ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE
8901 WISCONSIN AVENUE
BETHESDA, MARYLAND 20889-5603

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U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Sir or Madam:

Enclosed is a revision of the Technical Specifications for the Armed Forces Radiobiology Research Institute reactor (license R-84, docket 50-170). The Technical Specifications submitted on March 30, 2015 should be withdrawn and replaced by this February 26, 2016 version.

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

STEPHEN I. MILLER
Reactor Facility Director

A020
NRR

TECHNICAL SPECIFICATIONS FOR THE AFRRI REACTOR FACILITY

26 February 2016

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, line, amplifier, 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 plate. It is specified by a letter indicating the specific ring in the grid plate and a number indicating a particular position within that ring.

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

Any operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate nonroutine reactor characteristics or that is intended for irradiation within an experimental facility. Hardware rigidly secured to a core of shield structure so as to be a part of its design to carry out experiments is not normally considered an experiment.

1.10. EXPERIMENTAL FACILITIES

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

a. Exposure Room #1

b. Exposure Room #2

c. Reactor Pool

d. Core Experiment Tube (CET)

e. Portable Beam Tubes

f. Pneumatic Transfer System

g. In-core Locations

1.11. FUEL ELEMENT

A fuel element is a single TRIGA fuel rod or the fuel portion of a fuel follower control rod (FFCR).

1.12. INSTRUMENTED FUEL ELEMENT

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

1.13. MEASURED VALUE

The measured value is the value of a parameter as it appears on the output of a 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 their whereabouts and telephone number;

c. The individual remains at a reachable location 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

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

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 shutdown.

1.21. REACTOR OPERATOR

A reactor operator is 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 channels, 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; or,

b. All of 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;

   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 shutdown 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 reference core condition is when the core it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<$0.01).

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 that are normal to the operating environment, or by forces which can arise as a result of credible malfunctions.

1.29. SENIOR REACTOR OPERATOR

A senior reactor operator is 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 at power levels not exceeding 1.1 MW.

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 because 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 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 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 Safety Analysis Report 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 Safety Analysis Report, indicates 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 pulse mode.

Basis

Based upon calculations detailed in the AFRRI Safety Analysis Report, 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 should all control rods be 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 remains in its most reactive position.

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 shall 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.

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

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.

Bases

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 because 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. 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 in a TRIGA reactor the specified scram time is adequate to ensure the safety 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. 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 shall 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	3	3
(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.

Bases

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 pulse 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 control rod withdrawal through the interlock system. The pool water level ensures that a loss of biological shielding would result in a reactor scram. The watchdog scram ensures 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 below 0.5 cps as measured by the operational channel	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	

Bases

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 increase in steady state reactor power prior to initiation of a pulse. Requiring a minimum count rate to be measured by the operational channel 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 level 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.

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 the 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 core outlet water temperature exceeds 60°C;

b. The reactor shall not be operated if the conductivity of the bulk water is greater than 5 micromhos/cm; and

c. 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 data states that the resins in the water purification system break down with sustained operation in excess of 60°C. Based on experience, activation of impurities in the bulk water at power levels below 5 kW does not pose a significant radiological hazard. The conductivity limits are established to provide acceptable control of corrosion and are consistent with the fuel vendor recommendation and experience at similar reactors. The water level monitoring system provides prompt notification of a potential loss of primary coolant.

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 continuous hours to permit repair, maintenance, or testing. In the event of a release of airborne radioactivity in the reactor room above 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 continuous 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 Part 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. 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 if high levels of radiation are detected 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 radiation monitoring system which shall be available during reactor operations.

Objective

The objective is to ensure that adequate radiation monitoring channels shall be 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 to detect streaming radiation;

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

c. Continuous Air Particulate Monitor: The continuous air particulate monitor (CAM) shall sample the air above the reactor pool. This unit shall be sensitive to radioactive particulate matter. Alarm of this unit shall initiate closure of the ventilation system 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
Exp. Room 1 Area	Prep Area and Control Rooms
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 abnormal operations or conditions. The radiation monitoring system provides information to the operating personnel of any existing or impending danger from radiation. 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 AFRRI TRIGA reactor facility.

Objective

The objective is to ensure that the radiation dose to members of the public due to the discharge of argon-41 from the AFRRI TRIGA reactor facility shall be below the value specified in 10 CFR Part 20.

Specifications

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

b. If calculations, which shall be performed at least quarterly but not to exceed 20 MWh of operation, indicate that argon-41 release in excess of 313.5 curies to the unrestricted environment could be 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

As described in the AFRRI Safety Analysis Report, COMPLY analysis indicates that the release of 313.5 curies from the stack to the unrestricted environment in one calendar year yields a dose to the maximally exposed member of the public of 9.9 mrem. Therefore, limiting argon-41 release to less than 313.5 curies ensures that 10 CFR Part 20 limits on doses to the public are not exceeded. The upper limit of 20 MWh of reactor operation between gaseous effluent analyses ensures it is not possible to exceed 15% of the 10 mrem limit between reports.

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 such that airborne concentrations of activity averaged over a year do not exceed 10 CFR Part 20, Appendix B.

Specifications

The following limitations shall apply to the irradiation of experiments:

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 Part 20, assuming that 100% of the gases or aerosols escape;

b. Each fueled experiment shall be limited such 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 5.0 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 worth 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 worth of multiple moveable or unsecured experiments 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 where vapor formed by boiling the material can escape only through an undisturbed column of water above the core, up to 10% of the vapor can escape;

g. If a capsule fails and releases materials that could damage the reactor fuel or structure by corrosion or other means, physical inspection of the reactor fuel and structure 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; and

h. Experiments shall be designed such that failure of one experiment shall not contribute to the failure of any other experiment. All operations in an experimental facility 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 Part 20 will not be released to the atmosphere outside the facility boundary.

b. The 1.0 curie limitation on iodine isotopes 131 through 135 and 5.0 millicurie limitation on strontium-90 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 Part 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 such 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 $3.00 limit is less than the 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 Part 20, Appendix B, are not exceeded in the event of an experiment malfunction.

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

h. All experiments 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. 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 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.

e. The burnup of uranium-235 in the UZrH fuel matrix shall not exceed 50 percent of the initial concentration.

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 reactor core parameters.

Objective

The objective is to verify that the reactor does not exceed the authorized limits for power, shutdown margin, core excess reactivity, and verification of the total reactivity worth of each control rod.

Specifications

a. The reactivity worth of each standard control rod/transient rod and the shutdown margin shall be determined annually, 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 the 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 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. At a minimum, the core excess reactivity shall be measured annually, not to exceed 15 months.

d. The power coefficient of reactivity at 100 kW and 1 MW shall be measured annually, 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.

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

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

b. The control rod drop times of all rods shall be measured semiannually, 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.

c. 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, 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.

Bases

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. Measurement of the scram time on a semiannual basis or after mechanical maintenance is a verification of the scram system and provides 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 measurement, test, and calibration 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 test of the scram function of the high-flux safety channels shall be made 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, not to exceed 15 months.

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

Bases

TRIGA system components have operational proven reliability. Daily tests ensure accurate scram functions and ensure the detection of channel drift or other possible deterioration of operating characteristics. The channel checks 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 operated within 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

The objective is to ensure operability of the fuel temperature measuring channels.

Specifications

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

b. A channel calibration of the fuel temperature measuring channels shall be made annually, 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 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

The objective is to ensure performance and operability of the facility interlock system.

Specifications

Functional checks shall be made annually, 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 region 2 with the lead shield doors closed.

c. The warning horn shall sound in the exposure room before opening the lead shield doors unless cleared by two licensed reactor operators.

Bases

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

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, minimizing 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 bulk water shall be measured monthly, not to exceed 6 weeks.

c. The reactor coolant shall be measured for radioactivity quarterly, not to exceed 4 months.

d. The audible and visual reactor pool level alarms shall be tested quarterly, not to exceed 4 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 monitored 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 ventilation system dampers in the reactor room shall be verified to be operable and visually inspected monthly, not to exceed 6 weeks.

Basis

Experience accumulated over years of operation has demonstrated that the tests of the ventilation system dampers 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 AND EFFLUENTS

4.5.1. MONITORING SYSTEM

Applicability

This specification applies to surveillance requirements for the radiation 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, continuous air particulate monitoring, and stack gas monitoring systems shall be channel tested quarterly, not to exceed

4 months. A channel check of these systems shall be performed daily to verify operability when operations are planned. These systems shall be calibrated annually, not to exceed 15 months.

Basis

Experience has shown that quarterly verification of radiation area monitoring, continuous air particulate monitoring, and stack gas monitoring systems 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.

4.5.2. 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 the release of radioactive material to the environment.

Specifications

a. The unrestricted area outside of AFRRI shall be monitored by dosimeters that shall be analyzed quarterly, not to exceed 4 months.

b. Samples of soil, vegetation, and water in the vicinity of the reactor shall be collected and tested for radioactivity quarterly, not to exceed 4 months.

c. A gaseous effluent release report shall be generated quarterly or every 20 MW hours of reactor operations (whichever comes first) to ensure radioactive effluent will not exceed the annual limit.

Basis

Experience has shown that quarterly environmental monitoring is sufficient to detect and quantify any release of radioactive material from research reactors. The requirement for gaseous effluent release reports will ensure that Ar-41 production from normal reactor operations does not exceed 10CFR20 annual dose limits to the public.

4.6. 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), or at intervals not to exceed 500 pulses of insertion greater than $2.00, whichever comes first, 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.

5.0. DESIGN FEATURES

5.1. SITE AND FACILITY DESCRIPTION

Applicability

This specification applies to the reactor building.

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 dampers which automatically close off ventilation to the reactor room upon a signal from the reactor room continuous air particulate monitor.

d. The reactor room shall be designed to restrict air leakage when the ventilation system 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 Physical Security Plan for the AFRRI TRIGA Reactor Facility.

Bases

The facility is designed so that the ventilation will normally maintain a negative pressure with respect to the atmosphere, thus preventing 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 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 comparable to 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 shall 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 inches 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 9.0 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 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 with 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 replace B ring, C ring, and/or D ring fuel element positions within the reactor core.

Bases

TRIGA cores have been in use for decades and their safe operational characteristics are well documented. Analysis has shown that natural convection water flow provides sufficient cooling to ensure that the fuel temperature safety limit is not exceeded during reactor operations in accordance with the Technical Specifications. Placement of in-core experiments in the B ring, C ring, and/or D ring is restricted to ensure safe power peaking in adjacent fuel element positions.

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, 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 Technical Specification 5.2.1.

b. The transient control rod shall have scram capability and 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 and the fuel element or fueled device temperature will not exceed design values. Storage shall be such that groups of stored fuel elements remain subcritical under all conditions of moderation and reflection in a configuration where k_{eff} is not 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.

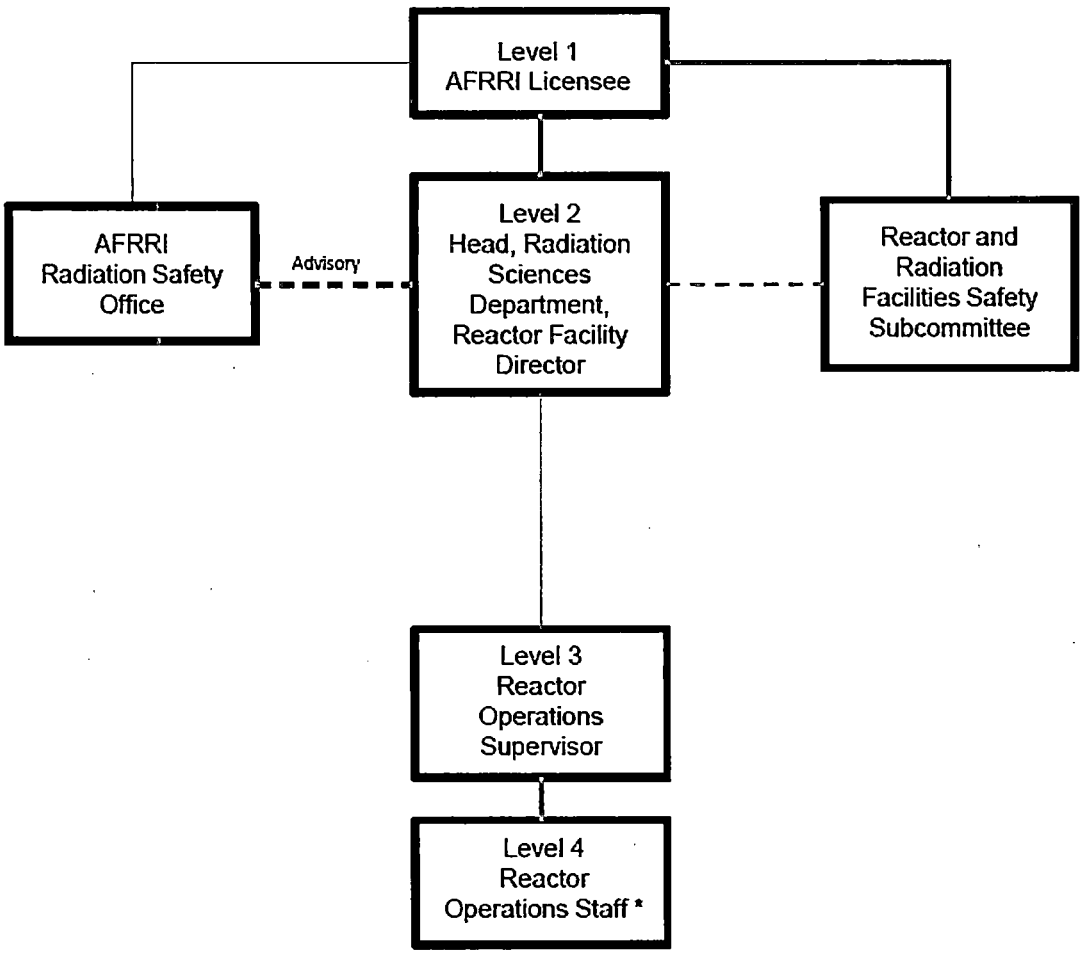
6.0. ADMINISTRATIVE CONTROLS

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 AFRRI 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



*Any reactor staff member has direct access to the AFRRI Licensee for matters concerning safety

6.1.2. RESPONSIBILITY

The AFRRI Licensee shall have license responsibility for the reactor facility. The Reactor Facility Director shall be responsible for administration and operation of the reactor facility and for determination of applicability of procedures, experiment authorizations, maintenance, and operations. The Reactor Facility Director may designate an individual who meets the requirements of Technical Specifications 6.1.3.1.a to discharge their responsibilities during an extended absence. During brief absences (periods less than 4 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 Part 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 Reactor 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

At the time of appointment to this position, the Reactor Operations Supervisor 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 Reactor Operations Supervisor shall hold a USNRC Senior Reactor Operator license on the AFRRI reactor. In addition, the Reactor Operations Supervisor 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 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 or Senior Reactor Operator 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 may fill both the on call and control room positions simultaneously. In that case, the minimum staff is three persons.

b. A Senior Reactor Operator shall be present at the reactor during the following operations:

1. All fuel or control rod relocations within the reactor core region (control rod movement associated with routine reactor operation is not considered to be a relocation);

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, Reactor Operations Supervisor)

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

b. AFRRI Reactor Facility Director

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 RRFSS 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 (or designated alternate), the Reactor Facility Director (or designated alternate), the Radiation Safety Officer (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 Radiation Safety Officer and the reactor facility director.

6.2.3.5. Minutes

a. Draft minutes of the previous meeting should be available to regular members at least one week before a regular scheduled meeting.
b. Once approved by the committee, final minuites will be submitted to level one management for review.

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 USNRC; and

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 RRFSS, but in no case by the personnel responsible for the item audited. The audits shall be performed annually, not to exceed 15 months. A report of the findings and recommendations resulting from the audit shall be submitted to the AFRRI Licensee. 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 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. Any other area of facility operations considered appropriate by the RRFSS or the AFRRI Licensee; and

f. 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 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. Operating procedures shall be used for the following items:

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 Part 20;

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

f. Fuel loading, unloading, and movement within the reactor core; 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 Reactor Operations Supervisor. 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 Department

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 Department, 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 (> $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 to ensure that the change does not impact compliance with TS 3.6, LIMITATIONS ON EXPERIMENTS. Minor changes that do not significantly alter the experiment (<$0.25) may be approved by the Reactor Operations Supervisor. 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 USNRC.

b. The safety limit violation shall be reported to the USNRC, 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 USNRC, 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 USNRC no later than the following working day after confirmation of the event, with a written follow-up report within 14 days. 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, such 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.

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 10 CFR Part 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. ACTIONS TO BE TAKEN IN CASE OF REPORTABLE OCCURRENCES

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 Reactor Facility Director or designated alternate and to the USNRC.

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 USNRC 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 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 $\times 10^{-6}$ microcuries/ml for fission and activation products;

Total radioactivity of each nuclide released during the reporting period and, based on representative isotopic analysis, average concentration at point of release during the reporting period;

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

b. Gaseous Waste (summarized on a quarterly basis)

Radioactivity discharged during the reporting period 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 10 CFR Part 20 value 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

a. Normal reactor operations;

b. Principal maintenance operations;

c. Reportable occurrences;

d. Surveillance activities required by Technical Specifications;

e. Reactor facility radiation and contamination surveys;

f. Experiments performed with the reactor;

g. Changes to operating procedures;

h. Fuel inventories and fuel transfers;

i. Records of training and qualification for members of the facility staff;

j. 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; and

k. Records of meetings of the RRFSS.

6.7.2. RECORDS TO BE RETAINED FOR AT LEAST ONE CERTIFICATION CYCLE

Records of retraining and requalification of certified reactor operators and senior reactor operators shall be retained at all times the individual is employed or until the certification is renewed.

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 monitored; and

d. Drawings of the reactor facility.