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27-Aug-13

Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Submittal of revised Technical Specifications Docket 50-170

Sir:

Enclosed is the Technical Specification for the AFRRI TRIGA Reactor Docket number 50-170. This document reflects the changes agreed to as of the last RAI submittal.

If you need further information, please contact Mr. Steve Miller at 301-295-9245 or millers@afrri.usuhs.mil.

I declare under penalty of perjury that the foregoing and all enclosed information is true and correct to the best of my knowledge. Executed on September 21, 2012.

Stephen Miller
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Radiation Facilities Program Manager (FRM)

Enclosures:
Technical Specifications
27 August 2013

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NRR

# TECHNICAL SPECIFICATIONS FOR THE AFRRI REACTOR FACILITY

**27 August 2013**

LICENSE R-84
DOCKET 50-170

# Preface

*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.*

TECHNICAL SPECIFICATIONS FOR THE
AFRRI REACTOR FACILITY
LICENSE NO. R-84
DOCKET # 50-170

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## 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 environment as low as reasonably achievable.

### 1.2. CHANNEL

A channel is the combination of sensor, interconnecting cables or lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a parameter.

### 1.3. 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 that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip, and shall be deemed to include a channel test.

### 1.4. CHANNEL CHECK

A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same parameter.

### 1.5. CHANNEL TEST

A channel test is the introduction of a signal into the channel for verification that it is operable.

### 1.6. CORE CONFIGURATION

The core configuration includes the number, type, or arrangement of fuel elements and standard control rods/transient rod occupying the core grid.

### 1.7. CORE GRID POSITION

The core grid position refers to the location of a fuel element, control rod, or experiment in the grid structure.

### 1.8. EXCESS REACTIVITY

Excess reactivity is that amount of reactivity that would exist if all control rods were moved to the maximum reactive condition from the point where the reactor is exactly critical ($k_{eff} = 1$) at reference core conditions or at a specific set of conditions.

## 1.9. EXPERIMENT

Experiment shall mean (a) any apparatus, device, or material that is not a normal part of the core or experimental facilities, but that is inserted into these facilities or is in line with a beam of radiation originating from the reactor core; or (b) any operation designed to measure non-routine reactor parameters or characteristics.

## 1.10. EXPERIMENTAL FACILITIES

The experimental or exposure facilities associated with the AFRRI TRIGA reactor shall be:

a. Exposure Room #1

b. Exposure Room #2

c. Reactor Pool

d. Core Experiment Tube

e. Portable Beam Tubes

f. Pneumatic Transfer System

g. In-core Locations

NOTE: Exposure facility protective barriers shall be differentiated from the primary protective barrier (fuel element cladding) for purposes of placement of experiments within these barriers.

## 1.11. FUEL ELEMENT

A fuel element is a single TRIGA fuel rod, or the fuel portion of a fuel follower control rod.

## 1.12. INSTRUMENTED FUEL ELEMENT

An instrumented fuel element is a special fuel element in which one or more thermocouples have been embedded for the purpose of measuring the fuel temperatures.

## 1.13. MEASURED VALUE

The measured value is the value of a parameter as it appears on the output of a measuring channel.

1.14. MOVABLE EXPERIMENT

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

1.15. ON CALL

A person is considered on call if:

a. The individual has been specifically designated and the operator knows of the designation;

b. The individual keeps the operator posted as to his/her whereabouts and telephone number;

c. The individual remains at a location where the individual is reachable, and is capable of getting to the reactor facility within 60 minutes under normal circumstances; and

d. The individual remains in a state of readiness to perform their duties.

1.16. OPERABLE

A system or component shall be considered operable when it is capable of performing its intended function(s).

1.17. OPERATING

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

1.18. PULSE MODE

Operation in the pulse mode shall mean that the reactor is intentionally placed on a prompt critical excursion by making a step insertion of reactivity above critical with the transient rod. The reactor may be pulsed from a critical or subcritical state.

1.19. REACTIVITY WORTH OF AN EXPERIMENT

The reactivity worth of an experiment is the value of the reactivity change that results from the experiment being inserted into or removed from its intended position.

1.20. REACTOR OPERATING

The reactor is operating whenever it is not secured or shut down.

1.21. REACTOR OPERATOR

An individual who is licensed to manipulate the controls of a reactor.

1.22. REACTOR SAFETY SYSTEMS

Reactor safety systems are those systems, including their associated input circuits that are designed to initiate a reactor scram for the primary purpose of protecting the reactor or to provide information for initiation of manual protective action.

1.23. REACTOR SECURED

The reactor is secured when:

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

b. Or the following conditions exist:

   1. All control rods are fully inserted into the core;

   2. The console key switch is in the off position and the key is removed from the lock;

   3. 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

   4. No experiments are being moved or serviced that have, on movement, a reactivity worth exceeding $1.00.

1.24. REACTOR SHUTDOWN

The reactor is shut down when it is subcritical by at least $1.00 of reactivity in the reference core condition with the reactivity worth of all installed experiments included.

1.25. REFERENCE CORE CONDITION

The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<$0.30).

1.26. SAFETY CHANNEL

A safety channel is a measuring channel in the reactor safety system.

1.27. SCRAM TIME

Scram time is the elapsed time between the initiation of a scram signal and the full insertion of the control rod.

1.28. SECURED EXPERIMENT

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

1.29. SENIOR REACTOR OPERATOR

An individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

1.30. SHALL, SHOULD, AND MAY

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

1.31. SHUTDOWN MARGIN

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

1.32. STANDARD CONTROL ROD

A standard control rod is a control rod having electromechanical drive and scram capabilities. It is withdrawn by an electromagnet/armature system.

1.33. STEADY STATE MODE

Operation in the steady state mode shall mean operation of the reactor either by manual operation of the control rods or by automatic operation of one or more control rods (servo control) at power levels not exceeding 1.1 MW, utilizing the appropriate scrams in Table 2 and the appropriate interlocks in Table 3.

1.34. SURVEILLANCE INTERVALS

Allowable surveillance intervals shall not exceed the following:

a. Biennial – interval not to exceed 30 months

b. Annual – interval not to exceed 15 months

c. Semi-annual – interval not to exceed 7.5 months

d. Quarterly – interval not to exceed 4 months

e. Monthly – interval not to exceed 6 weeks

f. Weekly – interval not to exceed 10 days

1.35. TRANSIENT ROD

The transient rod is a control rod with scram capabilities that can be rapidly ejected from the reactor core to produce a pulse. It is activated by applying compressed air to a piston.

1.36. TRUE VALUE

The true value is the actual value of a parameter.

1.37. UNSCHEDULED SHUTDOWN

An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation, not including shutdowns that occur during testing or checkout preparations.

## 2.0. SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTING

### 2.1. SAFETY LIMIT: FUEL ELEMENT TEMPERATURE

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 shall result.

Specification

The maximum temperature in a TRIGA fuel element shall not exceed 1,000°C under any mode of operation.

Basis

The important parameter for a TRIGA reactor is the 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 buildup of excessive pressure between the fuel-moderator and the cladding if the fuel temperature exceeds the safety limit. The pressure is caused by the presence of air, fission product gases, and hydrogen from the dissociation of the hydrogen and zirconium in the fuel-moderator. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium in the alloy.

The safety limit for the TRIGA fuel is based on data which indicates that the stress in the cladding due to hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress, provided that the temperature of the fuel does not exceed 1,000°C and the fuel cladding is water cooled.

## 2.2. LIMITING SAFETY SYSTEM SETTING FOR FUEL TEMPERATURE

Applicability

This specification applies to the scram settings which prevent the safety limit from being reached.

Objective

The objective is to prevent the safety limit from being reached.

Specification

The limiting safety system setting shall be equal to or less than 600°C as measured in the instrumented fuel elements. There shall be two fuel temperature safety channels. One channel shall utilize an instrumented fuel element in the B ring, and the second channel shall utilize an instrumented fuel element in the C ring.

Basis

The limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated, preventing the safety limit from being exceeded. A setting of 600°C provides a safety margin of 400°C for TRIGA fuel elements. Part of the safety margin is used to account for the difference between the true and the measured temperatures resulting from the actual location of the thermocouple. If the instrumented fuel element is located in the hottest position in the core, the difference between the true and measured temperatures will be only a few degrees. There are two fuel temperature monitoring channels within the reactor core (one in the B ring and one in the C ring). The highest power density occurs in these two rings, and therefore provides temperature monitoring in the hottest locations of the reactor core. Table 4-14 of the AFRRI SAR identifies the rod power factors for each fuel location in the reactor core. Within the B ring, the highest and lowest power factors are 1.552 and 1.525, respectively. Assuming the instrumented fuel element is located in the lowest power density position (B01), a temperature indication of 600°C would yield a peak temperature at the highest power density location (B04) of 611°C. Within the C ring, the highest and lowest power factors are 1.438 and 1.374, respectively. Assuming the instrumented fuel element is located in the lowest power density position (C12), a temperature indication of 600°C would yield a peak temperature at the highest power density location (C09) of 628°C.

# 3.0. LIMITING CONDITIONS FOR OPERATIONS

## 3.1. REACTOR CORE PARAMETERS

### 3.1.1. STEADY STATE OPERATION

Applicability

This specification applies to the maximum reactor power attained during steady state operation.

Objective

The objective is to ensure that the fuel temperature safety limit shall not be exceeded during steady state operation.

Specification

The reactor steady state power level shall not exceed 1.1 MW.

Basis

The thermal-hydraulic analysis of steady state operation using the RELAP5 computer code, as detailed in the AFRRI SAR, shows that the reactor may be safely operated with TRIGA fuel at a power level of 1.1 MW.

### 3.1.2. PULSE MODE OPERATION

Applicability

This specification applies to the maximum thermal energy produced in the reactor as a result of a prompt critical insertion of reactivity.

Objective

The objective is to ensure that the fuel temperature safety limit shall not be exceeded during pulse mode operation.

Specification

The maximum step insertion of reactivity shall be $3.50 (2.45% $\Delta k/k$) in the pulse mode.

Basis

Based upon the calculations detailed in the AFRRI SAR an insertion of \$3.50 (2.45% $\Delta k/k$)results in a peak fuel temperature of less than 830°C.

### 3.1.3. REACTIVITY LIMITATIONS

Applicability

These specifications apply to the reactivity condition of the reactor and the reactivity worth of control rods and experiments. They apply for all modes of operation.

Objective

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

Specifications

a. The reactor shall not be operated with the maximum available excess reactivity greater than \$5.00 (3.5% $\Delta k/k$).

b. The shutdown margin provided by the remaining control rods with the most reactive control rod in the most reactive position shall be greater than \$0.50 (0.35% $\Delta k/k$) with the reactor in the reference core condition, all irradiation facilities and experiments in place, and the total worth of all non-secured experiments in their most reactive state.

Bases

a. The limit on available excess reactivity establishes the maximum achievable power if all control rods are removed.

b. The value of the shutdown margin ensures that the reactor can be shut down from any operating condition even if the most reactive control rod should remain in the most reactive position.

### 3.1.4. SCRAM TIME

Applicability

The specification applies to the time required to fully insert any standard control rod/transient rod to a full down position from a full up position.

Objective

The objective is to achieve rapid shutdown of the reactor to prevent fuel damage.

Specification

The time from scram initiation to the full insertion of any control rod from a full up position shall be less than 1 second.

Basis

This specification ensures that the reactor will be promptly shut down when a scram signal is initiated. Experience and analysis indicate that, for the range of transients for a TRIGA reactor, the specified scram time is adequate to ensure the safety of the reactor.

## 3.2. REACTOR CONTROL AND SAFETY SYSTEMS

### 3.2.1. REACTOR CONTROL SYSTEM

Applicability

This specification applies to the channels monitoring the reactor core, which must provide information to the reactor operator during reactor operation. It also specifies the minimum number of operable control rod drives.

Objective

The objective is to require that sufficient information be available to the operator as well as a sufficient number of operable control rod drives to ensure safe operation of the reactor.

Specifications

a. The reactor shall not be operated unless the measuring channels listed in Table 1 are operable for the specific mode of operation.

b. The reactor shall not be operated unless the four control rod drives specified in Section 5.2.2.b. are operable or fully inserted.

**Table 1 Minimum Measuring Channels**

| Measuring Channel | Effective Mode | |
|---|---|---|
| | Steady State | Pulse |
| Fuel Temperature Safety Channel | 2 | 2 |
| Linear Power Channel | 1 | 0 |
| Log Power Channel | 1 | 0 |
| High-Flux Safety Channel | 2 | 1 |
| Power Pulsing Channel | 0 | 1 |

(1) Any single Linear Power, Log Power, High-Flux Safety or Fuel Temperature Safety Channel may be inoperable while the reactor is operating for the purpose of performing a channel check, test, or calibration.

(2) If any required measuring channel becomes inoperable while the reactor is operating for reasons other than that identified in the previous footnote (1) above, the channel shall be restored to operation within five minutes or the reactor shall be immediately shutdown.

<u>Bases</u>

Fuel temperature displayed at the control console gives continuous information on this parameter, which has a specified safety limit. The power level channels ensure that radiation indicating reactor core parameters are adequately monitored for both steady state and pulsing modes of operation. The specifications on reactor power level indication are included in this section, since the power level is related to the fuel temperature. The four control rod drives must be operable or inserted for the safe operation of the reactor. For footnote (1), taking these measuring channels off-line for short durations for the purpose of a check, test, or calibration is considered acceptable because in some cases, the reactor must be in operation in order to perform the check, test or calibration. Additionally, there exist two redundant power level indications operating at any given time while the third single channel is off-line. For footnote (2), events which lead to these circumstances are self-revealing to the operator.

3.2.2. <u>REACTOR SAFETY SYSTEM</u>

<u>Applicability</u>

This specification applies to the reactor safety system.

<u>Objective</u>

The objective is to specify 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 Tables 2 and 3 are operable for the specific mode of operation.

**Table 2 Minimum Reactor Safety System Scrams**

| Channel | Maximum Set Point | Effective Mode | |
|---|---|---|---|
| | | Steady State | Pulse |
| Fuel Temperature | 600°C | 2 | 2 |
| Percent Power, High Flux | 1.1 MW | 2 | 0 |
| Console Manual Scram Button | Closure switch | 1 | 1 |
| High Voltage Loss to Safety Channel | 20% Loss | 2 | 1 |
| Pulse Time | 15 seconds | 0 | 1 |
| Emergency Stop | Closure switch | 1 | 1 |
| (1 in each exposure room, 1 on console) | | | |
| Pool Water Level | 14 feet from the top of the core | 1 | 1 |
| Watchdog (DAC to CSC) | On digital console | 1 | 1 |

(1) Any single Linear Power Level, Log Power Level, Power Level measuring or Fuel Temperature Safety Channel may be inoperable while the reactor is operating for the purpose of performing a channel check, test, or calibration

(2) If any required measuring channel becomes inoperable while the reactor is operating for reasons other than that identified in the previous footnote (1) above, the channel shall be restored to operation within five minutes or the reactor shall be immediately shutdown.

Basis

The fuel temperature and power level scrams provide protection to ensure that the reactor can be shut down before the fuel temperature safety limit is exceeded. The manual scram allows the operator to shut down the system at any time if an unsafe or abnormal condition occurs. In the event of failure of the power supply for the safety channels, operation of the reactor without adequate instrumentation is prevented. The preset timer ensures that the reactor power level will reduce to a low level after pulsing. The emergency stop allows personnel trapped in a potentially hazardous exposure room or the reactor operator to stop actions through the interlock system. The pool water level ensures that a loss of biological shielding would result in a reactor shutdown. The watchdog scram will ensure reliable communication between the Data Acquisition

Computer (DAC) and the Control System Computer (CSC). For footnote (1), taking these measuring channels off-line for short durations for the purpose of a check, test, or calibration is considered acceptable because in some cases, the reactor must be in operation in order to perform the check, test or calibration. Additionally, there exist two redundant power level indications operating at any given time while the third single channel is off-line. For footnote (2), events which lead to these circumstances are self-revealing to the operator .

**Table 3 Minimum Reactor Safety System Interlocks**

| **Action Prevented** | **Effective Mode** | |
|---|---|---|
| | **Steady State** | **Pulse** |
| Pulse initiation at power levels greater than 1 kW | | X |
| Withdrawal of any control rod except transient | | X |
| Any rod withdrawal with count rate in the linear and log power channels below 0.5 cps | X | X |
| Simultaneous manual withdrawal of two standard rods | X | |
| Any rod withdrawal if high voltage is lost to the operational channel | X | X |
| Withdrawal of any control rod if reactor period is less than 3 seconds | X | |

Basis

The interlock preventing the initiation of a pulse at a critical level above 1 kW ensures that the pulse magnitude will not allow the fuel element temperature to exceed the safety limit. The interlock that prevents movement of standard control rods in pulse mode will prevent the inadvertent placing of the reactor on a positive period while in pulse mode. Requiring a minimum count rate to be measured by the operational channel(s) ensures sufficient source neutrons to bring the reactor critical under controlled conditions. The interlock that prevents the simultaneous manual withdrawal of two standard control rods limits the amount of reactivity added per unit time. High voltage to the operational channel ensures accurate power indications to the control systems and the console. Preventing the withdrawal of any control rod if the period is less than 3 seconds minimizes the possibility of exceeding the maximum permissible power or the fuel temperature safety limit.

3.2.3. FACILITY INTERLOCK SYSTEM

Applicability

This specification applies to the interlocks that prevent the accidental exposure of an individual in either exposure room.

Objective

The objective is to provide sufficient warning and interlocks to prevent movement of the reactor core to the exposure room in which someone may be working, or prevent the inadvertent movement of the core into the lead shield doors.

Specifications

Facility interlocks shall be provided so that:

a. The reactor cannot be operated unless the shield doors within the reactor pool are either fully opened or fully closed;

b. The reactor cannot be operated unless the exposure room plug door adjacent to the reactor core position is fully closed and the lead shield doors are fully closed; or if the lead shield doors are fully opened, both exposure rooms plug doors must be fully closed; and

c. The lead shield doors cannot be opened to allow movement into the exposure room projection unless a warning horn has sounded in that exposure room, or unless two licensed reactor operators have visually inspected the room to ensure that no personnel remain in the room prior to securing the plug door.

Bases

These interlocks prevent the operation and movement of the reactor core into an area until there is assurance that inadvertent exposures will be prevented.

## 3.3. COOLANT SYSTEMS

Applicability

This specification refers to operation of the reactor with respect to temperature and condition of the pool water.

Objective

a. To ensure the effectiveness of the resins in the water purification system;

b. To prevent activated contaminants from becoming a radiological hazard; and

c. To help prevent corrosion of fuel cladding and other components in the primary system.

Specifications

a. The reactor shall not be operated above a thermal power of 5 kW when the purification system inlet water temperature exceeds 60°C;

b. The reactor shall not be operated if the conductivity of the water is greater than 2 micromhos/cm (or less than $0.5 \times 10^6$ ohms-cm resistance) at the output of the purification system, averaged over one week;

c. The reactor shall not be operated if the conductivity of the bulk water is greater than 5 micromhos/cm (or less than $0.2 \times 10^6$ ohms-cm resistance), averaged over one week; and

d. Both audible and visual alarms shall be provided to alert the AFRRI security guards and other personnel to any drop in reactor pool water level greater than 6 inches.

Bases

Manufacturer's data states that the resins in the water purification system break down with sustained operation in excess of 60°C. The 2 micromhos/cm is an acceptable level of water contaminants in an aluminum/stainless-steel system of the type at AFRRI. Based on experience, activation at this level does not pose a significant radiological hazard. Also, the conductivity limits are consistent with the fuel vendor's experience and with similar reactors.

## 3.4. VENTILATION SYSTEM

Applicability

This specification applies to the operation of the facility ventilation system.

Objective

The objective is to ensure that the ventilation system is operable to mitigate the consequences of possible releases of radioactive material resulting from reactor operation.

Specification

The reactor shall not be operated unless the facility ventilation system is operating, except for periods of time not to exceed two (2) hours to permit repair, maintenance, or testing. In the event of a release of airborne radioactivity in the reactor room above both routine reactor operation and normal background values, the ventilation system to the reactor room shall be automatically secured via closure dampers by a signal from the reactor deck air particulate monitor.

Basis

During normal operation of the ventilation system, the concentration of argon-41 in unrestricted areas is below the limits allowed by 10 CFR 20. In the event of a fuel cladding rupture resulting in a substantial release of airborne particulate radioactivity, the ventilation system shall be shut down, thereby isolating the reactor room automatically by spring-loaded, positive sealing dampers. Therefore, operation of the reactor with the ventilation system shut down for short periods of time ensures the same degree of control of release of radioactive materials. Moreover, radiation monitors within the building independent of those in the ventilation system provide warning should high levels of radiation occur during operation with the ventilation system secured.

## 3.5. RADIATION-MONITORING SYSTEM AND EFFLUENTS

### 3.5.1. MONITORING SYSTEM

Applicability

This specification applies to the functions and essential components of the area radiation monitoring equipment and the system for continuously monitoring radioactivity and radiation levels, which must be available during reactor operations.

Objective

The objective is to ensure that adequate radiation-monitoring equipment and radiation information are available to the operator to ensure safe operation of the reactor.

Specifications

The reactor shall not be operated unless the following radiation-monitoring systems are operable:

a. Radiation Area Monitoring System. The radiation area monitoring (RAM) system shall have two detectors located in the reactor room, and one detector placed near each exposure room plug door so that streaming radiation will be detected;

b. Stack Gas Monitor. The stack gas monitor (SGM) will sample and measure the gaseous effluent in the building exhaust system;

c. Air Particulate Monitor. The air particulate monitor (CAM) will sample the air above the reactor pool. This unit will be sensitive to radioactive particulate matter. Alarm of this unit will cause closure of the positive sealing dampers, restricting air leakage from the reactor room; and

d. Table 4 specifies the alarm and readout system for the above monitors.

**Table 4 Locations of Radiation Monitoring Systems**

| Sampling Location | Readout Location(s) (Audible and Visual) |
|---|---|
| **RAM** | |
| Reactor Room (2 required) | Reactor and Control Rooms |
| E3, Exp. Room 1 Area | Prep Area and Control Rooms |
| E6, Exp. Room 2 Area | Prep Area and Control Rooms |
| **SGM** | |
| Reactor Exhaust | Reactor and Control Rooms |
| **CAM** | |
| Reactor Room | Reactor and Control Rooms |

Bases

This system is intended to characterize the normal operational radiological environment of the facility and to aid in evaluating any abnormal operations or conditions. The radiation monitors provide information to the operating personnel of any existing or impending danger from radiation, to give sufficient time to evacuate the facility and take necessary steps to prevent the spread of radioactivity to the surroundings. The automatic closure of the ventilation system dampers restricts the flow of airborne radioactive material to the environment.

3.5.2. EFFLUENTS: ARGON-41 DISCHARGE LIMIT

Applicability

This specification applies to the quantity of argon-41 that may be discharged from the TRIGA reactor facility.

Objective

The objective is to ensure the health and safety of the public are not endangered by the discharge of argon-41 from the TRIGA reactor facility.

Specifications

a. An environmental radiation-monitoring program shall be maintained to determine effects of the facility on the environs.

b. If calculations, which shall be performed at least semiannually indicate that an exposure of 9 millirem above background to the unrestricted environment has been reached during the year as a result of reactor operations, reactor operations that generate and release measurable quantities of argon-41 shall cease for the remainder of the calendar year.

Bases

Since argon-41 does not represent an uptake or bioaccumulation problem, only the direct exposure modality is pertinent with regard to limiting reactor operations. Since direct plume shine may be more controlling than submersion conditions, cumulative exposure is the more appropriate quantification of this limit than the air concentration values in Table 2, Appendix B, 10 CFR 20.

## 3.6. LIMITATIONS ON EXPERIMENTS

Applicability

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

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment malfunction, so that airborne concentrations of activity averaged over a year do not exceed 10 CFR 20, Appendix B.

Specifications

The following limitations shall apply to the irradiation of materials (other than air):

a. If the possibility exists that a release of radioactive gases or aerosols may occur, the amount and type of material irradiated shall be limited to ensure yearly compliance with Table 2, Appendix B, of 10 CFR 20, assuming that 100% of the gases or aerosols escape;

b. Each fueled experiment shall be limited so that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 1.0 curies and the maximum strontium-90 inventory is not greater than 5 millicuries;

c. Known explosive materials shall not be irradiated in the reactor in quantities greater than 25 milligrams. In addition, the pressure produced in the experiment container upon detonation of the explosive shall have been determined experimentally, or by calculations, to be less than half the design failure of the container;

d. Samples shall be doubly contained when release of the contained material could cause corrosion of the experimental facility or damage to the reactor;

e. The sum of the absolute reactivity worths of all experiments in the reactor and in the associated experimental facilities shall not exceed \$3.00 (2.1% $\Delta k/k$). This includes the total potential reactivity insertion that might result from experiment malfunction, accidental experiment flooding or voiding, and accidental removal or insertion of experiments. The absolute reactivity worth of any single secured experiment shall not exceed \$3.00 (2.1% $\Delta k/k$). The absolute reactivity worth of any single moveable or unsecured experiment shall be less than \$1.00 (0.70% $\Delta k/k$). The combined absolute reactivity worths of multiple moveable or unsecured in the reactor and associated experimental facilities at the same time shall be less than \$1.00 (0.70% $\Delta k/k$);

f. In calculations regarding experiments, the following assumptions shall be made:

   1. If the effluent exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of the particles produced can escape; and

   2. For a material whose boiling point is above 55°C and whose vapor (formed by boiling the material) can escape only through a column of water above the core, up to 10% of the vapor is permitted to escape;

g. If a capsule fails and releases materials that could damage the reactor fuel or structure by corrosion or other means, physical inspection shall be performed to identify damage and potential need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the Reactor Facility Director, and shall be determined to be satisfactory before operation of the reactor is resumed;

h. All experiments placed in the reactor exposure environment shall be either firmly secured or observed by a reactor operator for mechanical stability, to ensure that unintended movement will not cause an unplanned reactivity change or physical damage. Experiments shall be designed such that failure of one experiment cannot contribute to the failure of any other experiment. All operations in any experimental facilities shall be supervised by a member of the reactor operations staff.

Bases

a. This specification is intended to provide assurance that airborne activities in excess of the limits of Appendix B of 10 CFR 20 will not be released to the atmosphere outside the facility boundary.

b. The 1.0 curie limitation on iodine-131 through -135 ensures that, in the event of malfunction of a fueled experiment leading to total release of radioactive material including fission products, the dose to any individual will not exceed the limits of 10 CFR 20.

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

d. This specification is intended to provide an additional safety factor where damage to the reactor and components is possible if a capsule fails.

e. The maximum worth of experiments is limited so that their removal from the reactor at the reference core condition will not result in the reactor achieving a power level high enough to exceed the core temperature safety limit. The three dollar ($3.00) limit is less than the SAR-analyzed authorized pulse magnitude. Limiting moveable or unsecured experiments to a worth less than $1.00 will prevent unintended pulsing of the reactor and unnecessary fuel mechanical stress.

f. This specification is intended to ensure that the limits of 10 CFR 20, Appendix B, are not exceeded if an experiment malfunctions.

g. This specification is intended to ensure that operation of the reactor with damaged reactor fuel or structure is prevented.

h. All experiments placed in the reactor environment shall be either firmly secured or observed for mechanical stability to ensure that unintended movement will not cause an unplanned reactivity change or physical damage or contribute to the failure of any other experiment.

## 3.7. SYSTEM MODIFICATIONS

Applicability

This specification applies to any system related to reactor safety.

Objective

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

Specification

Any additions or modifications to SAR-described systems including the ventilation system, core and its associated support structure, pool, coolant system, rod drive mechanism, or reactor safety system shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated, or to specifications approved by the Reactor and Radiation Facilities Safety Subcommittee. A system shall not be considered operable until after it is successfully tested.

Basis

This specification is related to changes in reactor systems that could directly affect the safety of the reactor. As long as changes or replacements to these systems continue to meet the original design specifications, they meet the presently accepted operating criteria.

3.8. ALARA

Applicability

This specification applies to all reactor operations that could result in significant personnel exposures.

Objective

The objective is to maintain all exposures to ionizing radiation to the staff and the general public as low as is reasonably achievable.

Specification

As part of the review of all operations, consideration shall be given to alternative operational profiles that might reduce staff exposures, release of radioactive materials to the environment, or both.

Basis

Experience has shown that scheduling of experiments and operational requirements can in many cases be satisfied with a variety of combinations of facility options, core positions, power levels, time delays, and other modifying factors. Many of these can reduce radioactive effluents and staff radiation exposures. Consequently, ALARA must be a part of both the overall reactor scheduling and the detailed experiment planning.

## 3.9. FUEL PARAMETERS

Applicability

This specification applies to all fuel elements.

Objective

The objective is to maintain integrity of the fuel element cladding.

Specification

The reactor shall not operate with damaged fuel elements, except for the purpose of locating damaged fuel elements. A fuel element shall be considered damaged and must be removed from the core if:

a. The transverse bend exceeds 0.0625 inches over the length of the cladding;

b. The length exceeds its original length by 0.100 inches;

c. A cladding defect exists as indicated by the release of fission products; or

d. Visual inspection identifies bulges, gross pitting, or corrosion.

Basis

Gross failure or obvious visual deterioration of the fuel is sufficient to warrant declaration of the fuel element as damaged. The elongation and bend limits are the values found acceptable to the USNRC (NUREG-1537).

## 4.0. SURVEILLANCE REQUIREMENTS

No surveillance requirements shall be deferred during normal reactor operational periods. Any surveillance requirements that cannot be performed due to a reactor outage shall be performed prior to resuming normal reactor operations.

### 4.1. REACTOR CORE PARAMETERS

Applicability

These specifications apply to the surveillance requirements for reactivity control of experiments and systems affecting reactivity.

Objective

The objective is to measure and verify the worth, performance, and operability of those systems affecting the reactivity of the reactor.

Specifications

a. The reactivity worth of each standard control rod/transient rod and the shutdown margin shall be determined annually but at intervals not to exceed 15 months or following any significant (>$0.25) core configuration changes.

b. The reactivity worth of an experiment shall be estimated before reactor power operation with an experiment the first time it is performed. If the absolute reactivity worth is estimated to be greater than $0.25, the worth shall be measured at a power level less than 1 kW.

c. The standard control rods/transient rod shall be visually inspected for damage and deterioration annually, not to exceed 15 months.

d. On each day that pulse mode operation of the reactor is planned, a functional performance check of the transient rod system shall be performed. Semiannually, at intervals not to exceed 7.5 months, the transient rod drive cylinder and the associated air supply system shall be inspected, cleaned, and lubricated as necessary.

e. The core excess reactivity shall be measured at the beginning of each day of operation involving the movement of control rods, or prior to each continuous operation extending more than a day, and following any significant (>$0.25) core configuration changes. During extended reactor shutdown periods, the core excess reactivity shall be measured at least annually, not to exceed 15 months.

f. The power coefficient of reactivity at 100 kW and 1 MW shall be measured annually, at intervals not to exceed 15 months.

Bases

The reactivity worth of the control rods is measured to ensure that the required shutdown margin is available and to provide an accurate means for determining the reactivity worth of experiments inserted in the core.

Past experience with TRIGA reactors gives assurance that measurement of the reactivity worth, on an annual basis, is adequate to ensure that no significant changes in the shutdown margin have occurred. Visual inspection of the standard control rods/transient rod is made to evaluate corrosion and wear characteristics caused by operation in the reactor. Functional checks along with periodic maintenance ensure consistent performance.

Excess reactivity measurements ensure that core configuration remains unchanged with no fallen material of reactive value near the core. Knowledge of power coefficients allows the operator to accurately predict the reactivity necessary to achieve required power levels.

## 4.2. REACTOR CONTROL AND SAFETY SYSTEMS

### 4.2.1. REACTOR CONTROL SYSTEMS

Applicability

These specifications apply to the surveillance requirements for reactor control systems.

Objective

The objective is to verify the operability of system components that affect the safe and proper control of the reactor.

Specification

The control rod drop times of all rods shall be measured semiannually, but at intervals not to exceed 7.5 months. After work is done on any rod or its rod drive mechanical components, the drop time of that particular rod shall be verified.

Basis

Measurement of the scram time on a semiannual basis or after mechanical maintenance is a verification of the scram system, and is an indication of the capability of the control rods to perform properly.

### 4.2.2. REACTOR SAFETY SYSTEMS

Applicability

These specifications apply to the surveillance requirements for measurements, tests, and calibrations of the reactor safety systems.

Objective

The objective is to verify the performance and operability of the systems and components that are directly related to reactor safety.

Specifications

a. A channel check of the scram function of the high-flux safety channels shall be made on each day that the reactor is to be operated.

b. A channel test of each of the reactor safety system channels for the intended mode of operation shall be performed weekly, whenever operations are planned.

c. Channel calibration shall be made of the power level-monitoring channels annually, at intervals not to exceed 15 months.

d. A thermal power calibration shall be completed annually, at intervals not to exceed 15 months.

Bases

TRIGA system components have operational proven reliability. Daily checks ensure accurate scram functions. Weekly channel testing is sufficient to ensure the detection of possible channel drift or other possible deterioration of operating characteristics. The channel checks will ensure that the safety system channel scrams are operable on a daily basis or prior to an extended run. The power level channel calibration will ensure that the reactor is to be operated at the authorized power levels.

### 4.2.3. FUEL TEMPERATURE

Applicability

These specifications apply to the surveillance requirements for the safety channels measuring the fuel temperature.

Objective

To ensure operability of the fuel temperature-measuring channels.

Specifications

a. A channel check of the fuel temperature scrams shall be made on each day that the reactor is operated.

b. A channel calibration of the fuel temperature-measuring channels shall be made annually, at intervals not to exceed 15 months.

c. A weekly channel test shall be performed on fuel temperature-measuring channels, whenever operations are planned.

d. If a reactor scram caused by high fuel element temperature occurs, an evaluation shall be conducted to determine whether the fuel element temperature actually exceeded the safety limit.

Bases

Operational experience with the TRIGA system ensures that the thermocouple measurements have been sufficiently reliable as an indicator of fuel temperature with proven reliability. The weekly channel test ensures operability and indication of fuel temperature. The daily scram channel check ensures scram capabilities.

4.2.4. FACILITY INTERLOCK SYSTEM

Applicability

This specification applies to the surveillance requirements that ensure the integrity of the facility interlock system.

Objective

To ensure performance and operability of the facility interlock system.

Specifications

a. Functional checks shall be made annually, but not to exceed 15 months, to ensure the following:

   1. With the lead shield doors open, neither exposure room plug door can be electrically opened.

2. The core dolly cannot be moved into position 2 with the lead shield doors closed.

3. The warning horn shall sound in the exposure room before opening the lead shield doors, which allows the core to move to that exposure room unless cleared by two licensed operators.

Bases

These functional checks will verify operation of the interlock system. Experience at AFRRI indicates that this is adequate to ensure operability.

4.2.5. REACTOR FUEL ELEMENTS

Applicability

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

Objective

The objective is to verify the specifications for fuel elements are met.

Specification

Fuel elements shall be inspected visually for damage or deterioration and measured for length and bend in accordance with the following:

a. Before being placed in the core for the first time or following long-term storage;

b. Every two years (not to exceed 30 months), or at intervals not to exceed 500 pulses of insertion greater than $2.00, whichever comes first, for fuel elements in the B, C, and D rings;

c. Every four years (not to exceed 54 months) for fuel elements in the E and F rings; and

d. If damage, deterioration, or unacceptable length and bend measurements are found in one or more fuel elements, all fuel elements in the core shall be inspected for damage or deterioration and measured for length and bend.

Basis

The frequency of inspection and measurement is based on the parameters most likely to affect the fuel cladding of a pulse reactor. Inspecting fuel elements in rings with higher power factors will provide early indication of fuel damage while significantly reducing the amount of fuel movement required.

## 4.3. COOLANT SYSTEMS

Applicability

This specification applies to the surveillance requirements for monitoring the pool water and the water-conditioning system.

Objective

The objective is to ensure the integrity of the water purification system, thus maintaining the purity of the reactor pool water, eliminating possible radiation hazards from activated impurities in the water system, and limiting the potential corrosion of fuel cladding and other components in the primary water system.

Specifications

a. The pool water temperature, as measured near the input to the water purification system, shall be measured daily, whenever operations are planned.

b. The conductivity of the water at the output of the purification system shall be measured weekly, whenever operations are planned.

c. The reactor coolant shall be analyzed for radioactivity at least annually.

d. The audible and visual reactor pool level alarms in hallway 3101 shall be tested quarterly, not to exceed four months.

Bases

Based on experience, observation at these intervals provides acceptable surveillance of limits that ensure that fuel cladding corrosion and neutron activation of dissolved materials are minimized. Testing of the audible and visual alarms ensures that personnel will be able to detect and respond to pool water loss in a timely manner. The pool water temperature is continuously displayed on the reactor console and is manually recorded at the beginning of each day of reactor operations. The conductivity of the bulk pool water is restricted to help minimize the activation of impurities in the water system and monitor the possibility of corrosion in the fuel cladding or reactor system components.

## 4.4. VENTILATION SYSTEM

Applicability

This specification applies to the facility ventilation system isolation.

Objective

The objective is to ensure the proper operation of the ventilation system in controlling the release of radioactive material into the unrestricted environment.

Specification

The operating mechanism of the positive sealing dampers in the reactor room ventilation system shall be verified to be operable and visually inspected at least monthly (interval not to exceed six weeks).

Basis

Experience accumulated over years of operation has demonstrated that the tests of the ventilation system on a monthly basis are sufficient to ensure proper operation of the system and control of the release of radioactive material.

## 4.5. RADIATION-MONITORING SYSTEM

Applicability

This specification applies to surveillance requirements for the radiation area monitoring equipment and the air particulate monitoring system.

Objective

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

Specification

The radiation area monitoring system and the air particulate monitoring system shall be channel tested quarterly, but at intervals not to exceed four months. A channel check of both systems shall be performed daily to verify operability when operations are planned. Both systems shall be calibrated annually, not to exceed 15 months.

Basis

Experience has shown that quarterly verification of radiation area monitoring and air particulate monitoring system set points in conjunction with a quarterly channel test is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span. Annual calibration ensures that the units are within the specifications demanded by the extent of use.

## 4.6. EFFLUENTS

Applicability

This specification applies to surveillance requirements for environmental monitoring.

Objective

The objective is to ensure the health and safety of the public through detection of any release of radioactive material to the environment.

Specifications

a. The unrestricted area outside of AFRRI shall be monitored by thermoluminescent dosimeters that shall be changed quarterly.

b. Samples of soil, vegetation, and water in the vicinity of the reactor shall be collected and tested for radioactivity quarterly.

Basis

Experience has shown that quarterly environmental monitoring is sufficient to detect and quantify any release of radioactive material from research reactors.

## 5.0. DESIGN FEATURES

### 5.1. SITE AND FACILITY DESCRIPTION

Applicability

This specification applies to the building that houses the reactor.

Objective

The objective is to restrict the amount of radioactivity released into the environment.

Specifications

a. The reactor building, as a structurally independent building in the AFRRI complex, shall have its own ventilation system branch. The effluent from the reactor ventilation system shall exhaust through absolute filters to a stack having a minimum elevation that is 18 feet above the roof of the highest building in the AFRRI complex.

b. The reactor room shall contain a minimum free volume of 22,000 cubic feet.

c. The ventilation system air ducts to the reactor room shall be equipped with positive sealing dampers that are activated by fail-safe controls, which will automatically close off ventilation to the reactor room upon a signal from the reactor room air particulate monitor.

d. The reactor room shall be designed to restrict air leakage when the positive sealing dampers are closed.

e. The reactor areas exhausting through the reactor ventilation system shall include the Controlled Access Area and the Reactor Control Areas. The specific rooms included in each of those areas shall be listed in the reactor Physical Security Plan.

Bases

The facility is designed so that the ventilation will normally maintain a negative pressure with respect to the atmosphere, so that there will be no uncontrolled leakage to the environment. The free air volume within the reactor building is confined when there is an emergency shutdown of the ventilation system. Building construction and gaskets around doorways help restrict leakage of air into or out of the reactor room. The stack height ensures an adequate dilution of effluents well above ground level. The separate ventilation system branch ensures a dedicated air flow system for reactor effluents and shall exhaust from all reactor-related spaces.

## 5.2. REACTOR CORE AND FUEL

### 5.2.1. REACTOR FUEL

Applicability

These specifications apply to the fuel elements, to include fuel follower control rods, used in the reactor core.

Objective

These objectives are to (1) ensure that the fuel elements are designed 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 (2) ensure that the fuel elements used in the core are substantially those analyzed in the Safety Analysis Report.

Specifications

The individual non-irradiated TRIGA fuel elements shall have the following characteristics:

a. Uranium content: Maximum of 9.0 weight percent enriched to less than 20% uranium-235. In the fuel follower, the maximum uranium content will be 12.0 weight percent enriched to less than 20% uranium-235.

b. Hydrogen-to-zirconium atom ratio (in the $ZrH_x$): Nominal 1.7 H atoms to 1.0 Zr atoms with a range between 1.6 and 1.7.

c. Cladding: 304 stainless steel, nominal 0.020 inch thick.

d. Any burnable poison used for the specific purpose of compensating for fuel burnup or long-term reactivity adjustments shall be an integral part of the manufactured fuel elements.

Bases

A maximum uranium content of 9 weight percent in a TRIGA element is greater than the design value of 8.5 weight percent, and encompasses the maximum probable variation in individual elements. Such an increase in loading would result in an increase in power density of less than 6%. The hydrogen-to-zirconium ratio of 1.7 will produce a maximum pressure within the cladding that is well below the rupture strength of the cladding. The local power density of a 12.0 weight percent fuel follower is 21% greater than an 8.5 weight percent TRIGA fuel element in the D-ring. The volume of fuel in a fuel follower control rod is 56% of the volume of a TRIGA fuel element. Therefore, the actual power produced in the fuel follower rod is 33% less than the power produced in a TRIGA fuel element in the D-ring.

### 5.2.2. REACTOR CORE

Applicability

These specifications apply to the configuration of fuel and in-core experiments.

Objective

The objective is to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specifications

a. The reactor core shall consist of TRIGA reactor fuel elements in a close packed array and a minimum of two thermocouple instrumented TRIGA reactor fuel elements.

b. There shall be four single core positions occupied by the three standard control rods and transient rod, a neutron startup source with holder, and positions for possible in-core experiments.

c. The core shall be cooled by natural convection water flow.

d. In-core experiments shall not be placed in adjacent fuel positions of the B-ring and/or C-ring.

e. Fuel elements indicating an elongation greater than 0.100 inch, a lateral bending greater than 0.0625 inch or significant visible damage shall be considered damaged, and shall not be used in the reactor core.

Bases

TRIGA cores have been in use for decades, and their safe operational characteristics are well documented. 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 (a) ensure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment, and (b) ensure adequate coolant flow.

### 5.2.3. CONTROL RODS

Applicability

These specifications apply to the control rods used in the reactor core.

Objective

The objective is to ensure that the control rods are designed to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications

a. The standard control rods shall have scram capability, and shall contain borated graphite, $B_4C$ powder, or boron and its compounds in solid form as a poison in aluminum or stainless-steel cladding. These rods may have an aluminum, air, or fuel follower. If fuel followed, the fuel region will conform to Specification 5.2.1.

b. The transient control rod shall have scram capability, and shall contain borated graphite, $B_4C$ powder, or boron and its compounds in solid form as a poison in aluminum or stainless-steel cladding. This rod may incorporate an aluminum, poison, or air follower.

Bases

The poison requirements for the control rods are satisfied by using neutron absorbing borated graphite, $B_4C$ powder, or boron and its compounds. These materials must be contained in a suitable cladding material, such as aluminum or stainless steel, to ensure mechanical stability during movement and to isolate the poison from the pool water environment. Scram capabilities are provided by the rapid insertion of the control rods, which is the primary operational safety feature of the reactor. The transient control rod is designed for use in a pulsing TRIGA reactor.

## 5.3. SPECIAL NUCLEAR MATERIAL STORAGE

Applicability

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

Objective

The objective is to ensure that stored fuel does not become critical and does not reach an unsafe temperature.

Specification

All fuel elements not in the reactor core shall be stored and handled in accordance with applicable regulations. Irradiated fuel elements and fueled devices shall be stored in an array that will permit sufficient natural convective cooling by water or air, so that the fuel element or fueled device temperature will not exceed design values. Storage shall be such that groups of stored fuel elements will remain subcritical under all conditions of moderation and reflection in a configuration where $k_{eff}$ is no greater than 0.90.

Basis

The limits imposed by this specification are conservative and ensure safe storage and handling. Experience shows that approximately 67 TRIGA fuel elements in a closely packed array are required to achieve criticality. Calculations show that in the event of a full storage rack failure with all 12 elements falling in the most reactive nucleonic configuration, the mass would be less than that required for criticality. Therefore, under normal storage conditions, criticality cannot be reached.

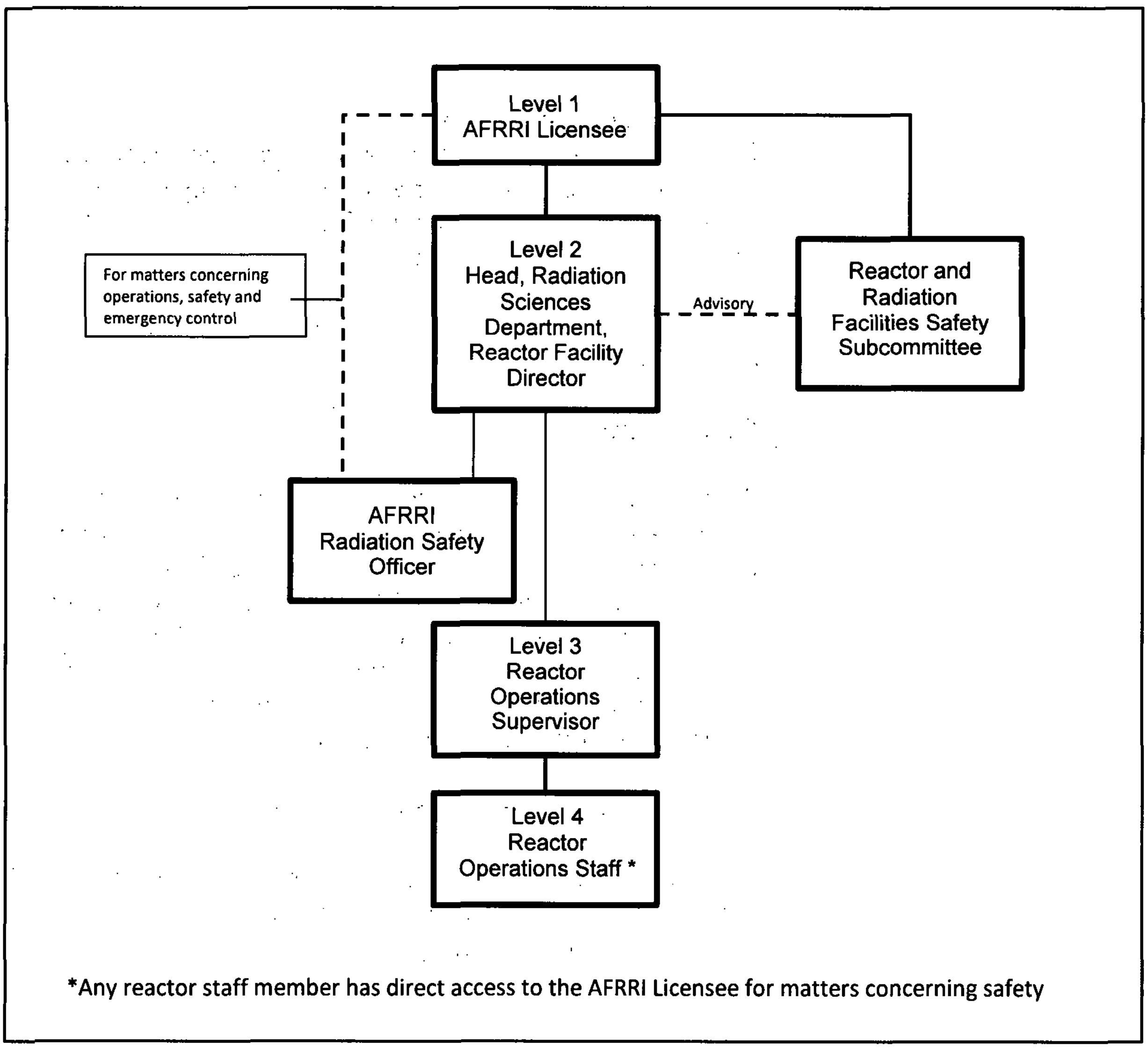
## 6.0. ADMINISTRATIVE CONTROLS

### 6.1. ORGANIZATION

#### 6.1.1. STRUCTURE

The organization of personnel for the management and operation of the AFRRI reactor facility is shown in Figure 1. Organizational changes may occur based on Institute requirements and will be depicted in internal documents. However, no changes may be made in the Operational, Safety, and Emergency Control Chain to alter the Reactor Facility Director having direct responsibility to the AFRRI Licensee.

Figure 1: Organization of Personnel for Management and Operation of the AFRRI Reactor Facility



### 6.1.2. RESPONSIBILITY

The AFRRI Licensee shall have license responsibility for the reactor facility. The Reactor Facility Director (RFD) shall be responsible for administration and operation of the Reactor Facility and for determination of applicability of procedures, experiment authorizations, maintenance, and operations. The RFD may designate an individual who meets the requirements of Section 6.1.3.1.a to discharge the RFD's responsibilities during an extended absence. During brief absences (periods less than four hours) of the Reactor Facility Director and his designee, the Reactor Operations Supervisor shall discharge these responsibilities. The Radiation Safety Officer shall implement a radiation protection program at AFRRI that satisfies the requirements of 10 CFR 20.

### 6.1.3. STAFFING

#### 6.1.3.1. Selection of Personnel

a. Reactor Facility Director

At the time of appointment to this position, the Reactor Facility Director shall have six or more years of nuclear experience. The individual shall have a baccalaureate or higher degree in an engineering or scientific field. The degree may fulfill up to four years of experience on a one-for-one basis. The Facility Director shall have held a USNRC Senior Reactor Operator license on the AFRRI reactor for at least one year before appointment to this position.

b. Reactor Operations Supervisor (ROS)

At the time of appointment to this position, the ROS shall have three years nuclear experience. Higher education in a scientific or engineering field may fulfill up to two years of experience on a one-for one basis. The ROS shall hold a USNRC Senior Reactor Operator license on the AFRRI reactor. In addition, the ROS shall have one year of experience as a USNRC licensed Senior Reactor Operator at AFRRI or at a similar facility before the appointment to this position.

c. Reactor Operators/Senior Reactor Operators

At the time of appointment to this position, an individual shall have a high school diploma or equivalent, and shall possess the appropriate USNRC license.

d. Additional reactor staff as required for support and training. At the time of appointment to the reactor staff, an individual shall possess a high school diploma or equivalent.

6.1.3.2. Operations

a. Minimum staff when the reactor is not secured shall include:

1. A licensed Senior Reactor Operator (SRO) on call, but not necessarily on site;

2. Radiation control technician on call, but not necessarily on site;

3. At least one licensed Reactor Operator (RO) or Senior Reactor Operator (SRO) present in the control room; and

4. Another person within the AFRRI complex who is able to carry out written emergency procedures, instructions of the operator, or to summon help in case the operator becomes incapacitated.

5. One licensed Senior Reactor Operator (SRO) may fill both the on call and control room positions simultaneously. In that case, the minimum staff is three persons.

b. An SRO shall be present at the reactor during the following operations:

1. All fuel or control rod relocations within the reactor core region;

2. Initial reactor startup and approach to power;

3. Recovery from an unplanned or unscheduled shutdown or significant power reduction; and

4. Relocation of any experiment with reactivity worth greater than $1.00.

c. A list of the names and telephone numbers of the following personnel shall be readily available to the operator on duty:

1. Management personnel (Reactor Facility Director, AFRRI Licensee) or designee;

2. Radiation safety personnel (AFRRI Radiation Safety Officer) or designee; and

3. Other operations personnel (Reactor Staff, ROS)

6.1.3.3. Training of Personnel

Training and retraining program shall be maintained to ensure adequate levels of proficiency in persons involved in the reactor and reactor operations.

## 6.2. REVIEW AND AUDIT - THE REACTOR AND RADIATION FACILITIES SAFETY SUBCOMMITTEE (RRFSS)

### 6.2.1. COMPOSITION AND QUALIFICATIONS

6.2.1.1. Composition

a. Regular RRFSS Members (Permanent Members)

1. The following shall be members of the RRFSS:

   a. AFRRI Radiation Safety Officer (RSO)

   b. AFRRI Reactor Facility Director (RFD)

   c. AFRRI Facility Radiation Manager (FRM)

2. The following shall be appointed to the RRFSS by the AFRRI Licensee:

   a. Chairman

   b. One to three non-AFRRI members who are knowledgeable in fields related to reactor safety. At least one shall be a Reactor Operations Specialist or a Health Physics Specialist.

b. Special RRFSS Members (Temporary Members)

1. Other knowledgeable persons to serve as alternates in section 6.2.1.1.a.2.b above as appointed by the AFRRI Licensee.

2. Voting ad hoc members, appointed by the AFRRI Licensee to assist in review of a particular problem.

c. Nonvoting members as appointed by the AFRRI Licensee.

6.2.1.2. Qualifications

The minimum qualifications for a person on the RRFSS shall be six years of professional experience in the discipline or specific field represented. A baccalaureate degree may fulfill four years of experience.

6.2.2. FUNCTION AND AUTHORITY

6.2.2.1. Function

The Reactor and Radiation Facilities Safety Subcommittee is directly responsible to the AFRRI Licensee. The committee shall review all radiological health and safety matters concerning the reactor and its associated equipment, the structural reactor facility, and those items listed in Section 6.2.4.

6.2.2.2. Authority

The RRFSS shall report to the AFRRI Licensee and shall advise the Reactor Facility Director in those areas of responsibility specified in Section 6.2.4.

6.2.3. CHARTER AND RULES

6.2.3.1. Alternates

Alternate members may be appointed in writing by the RRFSS Chairman to serve on a temporary basis. No more than two alternates shall participate on a voting basis in RRFSS activities at any one time.

6.2.3.2. Meeting Frequency

The RRFSS shall meet at least two times during a calendar year. Any member of the RRFSS may submit a written request to the RRFSS Chairman to convene a special meeting of the RRFSS to discuss urgent matters.

6.2.3.3. Quorum

A quorum of the RRFSS for review shall consist of the Chairman, the Reactor Facility Director (or designated alternate), the

Radiation Safety Officer (or designated alternate), the Facility Radiation Manager (or designated alternate), and one non-AFRRI member. A majority of those present shall be regular members.

6.2.3.4. Voting Rules

Each regular RRFSS member shall have one vote. Each special appointed member shall have one vote. The majority is 51% or more of the regular and special members present and voting and concurrence between the RSO and FRM.

6.2.3.5. Minutes

Minutes of the previous meeting should be available to regular members at least one week before a regular scheduled meeting.

6.2.4. REVIEW FUNCTION

The RRFSS shall review:

a. Safety evaluations for (1) changes to procedures, equipment, or systems having safety significance and (2) tests or experiments conducted without NRC approval under provisions of Section 50.59 of 10 CFR Part 50, to verify that such actions did not meet any of the criteria in paragraph (c)(2) of that Section;

b. Changes to procedures, equipment, or systems that change the original intent or use, and are non-conservative, or those that meet any of the criteria in paragraph (c) (2) of Section 50.59 of 10 CFR Part 50;

c. Proposed tests or experiments that are significantly different from previously approved tests or experiments, or those that might meet any of the criteria in paragraph (c)(2) of Section 50.59 of 10 CFR Part 50;

d. Proposed changes in technical specifications, the Safety Analysis Report, or other license conditions;

e. Violations of applicable statutes, codes, regulations, orders, technical specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant variations from normal and expected performance of facility equipment that might affect nuclear safety;

g. Events that have been reported to the NRC; and

g. Audit reports of the reactor facility operations.

6.2.5. AUDIT FUNCTION

Audits of reactor facility activities shall be performed under the cognizance of the RRFSS, but in no case by the personnel responsible for the item audited, annually not to exceed 15 months. A report of the findings and recommendations resulting from the audit shall be submitted to the AFRRI Licensee within three months after the audit has been completed. Audits may be performed by one individual who need not be an RRFSS member. These audits shall examine the operating records and the conduct of operations, and shall encompass the following:

a. Conformance of facility operation to the Technical Specifications and the license;

b. Performance, training, and qualifications of the reactor facility operations staff;

c. Results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems, or methods of operation that affect safety;

d. Facility emergency plan and implementing procedures;

e. Facility security plan and implementing procedures;

f. Any other area of facility operations considered appropriate by the RRFSS or the AFRRI Licensee; and

g. Reactor Facility ALARA Program. This program may be a section of the total AFRRI program.

6.3. PROCEDURES

Written instructions for certain activities shall be approved by the Reactor Facility Director and reviewed by the Reactor and Radiation Facilities Safety Subcommittee (RRFSS). The procedures shall be adequate to ensure safe operation of the reactor, but shall not preclude the use of independent judgment and action as deemed necessary. These activities are as follows:

a. Conduct of irradiation and experiments that could affect the operation and safety of the reactor;

b. Reactor staff training program;

c. Surveillance, testing, maintenance, and calibration of instruments, components, and systems involving nuclear safety;

d. Personnel radiation protection consistent with 10 CFR 20;

e. Implementation of required plans such as the Physical Security Plan and Emergency Plan, consistent with restrictions on Safeguards information;

f. Reactor core loading and unloading; and

g. Startup checklist, standard operations, and securing the facility.

Although substantive changes to the above procedures shall be made only with approval by the Reactor Facility Director, temporary changes to the procedures that do not change their original intent may be made by the ROS. All such temporary changes shall be documented and subsequently reviewed and approved by the Reactor Facility Director.

## 6.4. REVIEW AND APPROVAL OF EXPERIMENTS

Before issuance of a reactor authorization, new experiments shall be reviewed for radiological safety and approved by the following:

a. Reactor Facility Director

b. Health Physics Division

c. Reactor and Radiation Facilities Safety Subcommittee (RRFSS)

Prior to its performance, an experiment shall be included under one of the following types of authorizations:

a. Special Reactor Authorization for new experiments or experiments not included in a Routine Reactor Authorization. These experiments shall be performed under the direct supervision of the Reactor Facility Director or designee.

b. Routine Reactor Authorization for approved experiments safely performed at least once. These experiments may be performed at the discretion of the Reactor Facility Director and coordinated with the Health Physics Division when appropriate. These authorizations do not require additional RRFSS review.

c. Reactor Parameters Authorization for routine measurements of reactor parameters, routine core measurements, instrumentation and calibration checks, maintenance, operator training, tours, testing to verify reactor outputs, and other reactor testing procedures. This shall constitute a single authorization. These operations shall be performed under the authorization of the Reactor Facility Director or the Reactor Operations Supervisor.

Substantive (reactivity worth more than +/– $0.25) changes to previously approved experiments shall be made only after review by the RRFSS and after approval (in writing) by the Reactor Facility Director or designated alternate. Minor changes that do not significantly alter the experiment (reactivity worth of less than +/– $0.25) may be approved by the ROS. Approved experiments shall be carried out in accordance with established procedures.

## 6.5. REQUIRED ACTIONS

### 6.5.1. ACTIONS TO BE TAKEN IN CASE OF SAFETY LIMIT VIOLATION

a. The reactor shall be shut down immediately, and reactor operation shall not be resumed without authorization by the NRC.

b. The safety limit violation shall be reported to the NRC, the AFRRI Licensee, and the RRFSS not later than the next working day.

c. A Safety Limit Violation Report shall be prepared. This report shall be reviewed by the RRFSS, and shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation on facility components, structures, or systems, and (3) corrective action taken to prevent or reduce the probability of recurrence.

d. The Safety Limit Violation Report shall be submitted to the NRC, the AFRRI Licensee, and the RRFSS within 14 days of the violation.

### 6.5.2. REPORTABLE OCCURRENCES

The types of events listed below shall be reported as soon as possible by telephone and confirmed in writing by facsimile, e-mail, or similar transmission to the NRC no later than the following working day after confirmation of the event, with a written follow-up report within 14 days as per 10 CFR. The report shall include (as a minimum) the circumstances preceding the event, current effects on the facility, and status of corrective action. The report shall contain as much supplemental material as possible to clarify the situation. Supplemental reports may be required to fully describe the final resolution of the occurrence.

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

b. Operation in violation of any Limiting Condition for Operation, Section 3 unless prompt remedial action is taken.

c. Malfunction of a required reactor safety system component during operation that could render the system incapable of performing its intended safety function unless the malfunction or condition is caused by maintenance.

d. Any unanticipated or uncontrolled change in reactivity greater than $1.00.

e. An observed inadequacy in the implementation of either administrative or procedural controls, so that the inadequacy could have caused the existence or development of a condition that could result in operation of the reactor in a manner less safe than conditions covered in the Safety Analysis Report (SAR).

f. The release of fission products from a fuel element through degradation of the fuel cladding. Possible degradation may be determined through an increase in the background activity level of the reactor pool water.

g. Significant degradation of the reactor coolant boundary (excluding minor leaks).

h. A release of radioactivity that exceeds or could have exceeded the limits allowed by Title 10, Part 20 of the Code of Federal Regulations (10 CFR 20), or these technical specifications.

i. Unscheduled conditions arising from natural or man-made events that, as a direct result of the event, require operation of safety systems or other protective measures required by Technical Specifications.

j. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the Safety Analysis Report or in the bases for the Technical Specifications that have or could have permitted reactor operation with a smaller margin of safety than in the erroneous analysis.

k. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the Safety Analysis Report or Technical Specifications bases, or discovery during plant life of conditions not specifically considered in the Safety Analysis Report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

6.5.3. <u>ACTIONS TO BE TAKEN IN CASE OF REPORTABLE OCCURRENCES</u>

a. Reactor conditions shall be returned to normal, or the reactor shall be shut down. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by the Reactor Facility Director or designated alternate.

b. The occurrence shall be reported to the RFD or designated alternate and to the NRC.

c. The occurrence shall be reviewed by the RRFSS at its next scheduled meeting.

## 6.6. OPERATING REPORTS

In addition to the applicable reporting requirements of Title 10 of the Code of Federal Regulations, the following reports shall be submitted to NRC Office of Nuclear Reactor Regulation unless otherwise noted:

a. Startup Report: A summary report of planned startup and power escalation testing shall be submitted following (1) receipt of an operating license; (2) amendment of the license involving a planned increase in power level; (3) installation of fuel that has a different design; and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the reactor. The report shall address each of the tests identified in the Safety Analysis Report and shall, in general, include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Startup Reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of power operation, or (3) nine months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

b. Annual Operating Report: Routine operating reports covering the operation of the reactor during previous calendar year shall be submitted by March 31 of each year, covering the previous calendar year's operation. The Annual Operating Report shall provide a comprehensive summary of the operating experience having safety significance during the year, even though some repetition of previously reported information may be involved. References in the annual operating report to previously submitted reports shall be clear.

Each annual operating report shall include:

1. A brief narrative summary of:

    a. Changes in facility design, performance characteristics, and operating procedures related to reactor safety that occurred during the reporting period;

    b. Results of surveillance test and inspections;

2. A tabulation showing the energy generated by the reactor on a monthly basis, the cumulative total energy since initial criticality, and the number of pulses greater than $2.00;

3. List of the unscheduled shutdowns for which corrective was required to ensure safe operation of the reactor, including the reasons and the corrective actions taken;

4. Discussion of the major safety-related corrective maintenance performed during the period, including the effects (if any) on the safe operation of the reactor, and the reasons for the 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 Safety Analysis Report;

    b. Changes to the procedures as described in the Safety Analysis Report;

    c. Any new experiments or tests performed during the reporting period that is not encompassed in the Safety Analysis Report;

6. A summary of the safety evaluation made for each change, test, or experiment not submitted for Commission approval pursuant to Section 50.59 of 10 CFR Part 50. The summary shall clearly show the reason leading to the conclusion that the criteria in paragraph (c)(2) of that Section were not met and that no change to the Technical Specifications was required;

7. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as determined at or prior to the point of such release or discharge. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed, a statement to this effect is sufficient.

    a. <u>Liquid Waste</u> (summarized on a quarterly basis)

        i. Radioactivity discharged during the reporting period

Total radioactivity released (in curies);

Concentration limits used and isotopic composition if greater than 3 $\times 10^{-6}$ microcuries/ml for fission and activation products;

Total radioactivity (in curies), released by nuclide during the reporting period, and based on representative isotopic analysis Average concentration at point of release (in microcuries/cc) during the reporting period;

ii. Total volume (in gallons) of effluent water (including diluents) during periods of release;

b. Gaseous Waste (summarized on a quarterly basis)

Radioactivity discharged during the reporting period (in curies) for:

Argon-41;

Particulates with half-lives greater than eight days;

c. Solid Waste (summarized on a quarterly basis)

Total cubic feet of atomic number 3 to 83 materials in solid form disposed of under license R-84;

8. A description of the results of any environmental radiological surveys performed outside the facility;

9. A list of exposures greater than 25% of the allowed value (10 CFR 20) received by reactor personnel or visitors to the reactor facility;

c. Other Reports: A report shall be submitted within 30 days describing:

1. Any permanent change of either the AFRRI Licensee or the Reactor Facility Director; or

2. Significant changes in the transient or accident analysis described in the SAR.

## 6.7. RECORDS

### 6.7.1. RECORDS TO BE RETAINED FOR A PERIOD OF AT LEAST FIVE YEARS OR AS REQUIRED BY 10 CFR REGULATIONS

a. Operating logs or data that shall identify:

1. Completion of pre-startup checkout, startup, power changes, and shutdown of the reactor

2. Installation or removal of fuel elements, control rods, or experiments that could affect core reactivity

3. Installation or removal of jumpers, special tags, or notices of other temporary changes to bypass reactor safety circuitry

4. Rod worth measurements and other reactivity measurements

b. Principal maintenance operations

c. Reportable occurrences

d. Surveillance activities required by Technical Specifications

e. Facility radiation and contamination surveys

f. Experiments performed with the reactor

This requirement may be satisfied by the normal operations log book plus:

1. Records of radioactive material transferred from the Reactor Facility as required by license

2. Records required by the RRFSS for the performance of new or special experiments

g. Changes to operating procedures

h. Fuel inventories and fuel transfers

i. Records of transient or operational cycles for those components designed for limited number of transients or cycles

j. Records of training and qualification for members of the facility staff

k. Records of reviews performed for changes made to procedures or equipment, or reviews of tests and experiments pursuant to Section 50.59 of 10 CFR Part 50

l. Records of meetings of the RRFSS

6.7.2. <u>RECORDS TO BE RETAINED UNTIL AN OPERATOR'S LICENSE IS RENEWED OR CANCELLED, WHICHEVER OCCURS FIRST</u>

a. Training exams

b. Requalification records

6.7.3. RECORDS TO BE RETAINED FOR THE LIFE OF THE FACILITY

a. Gaseous and liquid radioactive effluents released to the environs

b. Appropriate offsite environmental monitoring surveys

c. Radiation exposures for all reactor personnel

d. Updated as-built drawings of the facility

e. Reviews and reports pertaining to a violation of the Safety Limit, a Limiting Safety System Setting, or an LCO