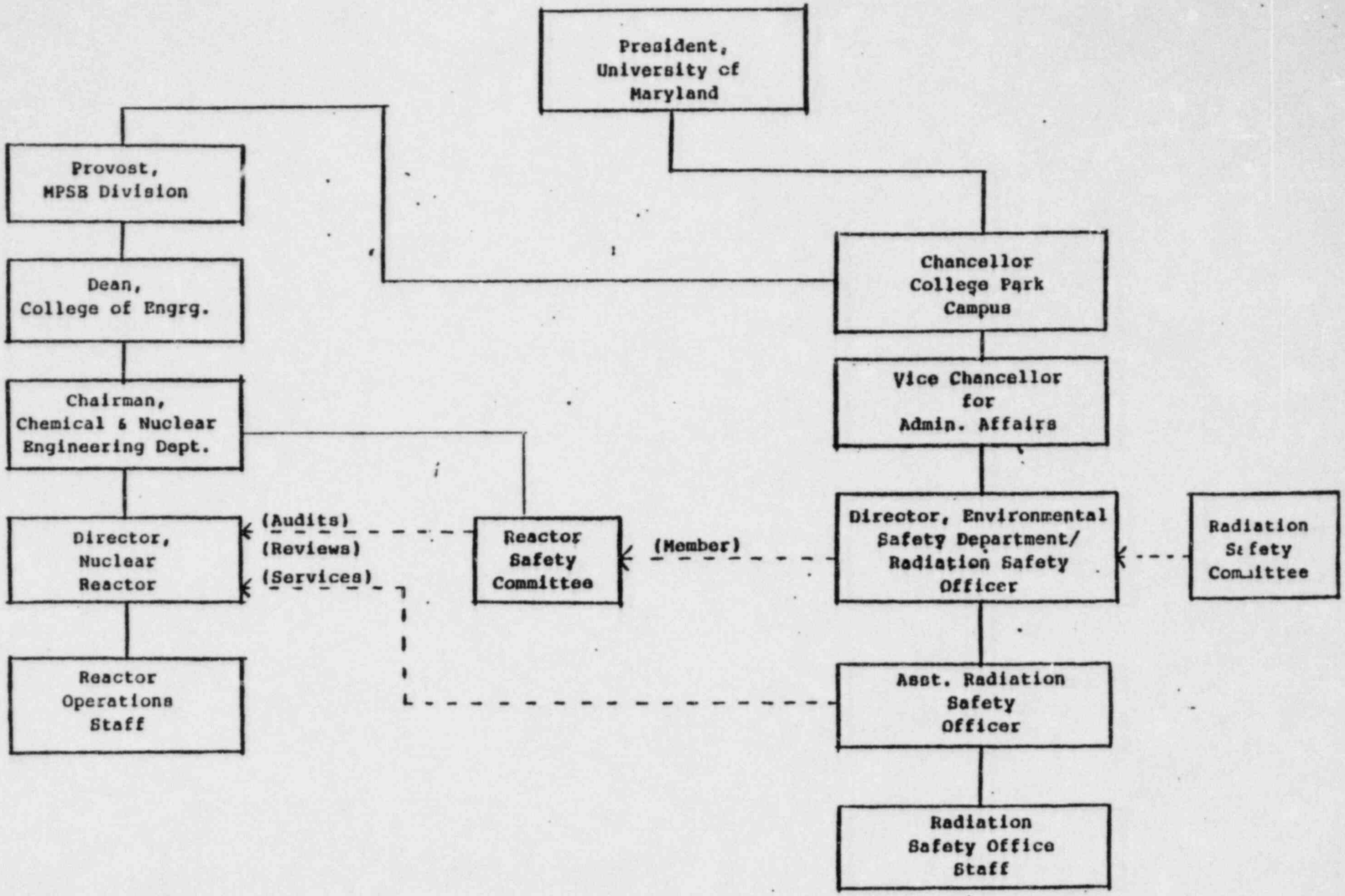
6.1.3 FACILITY STAFF REQUIREMENTS

(1) A licensed reactor operator (RO) or a licensed senior reactor operator (SRO) shall be present in the control room whenever the reactor is operating.

(2) 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.

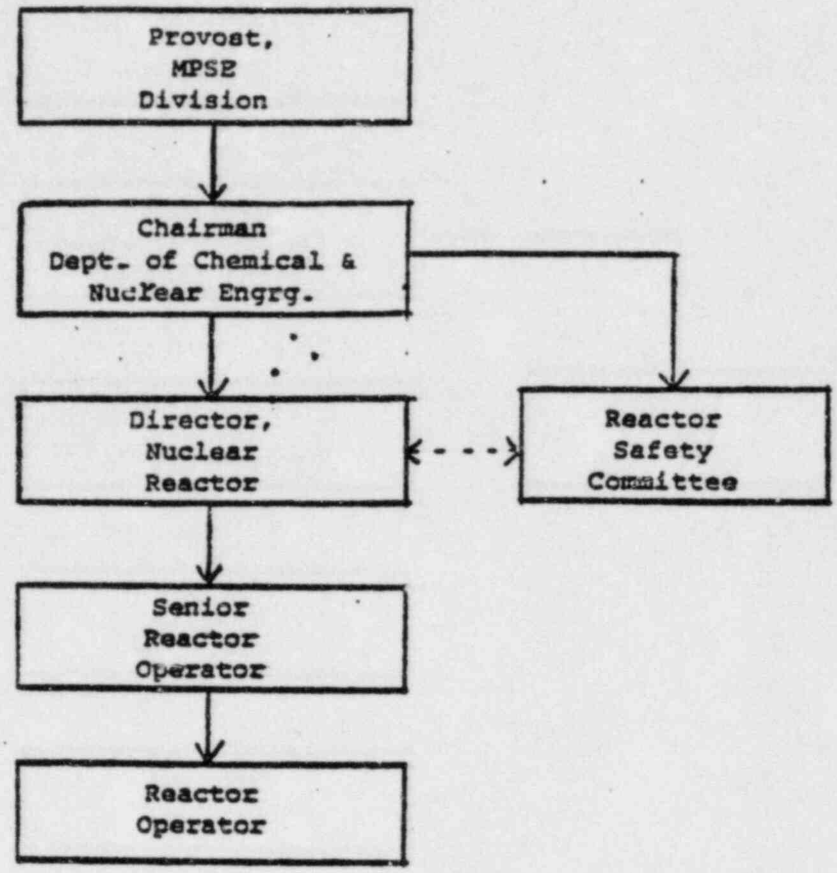
(3) The following operations must be supervised by a senior reactor operator:

- (a) fuel manipulations in the core
- (b) when experiments are being manipulated in the core that have an estimated worth greater than $0.80.



ADMINISTRATIVE ORGANIZATION

Fig. 6-1



MUTR ORGANIZATIONAL STRUCTURE

Fig. 6-2

(c) removal of control rods.

(d) resumption of operation following an unscheduled shutdown. (This requirement is waived if the shutdown is initiated by an interruption of electrical power to the plant.)

(4) A licensed SRO must be present or readily available on call at any time the reactor is in operation.

6.1.4 SELECTION AND TRAINING OF PERSONNEL - The selection and training of operations personnel shall be in accordance with the following:

(a) Responsibility - The Facility Director or his designated alternate is responsible for the training and requalification of the facility reactor operators and senior reactor operators.

(b) Requalification Program

(1) Purpose - To insure that all operating personnel maintain proficiency at a level equal to or greater than that required for initial licensing.

(2) Scope - Lectures, written examinations, and evaluated console manipulations will be used to insure operator proficiency is maintained.

6.2 REVIEW & 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. It is appointed by and reports to the chairman of the Chemical and Nuclear Engineering Department. Qualified alternates may serve on the committee. Alternates may be appointed by the Chairman of the RSC to serve on a temporary basis.

6.2.2 REACTOR SAFETY COMMITTEE CHARTER AND RULES

(1) 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 his designated alternate.

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

(3) A quorum of the RSC must have at least four members present. No more than two alternates may be used to make a quorum.

(4) 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:

(a) experiments referred to it by the Facility Director because of the degree of hazzard involved or the unusual nature of the experiment.

(b) report of occurrences (see Section 6.5)

(c) proposed changes to the facility license, changes to technical specifications, and experiments or changes made pursuant to 10CFR50.59.

(d) operating procedures.

6.2.4 REACTOR SAFETY COMMITTEE AUDIT FUNCTION

(1) A biennial 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

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.

6.3 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 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 rods, experiments, and experimental facilities.

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

d. Periodic surveillance of reactor instrumentation and safety systems and area monitors as required by these Technical Specifications.

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.

6.4 EXPERIMENT REVIEW AND APPROVAL

(1) Routine experiments may be performed at the discretion of the duty senior reactor operator without the necessity of any 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 designated alternate prior to initiation.

(4) The review of an experiment listed in subsections (2) and (3) above, shall consider 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.

6.5 REQUIRED ACTIONS

6.5.1 ACTIONS TO BE TAKEN IN THE EVENT OF A REPORTABLE OCCURRENCE

In the event of a reportable occurrence, as defined in 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 conditions shall be returned to normal or the reactor shall be shut down.

(3) The Reactor Safety Committee will investigate the causes of the occurrence. The Reactor Safety Committee will report its findings to the NRC and Provost, Division of Mathematics Physical Science and 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.

6.5.2 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

In the event a safety limit is exceeded:

(a) The reactor shall be shut down 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, Reactor Safety Committee, and reports shall be made to the NRC in accordance with Section 6.6.2 of these specifications, and

(c) 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.

6.6 REPORTS

6.6.1 ANNUAL OPERATING REPORT

A report summarizing facility operations will be prepared annually for the reporting period ending June 30th. A copy of this report shall be submitted to the NRC Region I Office of Inspection and Enforcement by September 30 of each year, with a copy to the Director, Division of Licensing, Office of Nuclear Reactor Regulation, NRC. 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 megawatt-hours for the year.

(3) A list of unplanned 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 and (b) review of changes, tests, and experiments made pursuant to 10CFR50.59.

(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 10CFR20.

(9) Changes in facility organization.

6.6.2 REPORTABLE OCCURRENCE REPORTS

Notification shall be made within 24 hours by telephone or telegraph to the Director of the Regional Inspection and Enforcement Office followed by a written report within 14 days in the event of a reportable occurrence, as defined in Section 1.0. The written report and, to the extent possible, the preliminary telephone or telegraph notification shall:

(1) describe, analyze, and evaluate safety implications

(2) outline the measures taken to ensure that the cause of the condition is determined

(3) indicate the corrective action taken to prevent repetition of the occurrence including changes 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.

6.6.3 UNUSUAL EVENT REPORT

A written report shall be forwarded within 30 days to the Director of the Regional Inspection and Enforcement Office 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 basis 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.

6.7 RECORDS

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

(2) 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) incore experiments

(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

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

(a) liquid radioactive effluents released to the environs

(b) radiation exposure for all facility personnel

(c) As-built facility drawing

(4) Requirement (2) (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 all of the required information may be used as records in subsection (3) above.