ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE

DOCKET NO. 50-170

RENEWED FACILITY OPERATING LICENSE

License No. R-84

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

   A. The application for renewal of Facility Operating License No. R-84 filed by the Armed Forces Radiobiology Research Institute ("the licensee"), dated June 24, 2004, as supplemented by letters dated March 4, August 13, September 27, October 21, and December 15, 2010; February 7, June 20, September 6, October 20, and November 28, 2011; January 17, April 20, and September 21, 2012; June 28, and August 27, 2013; December 4, 2014; March 30, 2015; and February 9, February 26, August 5, September 12, September 21, September 26, September 27, September 30, and November 16, 2016 ("the application"), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended ("the Act"), and the Commission's rules and regulations set forth in Title 10 of the *Code of Federal Regulations* (10 CFR), Chapter I;

   B. Construction of the Armed Forces Radiobiology Research Institute (AFRRI) TRIGA-type nuclear reactor was completed in substantial conformity with Construction Permit No. CPRR-61, originally issued to National Naval Medical Center on November 8, 1960, and later transferred to AFRRI, and the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, as supplemented, the provisions of the Act, and the rules and regulations of the Commission;

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

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

F. The licensee is a Federal agency and is therefore exempt from the financial protection requirement of subsection 170a of the Act. AFRRI has executed an indemnity agreement pursuant to 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements;"

G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public. The issuance of this license is in accordance with 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," of the Commission's regulations and all applicable requirements have been satisfied; and

H. The receipt, possession and use of byproduct and special nuclear materials as authorized by this facility operating license will be in accordance with the Commission's regulations in 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

2. Accordingly, Facility Operating License No. R-84 is hereby renewed in its entirety to read as follows:

A. This license applies to the Armed Forces Radiobiology Research Institute TRIGA-Mark F tank-type nuclear reactor facility (herein "the facility") owned by the Armed Forces Radiobiology Research Institute (herein "the licensee"). The facility is located on the grounds of Naval Support Activity Bethesda military installation, in Bethesda, Montgomery County, MD, and described in the licensee's application for license renewal, dated June 24, 2004, as supplemented.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the Armed Forces Radiobiology Research Institute as follows:

1. Pursuant to Subsection 104c of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility as a utilization facility at the designated location in accordance with the procedures and limitations described in the application and set forth in this license.

2. Pursuant to the Act and 10 CFR Part 70, the following activities are included:

a. to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 5.0 kilograms of contained uranium-235 enriched to less than 20 percent in the form of TRIGA-type reactor fuel;

b. to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 100 grams total of special nuclear material, of any enrichment, in the form of detectors, fission plates, foils, and solutions; and

c. to receive, possess, and use, but not separate, in connection with the operation of the facility, such special nuclear material as may be produced by the operation of the facility.

3. Pursuant to the Act and 10 CFR Part 30, the following activities are included:

a. to receive, possess, and use, in connection with the operation of the facility, a sealed 3-curie americium-beryllium neutron source; and,

b. to receive, possess, and use, in connection with operation of the facility, such byproduct material as may be produced by operation of the reactor, which cannot be separated except for byproduct material produced in non-fueled reactor experiments.

4. Pursuant to the Act and 10 CFR Part 40, “Domestic Licensing of Source Material,” to receive, possess, and use, in connection with the operation of the facility, not more than 5.0 kilograms of source material.

C. This license shall be deemed to contain, and is subject to the conditions specified 10 CFR Parts 20, 30, 40, 50, 51, 55, 70, and 73 of the Commission’s regulations; is subject to all provisions of the Act, and to the rules, regulations and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below:

1. <u>Maximum Power Level</u>

The licensee is authorized to operate the reactor at a steady-state power level up to a maximum of 1.1 megawatts (thermal) and to pulse the reactor in accordance with the limitations in the Technical Specifications.

2. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised by Amendment No. 26 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Physical Security Plan

The licensee shall maintain and fully implement all provisions of the Commission-approved physical security plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved physical security plan, entitled “Armed Forces Radiobiology Research Institute, Reactor Facility, Physical Security Plan, Facility Operating License R-84, Docket Number 50-170, June 2015,” consists of documents withheld from public disclosure pursuant to 10 CFR 73.21, “Protection of Safeguards Information: Performance Requirements.”

This license is effective as of the date of issuance and shall expire at midnight, 20 years from the date of issuance.

For the Nuclear Regulatory Commission

/RA/

William M. Dean, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendix A, Technical Specifications

Date of Issuance: November 30, 2016

TECHNICAL SPECIFICATIONS FOR THE AFRRI REACTOR FACILITY

30 September 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
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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 CONFINEMENT

Confinement is an enclosure of the overall facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways.

1.7 CONTROL ROD

A control rod is a device fabricated from neutron absorbing material or fuel, or both, that is used to establish neutron flux changes and to compensate for routine reactivity losses. Scrammable control rods can be quickly uncoupled from their drive units to rapidly shutdown the reactor if needed.

1.8. CORE CONFIGURATION

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

1.9. 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.

1.10. EMERGENCY STOP

Emergency Stop is a scram designed to prevent or cease reactor operations. Emergency stop buttons are provided in Exposure Room 1, Exposure Room 2 and on the console

1.11. 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.

1.12. 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 the core or shield structure so as to be a part of its design to carry out experiments is not normally considered an experiment.

1.13. 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

1.14. FUEL ELEMENT

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

1.15 HIGH FLUX SAFETY CHANNEL

A high flux safety channel is a power measuring safety channel in the reactor safety system, NP and NPP.

1.16. INITIAL STARTUP AND APPROACH TO POWER

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1.17. 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.

1.18. LONG-TERM STORAGE

Long-term storage of fuel applies to fuel that has been taken out of service with no plans for use for more than one fuel measurement cycle.

1.19. MEASURED VALUE

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

1.20. MOVABLE EXPERIMENT

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

1.21. 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.

1.22. OPERABLE

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

1.23. OPERATIONAL CHANNEL

Operational Channel: The Operational Channel is a power measuring channel used during steady state and square wave operations

1.24. OPERATING

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

1.25. POWER LEVEL MONITORING CHANNEL

A power level monitoring channel is defined to be a channel that is intended to provide real time power level readings to the operator.

1.26. PROTECTIVE ACTION

Protective action is the initiation of a signal or the operation of equipment within the reactor safety system in response to a parameter or condition of the reactor facility having reached a specified set point.

1.27. 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.

1.28. 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.

1.29. REACTOR OPERATING

The reactor is operating whenever it is not secured or shutdown.

1.30. REACTOR OPERATOR

A reactor operator is an individual who is licensed to manipulate the controls of a reactor.

1.31. 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.

1.32. 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 fuel movement, 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.

1.33. 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.

1.34. REFERENCE CORE CONDITION

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

1.35. SAFETY CHANNEL

A safety channel is a high flux safety channel with scram capability.

1.36. SCRAM TIME

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

1.37. 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.

1.38. 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.

1.39. 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.

1.40. 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.

1.41. 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.

1.42. 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. Square wave mode is a subset of the steady state mode of operation.

1.43. 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

1.44. 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.

1.45. TRUE VALUE

The true value is the actual value of a parameter.

1.46. UNSCHEDULED SHUTDOWN

An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety systems, 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.

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 600ºC, 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 (B-1), a temperature indication of 600ºC would yield a peak temperature at the highest power density location (B-4) of 611ºC. 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 (C-12), a temperature indication of 600ºC would yield a peak temperature at the highest power density location (C-9) of 628ºC.

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 in their most reactive positions.

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 are operable except:
   a. the reactor may be operated at a power level no greater than 250kw with no more than one control rod drive inoperable with the associated control rod drive 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

(1) Any Linear Power, Log Power, High-Flux Safety or Fuel Temperature Safety Channels 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 the control rods 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.

At 25% power, the peak fuel temperature has been measured to be approximately 200 degrees C, 50% of the maximum expected fuel temperature at full power. Even if 30% of the core were to be effectively disabled by the inoperable control rod, the remaining elements would stay well below the established safety margins. This provides sufficient safety margin to ensure safe operations.

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. For footnote (2), events which lead to these circumstances are self-revealing to the operator.

3.2.2. REACTOR SAFETY SYSTEMS

Applicability

This specification applies to the reactor safety systems.

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
Watchdogs (UIT and CCS)	15 seconds	2	2
AC Power Loss	15 seconds	1	1

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 return to a low level after pulsing. The emergency stop allows personnel trapped in a potentially hazardous exposure room, or the reactor operator, to scram the reactor through the facility 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 User Interface Terminal (UIT) and the Console Computer System (CCS). The AC power loss scram ensures that a loss of AC power to the uninterruptible power supply (UPS) for the reactor control console will result in a scram.

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 power level below 1 x 10^{-5} watts as measured by the Linear Power Channel (NMP-1100)	X	X
Simultaneous manual withdrawal of two standard rods	X	
Any rod withdrawal if high voltage is lost to the Log Power Channel (NLW-1000)	X	X
Withdrawal of any control rod if reactor period is less than 3 seconds	X	
Application of air if the transient rod drive is not fully down. This interlock is not required in square wave mode.	X	

* Reactor safety system interlocks shall be tested daily whenever operations involving these functions are planned

Bases

The interlock preventing the initiation of a pulse at a power 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 power level to be measured by the Linear Power 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. Correct high voltage to the Log Power Channel ensures accurate power indications. 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 contact between the core and the lead shield doors.

Specifications

Facility interlocks shall be provided so that:

a. The reactor cannot be operated unless the lead 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 or facility damage 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 protect the integrity of the reactor core

Specifications

a. The reactor shall not be operated if the bulk water temperature exceeds 60°C;

b. The reactor shall not be operated if periodic measurements taken IAW TS 4.3 show conductivity of the bulk water 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.

d. The reactor shall not be operated if the measurement required by TS 4.3 shows concentrations of radionuclides above the values in 10CFR part 20 appendix B table 2 are found in the primary coolant until the source of the activity is determined and appropriate corrective actions are taken.

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

a. 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. The ventilation system is designed such that if operable there is negative pressure in the reactor room. 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.

b. The reactor shall not be operated in exposure room 1 or 2

   1. If the relative air pressure in the exposure room in use is greater than the reactor prep area (room 1105) except for periods of time not to exceed two continuous hours to permit repair, maintenance, or testing when the dampers shall be closed.

      or;

   2. The prep area RAM E3 or E6 is alarming.

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 dampers shall be closed, thereby isolating the reactor room. Therefore, operation of the reactor with the ventilation secured 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 operable 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 be secured unless the following radiation monitoring systems are operable:

a. Radiation Area Monitoring System:

    i. 2 RAMS on the reactor Deck (Room 3160) are operable

    ii. If operating in an exposure room (ER1 or ER2) the RAM adjacent to the exposure room in use shall be operable

b. Stack Gas Monitor: The stack gas monitor (SGM) shall sample and measure the gaseous effluent in the 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	Location(s) of readouts Audible alarms and visual Indicators
RAM	
Reactor Room (2 required)	Reactor and Control Rooms
Exp. Room 1 Area	Prep Area and Control Room
Exp. Room 2 Area	Prep Area and Control Room
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 normal reactor operations, reactor operations that generate and release significant quantities of argon-41 shall be curtailed for the remainder of the year as needed to ensure adherence with the 10 mrem constraint.

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;
    1. 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;
    2. The ventilation system shall be operational while the samples are being transferred from the pool or the reactor core.

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 an experimental container 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. All experiments shall be either secured or observed for mechanical

stability to ensure that unintended movement will not cause an unplanned reactivity change in excess of $1.00.

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 an experiment container 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. This specification ensures that unintended movement will not cause an unplanned reactivity change, 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

1. 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.

2. 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

Applicability

These specifications apply to the surveillance requirements for reactor systems.

Objective

The objective is to allow for variation in the execution of surveillance when maintenance issues prevent the timely completion surveillance items.

Specifications

Surveillance requirements may be deferred during reactor shutdown (except TS 4.4, TS 4.5.1 and TS 4.5.2) however; they shall be completed prior to reactor startup unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practical after reactor startup. Scheduled surveillance which cannot be performed with the reactor operating may be deferred until a planned reactor shutdown.

	TS	Possible to defer during shutdowns?	Required prior to routine operations?
1.	4.1 Reactor core parameters	Yes	Yes
2.	4.2.1 Reactor Control Systems	Yes	Yes
3.	4.2.2 Reactor Safety Systems	Yes	Yes
4.	4.2.3 Fuel Temperature	Yes	Yes
5.	4.2.4 Facility Interlock System	Yes	Yes
6.	4.3 Coolant Systems	Yes	Yes
7.	4.4 Ventilation Systems	No	N/A
8.	4.5.1 Monitoring System	No	Yes
9.	4.5.2 Effluents	No	N/A
10.	4.6 Reactor Fuel Elements	Yes	Yes
11.	4.2.2 Low Pool Water Scram	Yes	No

Bases

The surveillances items listed in the table above cannot be executed during times when required equipment are out of service. During these times, reactor operations are also suspended; therefore there is no decrement in safety deferring these surveillance items.

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) changes to core configuration (excluding in-core experiments).

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 each day of operation involving the movement of control rods, or prior to each continuous operation exceeding more than a day, and following any significant (>$0.25) core configuration changes. At a minimum excess reactivity shall be measured annually, not to exceed 15 months. This measurement is also a complete channel test of the linear power channel and log power channel.

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. 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.

Specifications

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, the transient rod system is channel tested to verify that the system is operable. 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. Channel tests along with periodic maintenance ensure consistent performance. Measurement of the rod drop times 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 reactor operations are planned.

b. A channel test of each of the reactor safety system channels in Table 2 and Table 3 with the exception of the exposure room emergency stop and AC power loss scrams for the intended mode of operation shall be performed weekly, whenever operations are planned.

c. Channel calibration, including verification of the setpoints for the high voltage loss to safety channel scrams, shall be made of the NP, NPP, NLW, NMP or any other console instrumentation designated to provide direct power level information to the operator, annually not to exceed 15 months.

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

e. The exposure room emergency stop and AC Power Loss scram**s** shall be tested annually, not to exceed 15 months.

f. The low pool water scram shall be tested weekly not to exceed 10 days whenever operations are planned.

g. The console manual scram button shall be tested weekly not to exceed 10 days whenever operations are planned.

Bases

TRIGA system components have proven operational reliability. Daily tests ensure reliable 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 to be 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 systems demonstrates that annual calibration and weekly channel tests provides reliable fuel temperature measurements. 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 in region 2 with the lead shield doors closed except during the use of the core dolly interlock override switch.

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

Bases

These 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 isolation of the facility ventilation system.

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

1. 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.

2. The relative air pressure in the reactor room and exposure room to be used shall be verified to be negative each day operations in the affected exposure room are planned.

3. The reactor exhaust damper flow failure closure system shall be tested each day that reactor operations are planned.

Basis

Experience accumulated over years of operation has demonstrated that 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.

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 effluents will not exceed the annual dose limits to the public.

Bases

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.

Specifications

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.

Bases

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 more frequently 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 (CAA) and the Reactor Control Area (RCA). The specific rooms included in each of those areas shall be listed in the Physical Security Plan for the AFRRI TRIGA Reactor Facility.

f. The reactor is housed in building #42 of the AFRRI complex and the restricted areas are located within that structure. The restricted areas are described in the SAR for the AFRRI reactor facility section 1.3.1 including figures 2-2 through 2-4 which describe the location of the reactor in the AFRRI complex. Figures 3-1 through 3-4 are the floor plan layouts which identify the reactor areas

Bases

The facility is designed so that the ventilation system will normally maintain a negative pressure with respect to adjacent areas, limiting personnel exposure. 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 including fuel follower control rods used in the reactor core.

Objective

The 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 elements 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. FUEL STORAGE

Applicability

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

Objective

The objective is to 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 permits sufficient natural convective cooling by water or air and that prevents the fuel element or fueled device temperature from exceeding design values. Storage shall be such that stored fuel elements and fueled devices 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.

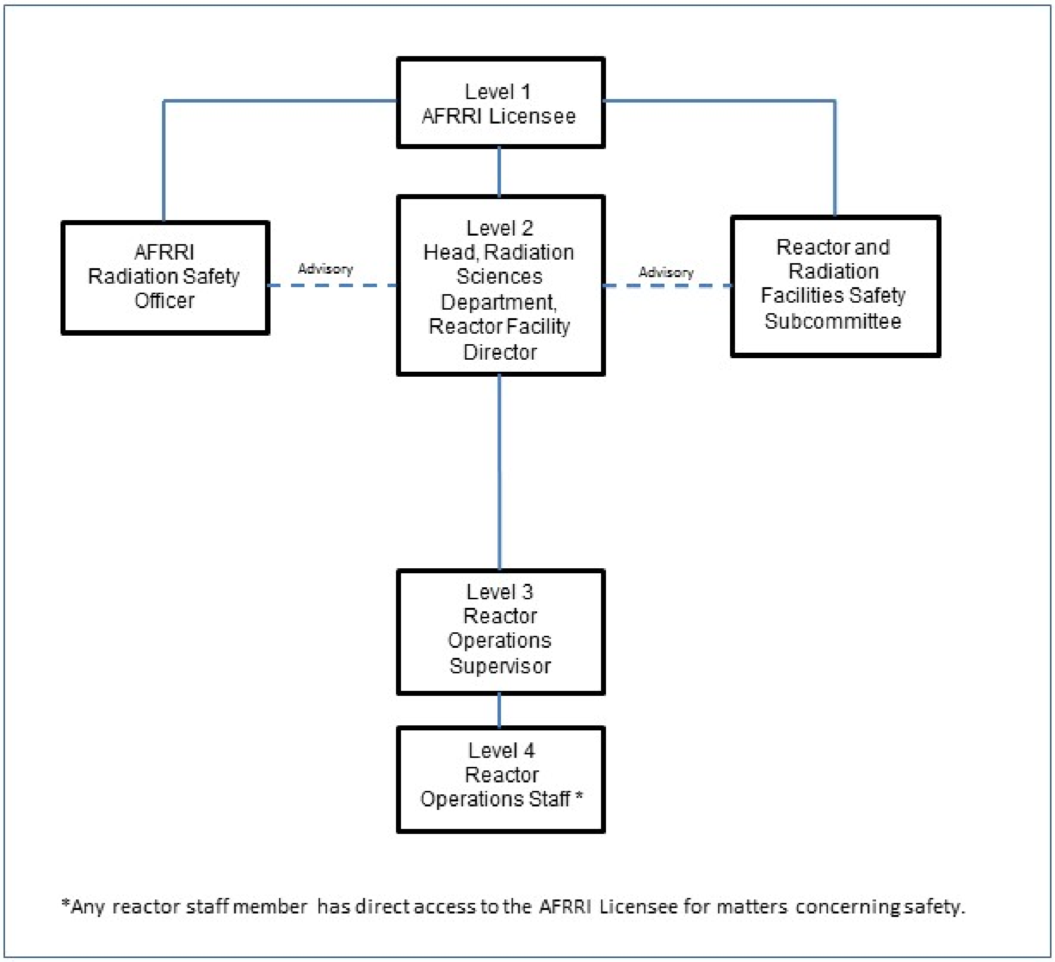
6.0. ADMINISTRATIVE CONTROLS

6.1. ORGANIZATION

6.1.1. STRUCTURE

The organizational structure of the reactor facility is depicted below.

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



MANAGEMENT LEVELS

Level 1: AFRRI Director: Responsible for the facility license.

Level 2: Reactor Facility Director: Responsible for reactor facility operations and administration shall report to Level 1.

Level 3: Reactor Operations Supervisor: Responsible for the day-to-day operation of the reactor and shall report to Level 2.

Level 4: Reactor Operating Staff: Licensed reactor operators and senior reactor operators and trainees. These individuals shall report to Level 3 for matters involving reactor operations.

6.1.2. RESPONSIBILITY

The AFRRI Licensee shall have license responsibility for the reactor facility. The Reactor Facility Director (RFD) shall be responsible for administration and operation of the reactor facility and for determination of applicability of procedures, experiment authorizations, maintenance, and operations. The Reactor Facility Director may designate an individual who meets the requirements of Technical Specifications 6.1.3.1.a to discharge these 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 be responsible for implementing the radiation safety program for the AFRRI TRIGA reactor. The requirements of the radiation safety program are established in 10CFR20. The program shall comply with the requirements in 10CFR20. Additional guidelines from ANSI/ANS-15.11-1993;R2004 "Radiation Protection at Research Reactor Facilities" should be considered.

6.1.3. STAFFING

6.1.3.1. Selection of Personnel

a. AFRRI Licensee

   The AFRRI Licensee is the AFRRI Director. The AFRRI Director has management responsibility for adhering to the terms and conditions of the AFRRI reactor license R-84, the AFRRI O2 byproduct license, the AFRRI Technical Specifications and for protecting the health and safety of the facility staff and members of the public.

b. 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. Education and/or experience that is job-related may be substituted for a degree on a case-by-case basis.

c. 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.

d. 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.

e. 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. <u>Operations</u>

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.; 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. <u>Training of Personnel</u>

Training and retraining program shall be maintained to ensure adequate levels of proficiency in persons involved in the reactor and reactor operations. The training and retraining program will be consistent with the NRC-approved reactor requalification plan.

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 shall be directly responsible to the AFRRI Licensee. The subcommittee 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. 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 a minimum of four members that can vote and occupy the following positions; the Chairman (or designated alternate), the Reactor Facility Director (or designated alternate), the Radiation Safety Officer (or designated alternate), and one or more non-AFRRI member. A majority of those present shall be regular members. The operating staff shall not constitute a majority. A member may occupy two positions but may only vote once.

6.2.3.4. Voting Rules

Each regular RRFSS member shall have one vote. Each special RRFSS 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 subcommittee, final minutes shall be submitted to level one management for review within 3 months.

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.

b. Changes to procedures, equipment, or systems that change the original intent or use, are non-conservative, or those that meet any of the applicable criteria in Section 50.59 of 10 CFR;

c. Proposed tests or experiments that could affect reactivity or result in the uncontrolled release of radioactivity, or those that might meet any of the applicable criteria in Section 50.59 of 10 CFR;

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 safety significance;

f. Operating abnormalities having safety significance;

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

h. Audit reports of the reactor facility operations.

6.2.5. AUDIT FUNCTION

Audits of reactor facility operations 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 either annually, not to exceed 15 months, or biennially, not to exceed 30 months. The audit frequency is indicated below for each item (in parenthesis). A report of the findings and recommendations resulting from the audit shall be submitted to the AFRRI Licensee within three months after the report has been received. Deficiencies uncovered that affect reactor safety shall immediately be reported to level one management. Audits may be performed by one or more individuals who need not be RRFSS members. 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 (annually);

b. Performance, training, and qualifications of the reactor facility staff (biennially);

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

d. Facility emergency plan and implementing procedures (biennially);

e. Facility Physical Security Plan (biennially);

f. Any other area of facility operations considered appropriate by the RRFSS or the AFRRI Director (annually); and

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

6.3. PROCEDURES

Written procedures 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. Operational 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. Surveillance, testing, maintenance, and calibration of instruments, components, and systems involving nuclear safety;

c. Personnel radiation protection consistent with 10 CFR Part 20;

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

e. Fuel loading, unloading, and movement within the reactor core; and

f. Reactor 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, when known, the cause and contributing factors (2) effects of the violation on facility components, structures, or systems, the health and safety of personnel and the public 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 for Fuel Temperature.

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

c. Malfunction of a required reactor safety system component during operation that renders or 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. Reactor trips resulting from a known cause are excluded.

e. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy causes or 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. Abnormal and significant degradation of the reactor coolant boundary or confinement 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 original 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 facility life of conditions not specifically considered in the Safety Analysis Report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

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

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

b. The occurrence shall be reported to the 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 Document Control Desk.

a. Annual Operating Report: Routine operating reports covering the operation of the reactor during the previous calendar year shall be submitted by March 31 of each year. 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 action was required to ensure safe operation of the reactor, including the reasons and the corrective actions taken;

   4. Discussion of the major safety-significant corrective and/or preventive 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 monthly basis)

      i. Radioactivity discharged during the reporting period:

         Total radioactivity released (in curies);

         Concentration limits used and isotopic composition 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 and combined activity in curies of 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 limit received by reactor personnel or visitors to the reactor facility;

b. Other Reports: A report shall be submitted to the USNRC 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 THAT SHALL 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 meetings of the RRFSS.

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

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

6.7.3. RECORDS THAT SHALL 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.

e. Reviews and reports pertaining to a violation of the safety limit, limiting safety system setting (LSSS) or limiting condition of operation (LCO).