10CFR 50.4(B)(1)

50-170

ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE
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June 29, 2001

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Dear Sir:

The Armed Forces Radiobiology Research Institute (AFRRI) reactor facility requests to amend the Technical Specifications for license R-84 (Docket 50-170) with the changes described in Enclosure 1. A change bar in the right margin shows proposed changes. These changes are necessary to comply with the new 10 CFR 50.59 as well as to provide for more efficient reactor operation. These changes are also in preparation for submittal of the reactor-relicensing package in 2003 and we request that they be reviewed using relicensing criteria. In reformatting the document, numerous lines of text moved to different pages. We are therefore resubmitting the entire document (at Enclosure 2) as proposed Amendment 24.

The AFRRI Reactor and Radiation Facility Safety Committee accepted these proposed changes on June 27, 2001.

We request that the Technical Specifications be reissued in their entirety as Amendment 24.

Should you need any further information, please contact the undersigned at (301) 295-9245.

Enclosures:
as stated

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Technical Specifications
Summary of Proposed Changes

1. Section 1.5 - The definition of "Cold Critical" is changed to require only that both the fuel and bulk water temperatures be less than 40°C. The requirement that they be equal to each other is removed. An exact equality is not necessary. The 100 watts and 40°C are the important requirements.

2. Section 1.14c. - Change the requirement for arrival within 30 minutes to arrival within a reasonable time. As detailed in the AFRRI reactor Safety Analysis Report, there is no credible accident scenario involving the reactor where a slightly longer period would lead to more serious accident consequences.

3. Section 3.4 - In the second line of the specification, delete the reference to operating up to 48 hours to test or repair the ventilation system. Instead, change the sentence to allow any operations for a "brief" period as long as the positive sealing dampers remain closed.

4. Section 3.5.2 - In the specification, delete references to dosimeters at environmental monitoring stations. Use of the NRC-approved COMPLY code has replaced actual dosimeters. Replace the old 400-500 millirem criteria for graduated curtailment of argon-41 production. The new single standard will be 90 millirem for complete cessation of Ar-41 production. This ensures compliance with the 100 millirem standard in 10 CFR 20.1301. Current AFRRI reactor operations never approach 90 millirem/year.

5. Section 4.2.5 - Change the specification to allow inspection and measurement of half of the core instead of the entire core each year. This change is in keeping with practices at other TRIGA reactors, considering the well-documented stability and history of TRIGA fuel. The inspection requirement after 500 pulses greater than $2.00 is retained. The AFRRI reactor never approaches 500 pulses above $2.00 in any year.

6. Section 6.1.1 - In the organization chart, change "Radiation Sources Department" to "Radiation Sciences Department" to conform to current AFRRI usage. This in no way affects reactor operation or safety.

7. Section 6.1.3.2a. - Add subparagraph 5 to clarify that a single Senior Reactor Operator (SRO) may be both the SRO on call and the operator present in the control room simultaneously. This clarification has been discussed with the NRC staff during several reactor inspections.

8. Section 6.2.3.2 - Delete the reference to subcommittee meetings and clarify that at least two meetings will be held each year instead of four. There will be no more subcommittees and all meetings will be full committee meetings. This change complies with ANSI/ANS-15.1-1990.

ENCL 1

9. Section 6.2.3.3 - Revise the description of a quorum to specifically require attendance by the Reactor Facility Director and Radiation Protection Officer. The requirement for one non-AFRRI member is retained.

10. Section 6.2.3.5 - Change to say that minutes of the previous meeting "should" be available rather than "shall." Most minutes are rather short and do not require much review time. This change has no safety implications.

11. Sections 6.2.4a, b, and c. - Implement the revision of 10 CFR 50.59 effective March 13, 2001 by replacing the obsolete term "unreviewed safety question" with references to the new criteria in 10 CFR 50.59(c)(2).

12. Sections 6.5.1b, 6.5.2, and 6.6 - Clarify that reports should be submitted to the Office of Nuclear Reactor Regulation rather than to Region I now that research reactor oversight has been consolidated at NRC Headquarters.

13. Section 6.6.1b(6) - Same change as in #11 above.

14. Grammatical corrections that do not change the meaning of the requirement or section have been made throughout the document.

15. When the document was converted to Microsoft Word format, numerous lines of text moved to different pages. We are therefore reissuing the entire document as proposed Amendment 24.

TECHNICAL SPECIFICATIONS FOR THE

AFRRI REACTOR FACILITY

27 June 2001

LICENSE R-84

DOCKET 50-170

ENCL 2

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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6.0 ADMINISTRATIVE CONTROLS Page

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 CALIBRATION

A channel calibration consists of using a known signal to verify or adjust a channel to produce an output that corresponds with acceptable accuracy to known values of the parameter that the channel measures. Calibration shall encompass the entire channel including equipment activation, alarm, or trip, and shall be deemed to include a channel test.

1.3 CHANNEL CHECK

A channel check is a verification of acceptable performance by observation of channel behavior.

1.4 CHANNEL TEST

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

1.5 COLD CRITICAL

The reactor is in a cold critical condition when it is critical at a power level less than 100 watts, with the fuel and bulk water temperatures equal to or less than 40°C.

1.6 CORE GRID POSITION

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

1.7 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 nonroutine reactor parameters or characteristics.

1.8 EXPERIMENTAL FACILITIES

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

a. Exposure Room #1

b. Exposure Room #2

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

c. Reactor Pool

d. Core Experiment Tube

e. Portable Beam Tubes

f. Pneumatic Transfer System

g. Incore Locations

1.9 FUEL ELEMENT

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

1.10 INSTRUMENTED ELEMENT

An instrumented element is a special fuel element in which sheathed chromel/alumel or equivalent thermocouples are embedded in the fuel.

1.11 LIMITING SAFETY SYSTEM SETTING

Limiting safety system settings are settings for automatic protective devices related to those variables having significant safety functions.

1.12 MEASURED VALUE

A measured value is the magnitude of a variable as it appears on the output of a measuring channel.

1.13 MEASURING CHANNEL

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

1.14 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; and

c. The individual is capable of getting to the reactor facility within a reasonable time under normal circumstances.

1.15 OPERABLE

A system channel, device, or component shall be considered operable when it is capable of performing its intended function(s) in a normal manner.

1.16 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, utilizing the appropriate scrams in Table 2 and the appropriate interlocks in Table 3. The reactor may be pulsed from a critical or subcritical state.

1.17 REACTOR OPERATION

Reactor operation is any condition wherein the reactor is not shut down, core maintenance is being performed, or there is movement of any control rod.

1.18 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 that may require manual protective action to be initiated.

1.19 REACTOR SECURED

The reactor is secured when all the following conditions are satisfied:

a. The reactor is shut down.

b. The console key switch is in the "off" position, and the key is removed from the console and is under the control of a licensed operator, or is stored in a locked storage area.

c. No work is in progress involving in-core fuel handling or refueling operations, maintenance of the reactor or its control mechanisms, or insertion or withdrawal of in-core experiments, unless sufficient fuel is removed to ensure a $0.50 (or greater) shutdown margin with the most reactive control rod removed.

1.20 REACTOR SHUTDOWN

The reactor is shut down when the reactor is subcritical by at least $0.50 of reactivity.

1.21 REPORTABLE OCCURRENCE

A reportable occurrence is any of the following that occurs during reactor operation:

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.

c. Malfunction of a required reactor or experiment safety system component that could render the system incapable of performing its intended safety function unless the malfunction is discovered during tests.

d. Any unanticipated or uncontrolled positive 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. An unplanned or uncontrolled 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.

1.22 SAFETY CHANNEL

A safety channel is a measuring channel in the reactor safety system that provides a reactor protective function.

1.23 SAFETY LIMIT

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

1.24 SHUTDOWN MARGIN

Shutdown margin shall mean the minimum shutdown reactivity considered necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating conditions, and that the reactor will remain subcritical without further operator action.

1.25 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.

1.26 STEADY STATE MODE

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

1.27 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.

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 will result.

Specification

The maximum temperature in a standard TRIGA fuel element shall not exceed 1000°C under any condition 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 standard TRIGA fuel is based on data that includes the large mass of experimental evidence obtained during high-performance reactor tests on this fuel. These data indicate that the stress in the cladding due to hydrogen pressure from the dissociation of zirconium hydride will remain below the failure point, provided that the temperature of the fuel does not exceed 1000°C while immersed in water.

2.2 LIMITING SAFETY SYSTEM SETTING FOR FUEL TEMPERATURE

Applicability

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

Objective

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

Specification

There shall be two fuel temperature safety channels. The limiting safety system setting for these instrumented fuel elements' temperature shall not exceed 600°C. One channel shall utilize an instrumented element in the "B" ring, and the second channel shall utilize an instrumented 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 at least 400°C for standard TRIGA stainless-steel-clad 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 element is located in the hottest position in the core, the difference between the true and measured temperatures will be only a few degrees. If the instrumented element is located in a region of lower temperature, the measured temperature will differ by a greater amount from that actually occurring at the core hot spot. To lessen this difference, the requirement is to locate the element in the hottest region of the core. These margins are sufficient 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.

In the pulse mode of operation, the same limiting safety system setting shall apply. However, the temperature channel will have no effect on limiting the peak power generated, because of its relatively long time constant (seconds), compared with the width of the pulse (milliseconds). In this mode, however, the temperature trip will act to reduce the amount of energy generated in the entire pulse transient, by cutting the "tail" of the power transient if the pulse rod remains stuck in the fully withdrawn position with enough reactivity to exceed the temperature-limiting safety system setting.

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

To assure that the reactor safety limit (fuel temperature) is not exceeded, and to provide for a set point for the high flux limiting safety systems, so that automatic protective action will prevent the safety limit from being reached during steady state operations.

Specification

The reactor steady state power level shall not exceed 1.1 megawatts. The normal steady state demand power limit of the reactor should be 1.0 megawatt. For purposes of testing and calibration, the reactor may be operated at power levels not to exceed 1.1 megawatts during the testing period.

Basis

Thermal and hydraulic calculations and operational experience indicate that TRIGA fuel may be safely operated up to power levels of at least 1.5 megawatts with natural convective cooling.

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 assure that the fuel temperature safety limit will not be exceeded.

Specification

The maximum step insertion of reactivity shall be 2.8% $\Delta k/k$ ($4.00) in the pulse mode.

Basis

Based upon the Fuchs-Nordheim mathematical model (cited by C.E. Clifford et al. in the April 1961 GA Report # 2119, "Model of the AFRRI-TRIGA

Reactor"), an insertion of 2.8% $\Delta k/k$ results in a maximum average fuel temperature of less than 550°C, thereby staying within the limiting safety settings that protect the safety limit. The 50°C margin to the Limiting Safety System Setting and the 450°C margin to the safety limit amply allow for uncertainties due to extrapolation of measured data, accuracy of measured data, and location of instrumented fuel elements in the core.

3.1.3 REACTIVITY LIMITATIONS

Applicability

These specifications apply to the reactivity condition of the reactor and the reactivity worths 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 will not be exceeded.

Specifications

a. The reactor shall not be operated with the maximum available excess reactivity above cold critical with or without all experiments in place greater than $5.00 (3.5% $\Delta k/k$).

b. The minimum shutdown margin provided by the remaining control rods with the most reactive control rod fully withdrawn or removed shall be $0.50 (0.35% $\Delta k/k$) for any condition of operation.

Bases

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

b. The shutdown margin assures that the reactor can be shut down from any operating condition even if the highest worth control rod remains in the fully withdrawn position or is completely removed.

3.1.4 SCRAM TIME

Applicability

The specification applies to the time required to fully insert any control 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 assures 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 assure 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.

Objective

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

Specification

The reactor shall not be operated unless the measuring channels listed in Table 1 are operable.

TABLE 1. MEASURING CHANNELS

	Minimum Number Operable in Effective Mode	
	Steady State	Pulse
Fuel Temperature Safety Channel	2	2
Linear Power Channel	1	1
Log Power Channel	1	0
High-Flux Safety Channel	2	1*
Pulse Energy Integrating Channel	0	1*

(*NOTE: Same channel as linear power in this mode)

Basis

Fuel temperature displayed at the control console gives continuous information on this parameter, which has a specified safety limit. The power level channels assure 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.

3.2.2 REACTOR SAFETY SYSTEM

Applicability

This specification applies to the reactor safety system.

Objective

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.

TABLE 2. MINIMUM REACTOR SAFETY SYSTEM SCRAMS

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

Basis

The fuel temperature and power level scrams provide protection to assure that the reactor can be shut down before the safety limit on the fuel element temperature will be 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).

TABLE 3. MINIMUM REACTOR SAFETY SYSTEM INTERLOCKS

	Effective Mode	
Action Prevented	Steady State	Pulse
Pulse initiation at power levels greater than 1 kilowatt		X
Withdrawal of any control rod except transient		X
Any rod withdrawal with count rate in operational channel below 0.5 cps	X	X
Simultaneous manual withdrawal of two standard rods	X	

Basis

The interlock preventing the initiation of a pulse at a critical level above 1 kilowatt assures that the pulse magnitude will not allow the fuel element temperature to approach 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 count rate to be seen by the operational channels 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.

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.

Specification

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.

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 operators have visually inspected the room to ensure that no personnel remain in the room prior to securing the plug door.

Basis

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.

c. To help preclude 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 kilowatts when the purification system inlet water temperature exceeds $60^{\circ}C$.

b. The reactor shall not be operated if the conductivity of the water is greater than 2 micromhos/cm (or less than 0.5×10^6 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 0.2×10^6 ohms-cm resistance) averaged over 1 week.

Bases

Manufacturer's data states that the resins in the water purification system break down with sustained operation in excess of $60^{o}C$. 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 assure that the ventilation system is operable.

Specification

The reactor shall not be operated unless the facility ventilation system is operable, except for brief periods of time during which the dampers shall be closed. In the event of a significant release of airborne radioactivity in the reactor room, the ventilation system to the reactor room shall be secured via closure dampers automatically 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 clad 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 will give warning of high levels of radiation that might 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 assure safe operation of the reactor.

Specification

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

a. Area Radiation-Monitoring System. The area radiation-monitoring (ARM) 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. Gas Stack Monitor. The gas stack monitor (GSM) will sample and measure the gaseous effluent in the building exhaust system.

c. Air Particulate Monitor. The air particulate monitor (APM) will sample the air above the reactor pool. This unit will be sensitive to particulate matter from decayed fission products. Alarm of this unit will cause closure of the positive sealing dampers, causing reactor room isolation.

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

TABLE 4. LOCATIONS OF RADIATION MONITORING SYSTEMS

Monitor	Location of Alarm (A = Audible; V = Visual)	Readout Location
ARM		
R1, Reactor Room	Control Room A&V	Control Room
R2, Reactor Room	Control Room V	Control Room
E3, Exp. Room 1 Area	Control Room V	Control Room
E6, Exp. Room 2 Area	Control Room V	Control Room
GSM - Reactor exhaust	Control Room V	Control Room
APM - Reactor room	Control Room A&V	Control Room

Basis

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 provides reactor room isolation from the outside environment, in the event of airborne radioactivity within the reactor room from fission products decay.

3.5.2 EFFLUENTS: ARGON-41 DISCHARGE LIMIT

Applicability

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

Objective

To ensure that 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 indicate that an exposure of 90 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 assure the 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 1.3 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 the design pressure of the container.

d. Samples shall be doubly contained when release of the contained material could cause corrosion of the experimental facility.

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.

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.

    (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 determine the consequences and 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 Senior Reactor Operator for mechanical stability, to ensure that unintended movement will not cause an unplanned reactivity change or physical damage. All operations in any experimental area 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 1.3 curie limitation on iodine-131 through -135 assures that, in the event of malfunction of a fueled experiment leading to total release of the iodine, the particulate iodine trapped by the absolute filtering system will present a minimal hazard to staff personnel should a release occur.

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 cold critical reactor 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.

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

g. To assure that operation of the reactor with damaged reactor fuel or structure is prevented, the release of fission products to the environment is limited.

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.

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-stated systems including the ventilation system, the core and its associated support structure, the pool, coolant system, the rod drive mechanism, or the 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 Facility Safety Committee. 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

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 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 or staff radiation exposures. Similarly, overall reactor scheduling achieves significant reductions in staff exposures. Consequently, ALARA must be a part of both the overall reactor scheduling and the detailed experiment planning.

4.0 SURVEILLANCE REQUIREMENTS

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 control rod and the shutdown margin shall be determined annually but at intervals not to exceed 15 months.

b. The reactivity worth of an experiment shall be estimated or measured as appropriate, before reactor power operation with an experiment, the first time it is performed.

c. The control rods shall be visually inspected for 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 (pulse) rod system shall be performed. Semiannually, at intervals not to exceed 7.5 months, the transient (pulse) 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.

f. The power coefficient of reactivity at 100 kilowatts and 1 megawatt will be measured annually, at intervals not to exceed 15 months.

Bases

The reactivity worth of the control rods is measured to assure that the required shutdown margin is available and to provide an accurate means for determining the reactivity worths 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 control rods is made to evaluate corrosion and wear characteristics caused by operation in the reactor. Functional checks along with periodic maintenance assure repeatable performance.

Excess reactivity measurements assure that core configuration is the same, 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 shall be measured semiannually, but at intervals not to exceed 7.5 months.

Basis

Measurement of the scram time on a semiannual basis 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 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.

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 assure that the safety system channel scrams are operable on a daily basis or prior to an extended run. The power level channel calibration will assure that the reactor is to be operated at the authorized power levels.

4.2.3 FUEL TEMPERATURE

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

b. A calibration of the fuel temperature-measuring channel 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.

Basis

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

Specification

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

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

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

c. 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.

Basis

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 integrity of the fuel element cladding.

Specification

Half of the fuel elements present in the reactor core, to include all the fuel follower control rods, shall be inspected for damage or deterioration, and measured for length and bow annually (not to exceed 15 months). During a two year cycle, all fuel elements except those in long-term storage shall be measured and inspected. Fuel elements and fuel follower control rods shall also be inspected and measured if they have been in the core for 500 pulses of insertion greater than $2.00 since their last biannual inspection and measurement. Fuel elements in long-term storage need not be measured or inspected until returned to the core; however fuel elements routinely moved to temporary storage shall be measured and inspected every 500 pulses of insertion greater than $2.00 or annually (not to exceed 15 months), whichever occurs first.

Basis

The frequency of inspection and measurement is based on the parameters most likely to affect the fuel cladding of a pulse reactor, and the utilization of fuel elements whose characteristics are well known.

The limit of transverse bend has been shown to result in no difficulty in disassembling the core. Analysis of a worst case scenario in which two adjacent fuel elements suffer sufficiently severe transverse bends to result in the touching of the fuel elements has shown that no damage to the fuel elements will result via a hot spot or any other known mechanism.

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 assure 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.

Basis

Based on experience, observation at these intervals provides acceptable surveillance of limits that assure that fuel clad corrosion and neutron activation of dissolved materials will not occur.

4.4 VENTILATION SYSTEM

Applicability

This specification applies to the facility ventilation system isolation.

Objective

The objective is to assure 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 assure 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 area radiation-monitoring equipment and the air particulate monitoring system.

Objective

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

Specification

The area radiation-monitoring system and the air particulate monitoring system shall be channel tested quarterly, but at intervals not to exceed 4 months. They shall be verified to be operable by a channel check daily when the reactor is in operation, and shall be calibrated annually, not to exceed 15 months.

Basis

Experience has shown that quarterly verification of area radiation-monitoring and air 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.

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.

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.

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) assure 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) assure that the fuel elements used in the core are substantially those analyzed in the Safety Analysis Report.

Specification

The individual nonirradiated standard 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.

Basis

A maximum uranium content of 9 weight percent in a standard 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%. An increase in local power density of 6% in an individual fuel element reduces the safety margin by 10%, at most. The hydrogen-to-zirconium ratio of 1.7 will produce a maximum pressure within the cladding 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 standard TRIGA fuel element in the D-ring. The volume of fuel in a fuel follower rod is 56% of the volume of a standard TRIGA fuel element. Therefore, the actual power produced in the fuel follower rod is 33% less than the power produced in a standard 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 standard 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

Standard TRIGA cores have been in use for years, 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) assure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment, and (b) assure 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 assure 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 assure that stored fuel will not become critical and will 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.

Basis

The limits imposed by this specification are conservative and assure safe storage and handling. Experience shows that approximately 67 fuel elements are required, of the design used at AFRRI, in a closely packed array 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. Organization changes may occur, based on Institute requirements, and they will be depicted on internal documents. However, no changes may be made in the Operational, Safety, and Emergency Control Chain in which the Reactor Facility Director has direct responsibility to the Director, AFRRI.

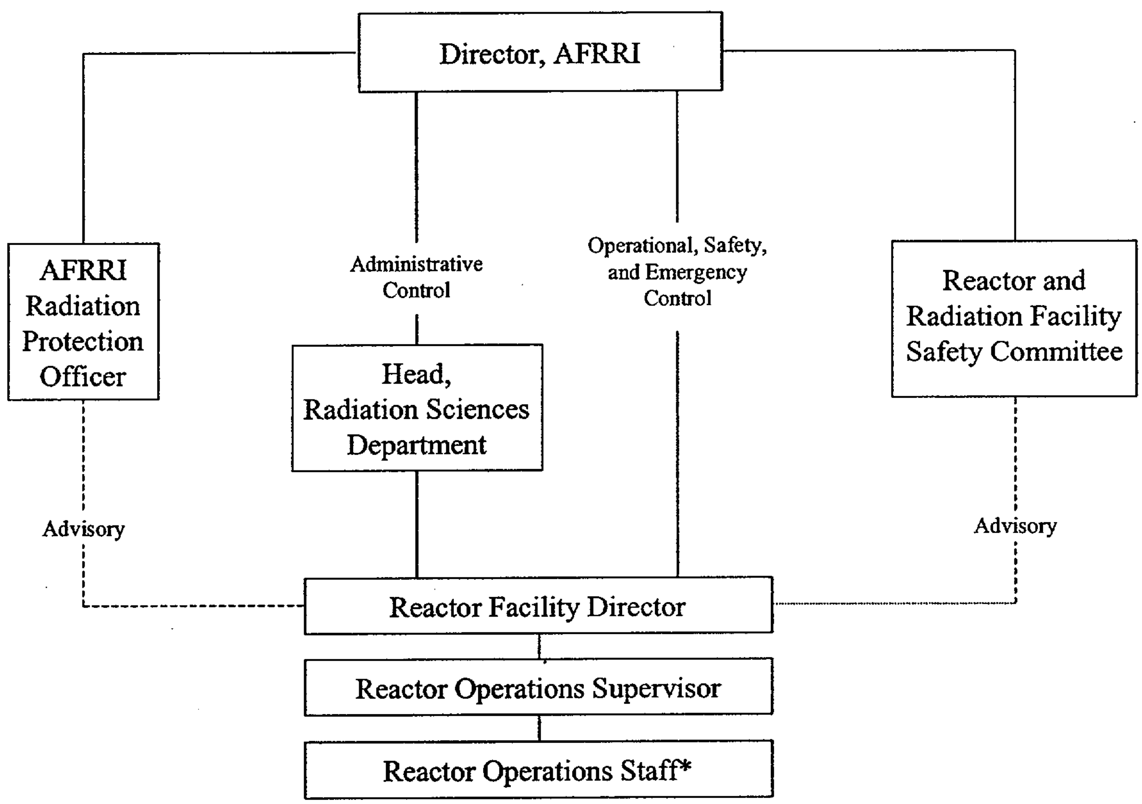

Figure 1. Organization of Personnel for Management and Operation of the AFRRI Reactor Facility

* Any reactor staff member has access to the Director for matters of safety

6.1.2 RESPONSIBILITY

The Director, AFRRI, 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 in the RFD's 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.

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 6 or more years of nuclear experience. Higher education in a scientific or nuclear engineering field may fulfill up to 4 years of experience on a one-for-one basis. The Facility Director must have held a USNRC Senior Reactor Operator license on the AFRRI reactor for at least 1 year before appointment to this position.

b. Reactor Operations Supervisor (ROS)

At the time of appointment to this position, the ROS shall have 3 years nuclear experience. Higher education in a science or nuclear engineering field may fulfill up to 2 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 1 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 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. Maintenance activities that could affect the reactivity of the reactor shall be accomplished under the supervision of an SRO.

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 Director)

(2) Radiation safety personnel (AFRRI Radiation Protection Officer)

(3) Other operations personnel (Reactor Staff, ROS)

6.1.4 TRAINING OF PERSONNEL

A training and retraining program will 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 FACILITY SAFETY COMMITTEE (RRFSC)

6.2.1 COMPOSITION AND QUALIFICATIONS

6.2.1.1 Composition

a. Regular RRFSC Members (Permanent Members)

(1) The following shall be members of the RRFSC:

(a) AFRRI Radiation Protection Officer

(b) AFRRI Reactor Facility Director

(2) The following shall be appointed to the RRFSC by the Director, AFRRI:

(a) Chairman as appointed by the AFRRI Director

(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 RRFSC Members (Temporary Members)

(1) Other knowledgeable persons to serve as alternates in item a(2)(b) above as appointed by the AFRRI Director.

(2) Voting ad hoc members, invited by the Director of AFRRI, to assist in review of a particular problem.

c. Nonvoting members as invited by the Chairman, RRFSC.

6.2.1.2 Qualifications

The minimum qualifications for a person on the RRFSC shall be 6 years of professional experience in the discipline or specific field represented. A baccalaureate degree may fulfill 4 years of experience.

6.2.2 FUNCTION AND AUTHORITY

6.2.2.1 Function

The Reactor and Radiation Facility Safety Committee is directly responsible to the Director, AFRRI. 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 RRFSC shall report to the Director, AFRRI, 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 RRFSC Chairman to serve on a temporary basis. No more than two alternates shall participate on a voting basis in RRFSC activities at any one time.

6.2.3.2 Meeting Frequency

The RRFSC shall meet at least two times during a calendar year.

6.2.3.3 Quorum

A quorum of the RRFSC for review shall consist of the Chairman, the Reactor Facility Director, the Radiation Protection Officer (or designated alternates), and one non-AFRRI member (or alternate member). A majority of those present shall be regular members.

6.2.3.4 Voting Rules

Each regular RRFSC 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.

6.2.3.5 Minutes

Minutes of the previous meeting should be available to regular members at least 1 week before a regular scheduled meeting.

6.2.4 REVIEW FUNCTION

The RRFSC 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.

h. 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 RRFSC, 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 Director. Audits may be performed by one individual who need not be an RRFSC 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 RRFSC or the Director of AFRRI.

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 Facility Safety Committee (RRFSC). The procedures shall be adequate to assure 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, 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 Security Plan and Emergency Plan.

f. Reactor core loading and unloading.

g. Checkout startup, 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. Safety and Health Department

c. Reactor and Radiation Facility Safety Committee (RRFSC)

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 experiments safely performed at least once. These experiments may be performed at the discretion of the Reactor Facility Director and coordinated with the Safety and Health Department when appropriate. These authorizations do not require additional RRFSC 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 may 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 RRFSC 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 Office of Nuclear Reactor Regulation, the Director of AFRRI, and the RRFSC not later than the next working day.

c. A Safety Limit Violation Report shall be prepared. This report shall be reviewed by the RRFSC, 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 Director of AFRRI, and the RRFSC within 14 days of the violation.

6.5.2 REPORTABLE OCCURRENCES

Reportable occurrences as defined in 1.21 (including causes, actual or probable consequences, corrective actions, and measures to prevent recurrence) shall be reported to the NRC. Supplemental reports may be required to fully describe the final resolution of the occurrence.

Prompt Notification With Written Follow-up. The types of events listed below shall be reported as soon as possible by telephone and confirmed by telegraph, mailgram, or similar transmission to the NRC Office of Nuclear Reactor Regulation no later than the first workday following the event, with a written followup report 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.

(1) 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.

(2) 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.

(3) 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.

6.6 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) 9 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 3 months until all three events have been completed.

b. Annual Operating Report: Routine operating reports covering the operation of the unit during previous calendar year shall be submitted prior to 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:

   (l) 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, including the reasons and the corrective actions taken, if applicable.

(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 are 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) Liquid Waste (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 3×10^{-6} microcuries/ml for fission and activation products

Total radioactivity (in curies), released by nuclide during the reporting period, 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 diluent) 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 8 days

(c) Solid Waste (summarized on a quarterly basis)

Total cubic feet of 3 to 83 material in solid form disposed of under 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.

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 RRFSC 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 RRFSC

6.7.2 <u>RECORDS TO BE RETAINED FOR AT LEAST ONE COMPLETE TRAINING CYCLE</u>

a. Training exams

b. Requalification records

6.7.3 <u>RECORDS TO BE RETAINED FOR THE LIFE OF THE FACILITY</u>

a. Gaseous and liquid radioactive effluents released to the environs

b. Appropriate offsite environmental monitoring surveys

c. Radiation exposures for all personnel

d. Updated as-built drawings of the facility