ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE (AFRRI)
RESEARCH REACTOR
LICENSE NO. R-84
DOCKET NO. 50-170

TECHNICAL RAI RESPONSES (DATED 10/20/2011)

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

10CFR 50.90

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

October 20, 2011

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE APPLICATION FOR LICENSE RENEWAL (TAC NO. ME1587)

Sir:

By letter dated July 19, 2010, the Nuclear Regulatory Commission requested additional information necessary to allow processing of our research reactor license renewal application (License R-84, Docket 50-170).

Our response to the Technical Specification-related questions (Questions 14-41) and the revised Technical Specifications were submitted on September 27, 2010. Discussions with Mr. Walter Meyer have resulted in further changes to both the question answers and the Technical Specifications. The Answers to TS-Related Questions (14-41) and the Revised Technical Specifications submitted on September 27, 2010 should be withdrawn and replaced by the enclosed two documents.

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

I declare under penalty of perjury that the foregoing and all enclosed information is true and correct to the best of my knowledge. Executed on October 20, 2011.

Enclosure:
as

MARK A. MELANSON
COL, MS, USA
Director

A020
NRR

TECHNICAL SPECIFICATIONS FOR THE AFRRI REACTOR FACILITY

20 OCTOBER 2011

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
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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 CALIBRATION

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

1.3. CHANNEL CHECK

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

1.4. CHANNEL TEST

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

1.5. CORE GRID POSITION

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

1.6. 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.

1.7. EXPERIMENT

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

1.8. EXPERIMENTAL FACILITIES

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

a. Exposure Room #1

b. Exposure Room #2

c. Reactor Pool

d. Core Experiment Tube

e. Portable Beam Tubes

f. Pneumatic Transfer System

g. Incore Locations

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

1.9. FUEL ELEMENT

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

1.10. INITIAL REACTOR STARTUP

The first reactor startup following fuel element relocation within the core.

1.11. INSTRUMENTED FUEL ELEMENT

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

1.12. LIMITING SAFETY SYSTEM SETTING

Limiting safety system settings are those limiting values for settings of the safety channels by which point protective action must be initiated. The settings are chosen so that automatic protective action will terminate the abnormal situation before a safety limit is reached.

1.13. MEASURED VALUE

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

1.14. MEASURING CHANNEL

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

1.15. MOVABLE EXPERIMENT

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

1.16. ON CALL

A person is considered on call if:

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

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

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

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

1.17. OPERABLE

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

1.18. OPERATING

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

1.19. PULSE MODE

Operation in the pulse mode shall mean that the reactor is intentionally placed on a prompt critical excursion by making a step insertion of reactivity above critical with the transient rod, utilizing the appropriate scrams in Table 2 and the appropriate interlocks in Table 3. The reactor may be pulsed from a critical or subcritical state.

1.20. 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.

1.21. REACTOR OPERATION

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

1.22. REACTOR OPERATOR

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

1.23. REACTOR SAFETY SYSTEMS

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

1.24. REACTOR SECURED

The reactor is secured when:

a. Either sufficient fuel is removed to ensure a $1.00 (or greater) shutdown margin both with the most reactive control rod removed and negligible reactivity worth of xenon, thus ensuring there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection;

b. Or the following conditions exist:

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

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

   3. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are magnetically decoupled from the control rods;

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

1.25. REACTOR SHUTDOWN

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

1.26 REACTOR STARTUP

Startup of the reactor and approach to power following a period when the reactor is shutdown or secured.

1.27. REFERENCE CORE CONDITION

The reactor is in the reference core condition when the core is at ambient temperature and the reactivity worth of xenon is negligible.

1.28. SAFETY CHANNEL

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

1.29. SAFETY LIMIT

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

1.30. SCRAM TIME

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

1.31. SECURED EXPERIMENT

A secured experiment is any experiment or experimental component held in a stationary position relative to the reactor by mechanical means. The experiment must not move relative to the reactor as a result of any credible forces.

1.32. SENIOR REACTOR OPERATOR

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

1.33. 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.

1.34. 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.

1.35. 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.

1.36. STEADY STATE MODE

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

1.37. 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.

1.38. TRUE VALUE

The true value is the actual value of a parameter.

2.0. SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTING

2.1. SAFETY LIMIT: FUEL ELEMENT TEMPERATURE

Applicability

This specification applies to the temperature of the reactor fuel.

Objective

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

Specification

The maximum temperature in a standard TRIGA fuel element shall not exceed 1000°C under any condition of operation.

Basis

The important parameter for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification, especially since it can be measured. A loss in the integrity of the fuel element cladding could arise from a buildup of excessive pressure between the fuel-moderator and the cladding if the fuel temperature exceeds the safety limit. The pressure is caused by the presence of air, fission product gases, and hydrogen from the dissociation of the hydrogen and zirconium in the fuel-moderator. The magnitude of this pressure is determined by the fuel-moderator temperature and the ratio of hydrogen to zirconium in the alloy.

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

2.2. LIMITING SAFETY SYSTEM SETTING FOR FUEL TEMPERATURE

Applicability

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

Objective

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

Specification

There shall be two fuel temperature safety channels. The limiting safety system setting for these instrumented fuel elements' temperature shall not exceed 600°C. One channel shall utilize an instrumented element in the "B" ring, and the second channel shall utilize an instrumented element in the "C" ring.

Basis

The limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated, preventing the safety limit from being exceeded. A setting of 600°C provides a safety margin of at least 400°C for standard TRIGA stainless-steel-clad fuel elements. Part of the safety margin is used to account for the difference between the true and the measured temperatures resulting from the actual location of the thermocouple. If the instrumented element is located in the hottest position in the core, the difference between the true and measured temperatures will be only a few degrees. If the instrumented element is located in a region of lower temperature, the measured temperature will differ by a greater amount from that actually occurring at the core hot spot. To lessen this difference, the requirement is to locate the element in the hottest region of the core. These margins are sufficient to account for the remaining uncertainty in the accuracy of the fuel temperature measurement channel and any overshoot in reactor power resulting from a reactor transient during steady state mode operation.

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

3.0. LIMITING CONDITIONS FOR OPERATIONS

3.1. REACTOR CORE PARAMETERS

3.1.1. STEADY STATE OPERATION

Applicability

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

Objective

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

Specification

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

Basis

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

3.1.2. PULSE MODE OPERATION

Applicability

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

Objective

The objective is to assure that the fuel temperature safety limit will not be exceeded.

Specification

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

Basis

Based upon the calculations detailed in Section 4.5.9. of the AFRRI Safety Analysis Report, an insertion of 2.45% $\Delta k/k$ results in a peak fuel temperature of less than 830°C, thereby staying within the vendor's specifications.

3.1.3. REACTIVITY LIMITATIONS

Applicability

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

Objective

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

Specifications

a. The reactor shall not be operated with the maximum available excess reactivity 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 fully withdrawn 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 power if all control rods are removed.

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

3.1.4. SCRAM TIME

Applicability

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

Objective

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

Specification

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

Basis

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

3.2. REACTOR CONTROL AND SAFETY SYSTEMS

3.2.1. REACTOR CONTROL SYSTEM

Applicability

This specification applies to the channels monitoring the reactor core, which must provide information to the reactor operator during reactor operation. 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 assure safe operation of the reactor.

Specifications

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

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

Table 1 Measuring Channels

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

Bases

Fuel temperature displayed at the control console gives continuous information on this parameter, which has a specified safety limit. The power level channels assure that radiation indicating reactor core parameters are adequately monitored for both steady state and pulsing modes of operation. The specifications on reactor power level indication are included in this section, since the power level is related to the fuel temperature. The four control rod drives must be operable for the safe operation of the reactor.

3.2.2. REACTOR SAFETY SYSTEM

Applicability

This specification applies to the reactor safety system.

Objective

The objective is to specify the minimum number of reactor safety system channels that must be operable for safe operation.

Specification

The reactor shall not be operated unless the safety systems described in Tables 2 and 3 are operable for the specific mode of operation.

Table 2 Minimum Reactor Safety System Scrams

Channel	Maximum Set Point	Minimum Number in Mode	
		Steady State	Pulse
Fuel Temperature	600°C	2	2
Percent Power, High Flux	1.1 MW	2	0
Console Manual Scram Button	Closure switch	1	1
High Voltage Loss to Safety Channel	20% Loss	2	1
Pulse Time	15 seconds	0	1
Emergency Stop	Closure switch	1	1
(1 each exposure room, 1 on console)			
Pool Water Level	2 inches below siphon breaks	1	1
Watchdog (DAC to CSC)	On digital console	1	1

Basis

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

Table 3 Minimum Reactor Safety System Interlocks

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

Basis

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

3.2.3. FACILITY INTERLOCK SYSTEM

Applicability

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

Objective

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

Specifications

Facility interlocks shall be provided so that:

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

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

c. The lead shield doors cannot be opened to allow movement into the exposure room projection unless a warning horn has sounded in that exposure room, or unless two licensed operators have visually

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

Bases

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

3.3. COOLANT SYSTEMS

Applicability

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

Objective

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

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

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

Specifications

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

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

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

d. Both audible and visual alarms shall be provided to alert the AFRRI security guards and other personnel to any drop in reactor pool water level greater than 12 inches.

Bases

Manufacturer's data states that the resins in the water purification system break down with sustained operation in excess of 60°C. The 2 micromhos/cm is an

acceptable level of water contaminants in an aluminum/stainless-steel system of the type at AFRRI. Based on experience, activation at this level does not pose a significant radiological hazard. Also, the conductivity limits are consistent with the fuel vendor's experience and with similar reactors.

3.4. VENTILATION SYSTEM

Applicability

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

Objective

The objective is to assure that the ventilation system is operable.

Specification

The reactor shall not be operated unless the facility ventilation system exhaust fan is operating, except for periods of time during which the dampers shall be closed. In the event of a release of airborne radioactivity in the reactor room above both routine reactor operation and normal background values, the ventilation system to the reactor room shall be secured via closure dampers automatically by a signal from the reactor deck air particulate monitor.

Basis

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

3.5. RADIATION-MONITORING SYSTEM AND EFFLUENTS

3.5.1. MONITORING SYSTEM

Applicability

This specification applies to the functions and essential components of the area radiation monitoring equipment and the system for continuously

monitoring radioactivity and radiation levels, which must be available during reactor operations.

Objective

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

Specifications

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

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

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

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

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

Table 4 Locations of Radiation Monitoring Systems

Monitor	Location of Alarm (A = Audible, V = Visual)	Readout Location
RAM		
R1, Reactor Room	Control Room A & V	Control Room
R2, Reactor Room	Control Room V	Control Room
E3, Exp. Room 1 Area	Control Room V	Control Room
E6, Exp. Room 2 Area	Control Room V	Control Room
SGM		
Reactor Exhaust	Control Room V	Control Room
CAM		
Reactor Room	Control Room A & V	Control Room

Bases

This system is intended to characterize the normal operational radiological environment of the facility and to aid in evaluating any abnormal operations or conditions. The radiation monitors provide information to the operating personnel of any existing or impending danger from radiation, to give sufficient time to evacuate the facility and take necessary steps to prevent the spread of radioactivity to the surroundings. The automatic closure of the ventilation system dampers provides reactor room isolation from the outside environment, in the event of airborne radioactivity within the reactor room from fission products decay.

3.5.2. EFFLUENTS: ARGON-41 DISCHARGE LIMIT

Applicability

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

Objective

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

Specifications

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

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

Bases

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

3.6. LIMITATIONS ON EXPERIMENTS

Applicability

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

Objective

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

Specifications

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

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

b. Each fueled experiment shall be limited so that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 1.3 curies and the maximum strontium-90 inventory is not greater than 5 millicuries.

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

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

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

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

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

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

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

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

Bases

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

b. The 1.3 curie limitation on iodine-131 through -135 assures that, in the event of malfunction of a fueled experiment leading to total release of radioactive material including fission products, the dose to members of the public will not exceed the limits of 10 CFR 20.

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

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

e. The maximum worth of experiments is limited so that their removal from the reactor at the reference core condition will not result in the reactor achieving a power level high enough to exceed the core temperature safety limit. The

three dollar ($3.00) limit is less than the SAR-analyzed authorized pulse magnitude. Limiting moveable or unsecured experiments to a worth less than $1.00 will prevent unintended pulsing of the reactor and unnecessary fuel mechanical stress.

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

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

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

3.7. SYSTEM MODIFICATIONS

Applicability

This specification applies to any system related to reactor safety.

Objective

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

Specification

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

Basis

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

3.8. ALARA

Applicability

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

Objective

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

Specification

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

Basis

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

3.9. FUEL ELEMENT MEASUREMENTS

Applicability

This specification applies to all TRIGA fuel elements in the reactor core or in storage.

Objective

The objective is to ensure that only undamaged fuel elements are used in the reactor core.

Specification

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

Basis

This specification is based on the parameters most likely to affect the fuel element cladding of a pulsing reactor to minimize the possibility of fuel element cladding failure during reactor operation.

4.0. SURVEILLANCE REQUIREMENTS

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

4.1. REACTOR CORE PARAMETERS

Applicability

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

Objective

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

Specifications

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

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

c. The control rods shall be visually inspected for deterioration annually, not to exceed 15 months.

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

e. The core excess reactivity shall be measured at the beginning of each day of operation involving the movement of control rods, or prior to each continuous operation extending more than a day. During extended reactor shutdown periods, the core excess reactivity shall be measured at least annually, not to exceed 15 months.

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

Bases

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

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

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

4.2. REACTOR CONTROL AND SAFETY SYSTEMS

4.2.1. REACTOR CONTROL SYSTEMS

Applicability

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

Objective

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

Specification

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

Basis

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

4.2.2. REACTOR SAFETY SYSTEMS

Applicability

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

Objective

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

Specifications

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

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

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

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

Bases

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

4.2.3. FUEL TEMPERATURE

Applicability

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

Objective

To ensure operability of the fuel temperature-measuring channels.

Specifications

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

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

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

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

Bases

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

4.2.4. FACILITY INTERLOCK SYSTEM

Applicability

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

Objective

To ensure performance and operability of the facility interlock system.

Specifications

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

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

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

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

Bases

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

4.2.5. REACTOR FUEL ELEMENTS

Applicability

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

Objective

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

Specification

Half of the fuel elements present in the reactor core, to include all the fuel follower control rods (FFCRs), shall be inspected for damage or deterioration, and measured for length and bow annually (not to exceed 15 months). During a two year cycle, all fuel elements except those in long-term storage shall be measured and inspected. Fuel elements and FFCRs indicating an elongation greater than 0.100 inch, a lateral bending greater than 0.0625 inch, or significant visible damage shall be considered damaged, and shall not be used in the reactor core. If damage, deterioration, or unacceptable length or bow measurements are found in one or more fuel elements or FFCRs, all fuel elements in the core shall be inspected for damage or deterioration and measured for length and bow. Fuel elements and fuel follower control rods shall also be inspected and measured if they have been in the core for 500 pulses of insertion greater than $2.00 since their last biennial inspection and measurement. Fuel elements in long-term storage need not be measured or inspected until returned to the core; however fuel elements routinely moved to temporary storage shall be measured and inspected every 500 pulses of insertion greater than $2.00 or annually (not to exceed 15 months), whichever occurs first.

Basis

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

The limit of transverse bend has been shown to result in no difficulty in disassembling the core. Extensive tests on the dimensional stability of TRIGA fuel elements showed no transverse bend (bow) failures until the elements were subjected to at least 100 $4.60 pulses, well beyond the pulse limit for the AFRRI reactor.

4.3. COOLANT SYSTEMS

Applicability

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

Objective

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

Specifications

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

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

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

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

Bases

Based on experience, observation at these intervals provides acceptable surveillance of limits that assure that fuel cladding corrosion and neutron activation of dissolved materials will not occur. Testing of the audible and visual alarms ensures that AFRRI personnel will be able to alert the appropriate staff and elicit timely response to a pool water loss. The pool water temperature is continuously displayed on the reactor console and is manually recorded at the beginning of each day of reactor operations. The conductivity of the bulk pool water is restricted to help minimize the possibility of corrosion in fuel cladding or reactor system components.

4.4. VENTILATION SYSTEM

Applicability

This specification applies to the facility ventilation system isolation.

Objective

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

Specification

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

Basis

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

4.5. RADIATION-MONITORING SYSTEM

Applicability

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

Objective

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

Specification

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

Basis

Experience has shown that quarterly verification of radiation area monitoring and air particulate monitoring system set points in conjunction with a quarterly channel test is adequate to correct for any variation in the system due to a change of operating

characteristics over a long time span. Annual calibration ensures that the units are within the specifications demanded by the extent of use.

4.6. EFFLUENTS

Applicability

This specification applies to surveillance requirements for environmental monitoring.

Objective

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

Specifications

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

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

Basis

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

5.0. DESIGN FEATURES

5.1. SITE AND FACILITY DESCRIPTION

Applicability

This specification applies to the building that houses the reactor.

Objective

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

Specifications

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

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

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

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

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

Bases

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

5.2. REACTOR CORE AND FUEL

5.2.1. REACTOR FUEL

Applicability

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

Objective

These objectives are to (1) assure that the fuel elements are designed and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics, and (2) assure that the fuel elements used in the core are substantially those analyzed in the Safety Analysis Report.

Specifications

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

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

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

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

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

Bases

A maximum uranium content of 9 weight percent in a standard TRIGA element is greater than the design value of 8.5 weight percent, and encompasses the maximum probable variation in individual elements. Such an increase in loading would result in an increase in power density of less than 6%. An increase in local power density of 6% in an individual fuel element reduces the safety margin by 10%, at most. The hydrogen-to-zirconium ratio of 1.7 will produce a maximum pressure within the cladding well below the rupture strength of the cladding. The local power density of a 12.0 weight percent fuel follower is 21% greater than an 8.5 weight percent

standard TRIGA fuel element in the D-ring. The volume of fuel in a fuel follower control rod is 56% of the volume of a standard TRIGA fuel element. Therefore, the actual power produced in the fuel follower rod is 33% less than the power produced in a standard TRIGA fuel element in the D-ring.

5.2.2. REACTOR CORE

Applicability

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

Objective

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

Specifications

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

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

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

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

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

Bases

Standard TRIGA cores have been in use for years, and their safe operational characteristics are well documented. Experience with TRIGA reactors has shown that fuel element bowing that could result in touching has occurred without deleterious effects. The elongation limit has been specified to (a) assure that the cladding material will not be subjected to stresses that could

cause a loss of integrity in the fuel containment, and (b) assure adequate coolant flow.

5.2.3. CONTROL RODS

Applicability

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

Objective

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

Specifications

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

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

Bases

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

5.3. SPECIAL NUCLEAR MATERIAL STORAGE

Applicability

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

Objective

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

Specification

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

Basis

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

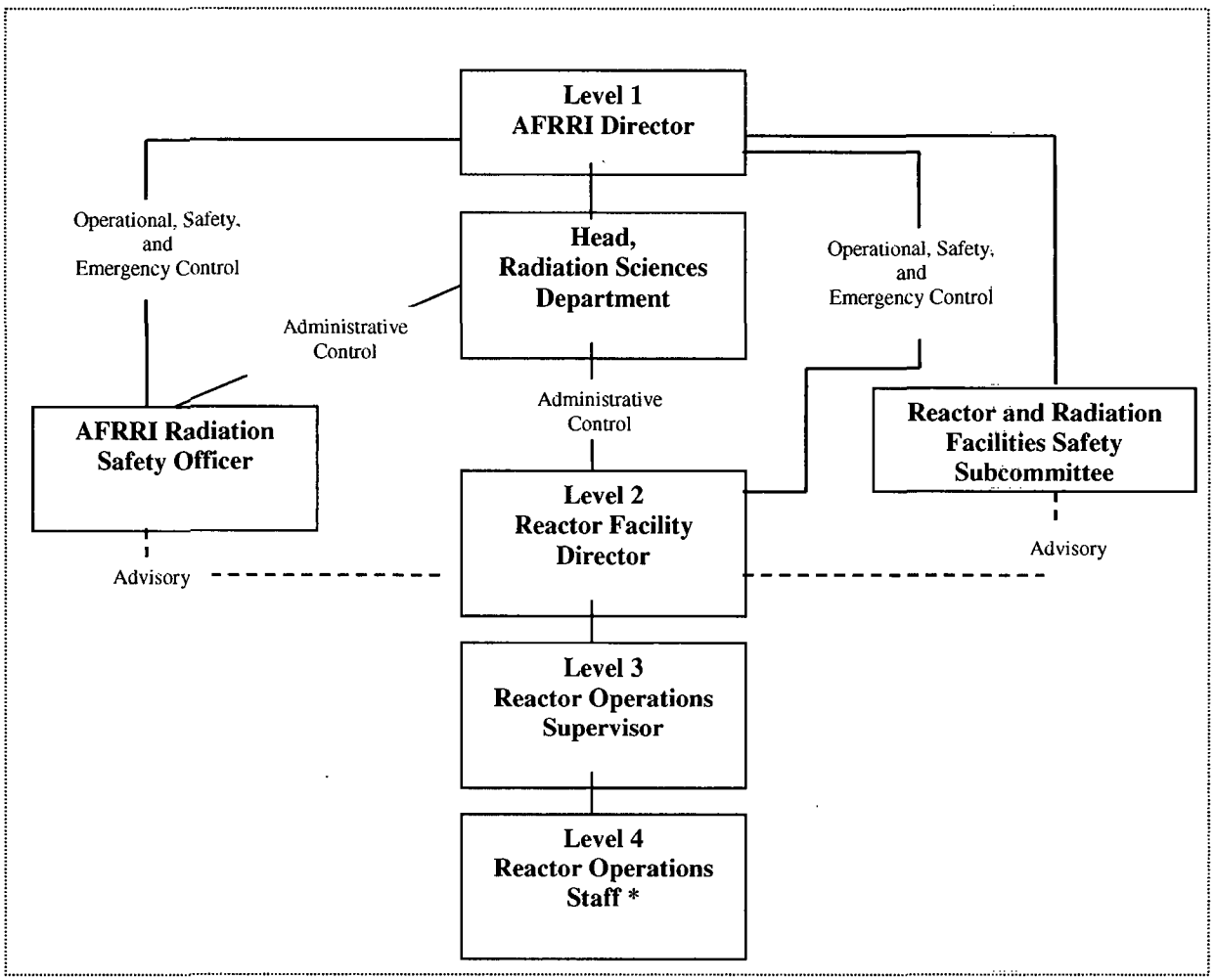
6.0. ADMINISTRATIVE CONTROLS

6.1. ORGANIZATION

6.1.1. STRUCTURE

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

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



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

6.1.2. RESPONSIBILITY

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

6.1.3. STAFFING

6.1.3.1. Selection of Personnel

a. Reactor Facility Director

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

b. Reactor Operations Supervisor (ROS)

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

c. Reactor Operators/Senior Reactor Operators

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

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

6.1.3.2. Operations

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

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

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

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

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

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

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

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

2. Initial reactor startup and approach to power.

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

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 Director) or designee

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

3. Other operations personnel (Reactor Staff, ROS)

6.1.3.3. Training of Personnel

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

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

6.2.1. COMPOSITION AND QUALIFICATIONS

6.2.1.1. Composition

a. Regular RRFSS Members (Permanent Members)

1. The following shall be members of the RRFSS:

a. AFRRI Radiation Safety Officer (RSO)

b. AFRRI Reactor Facility Director (RFD)

c. AFRRI Facility Radiation Manager (FRM)

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

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 item 2.b. above as appointed by the AFRRI Director.

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

c. Nonvoting members as invited by the AFRRI Director.

6.2.1.2. Qualifications

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

6.2.2. FUNCTION AND AUTHORITY

6.2.2.1. Function

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

6.2.2.2. Authority

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

6.2.3. CHARTER AND RULES

6.2.3.1. Alternates

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

6.2.3.2. Meeting Frequency

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

6.2.3.3. Quorum

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

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

6.2.3.4. Voting Rules

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

6.2.3.5. Minutes

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

6.2.4. REVIEW FUNCTION

The RRFSS shall review:

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

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

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

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

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

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

g. Events that have been reported to the NRC.

h. Audit reports of the reactor facility operations.

6.2.5. AUDIT FUNCTION

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

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

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

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

d. Facility emergency plan and implementing procedures.

e. Facility security plan and implementing procedures.

f. Any other area of facility operations considered appropriate by the RRFSS or the AFRRI Director.

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

6.3. PROCEDURES

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

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

b. Reactor staff training program.

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

d. Personnel radiation protection consistent with 10 CFR 20.

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

f. Reactor core loading and unloading.

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

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

6.4. REVIEW AND APPROVAL OF EXPERIMENTS

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

a. Reactor Facility Director

b. Health Physics Division

c. Reactor and Radiation Facilities Safety Subcommittee (RRFSS)

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

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

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

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

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

6.5. REQUIRED ACTIONS

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

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

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

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

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

6.5.2. REPORTABLE OCCURRENCES

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

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

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

c. Malfunction of a required reactor safety system component during operation that could render the system incapable of performing its

intended safety function unless the malfunction or condition is caused by maintenance.

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

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

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

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

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

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

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

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

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

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

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

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

6.6. OPERATING REPORTS

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

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

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

   Each annual operating report shall include:

   1. A brief narrative summary of:

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

b. Results of surveillance test and inspections.

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

3. List of the unscheduled shutdowns, including the reasons and the corrective actions taken, if applicable.

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

5. A brief description of:

   a. Each change to the facility to the extent that it changes a description of the facility in the Safety Analysis Report.

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

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

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

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

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

      i. Radioactivity discharged during the reporting period

         Total radioactivity released (in curies)

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

         Total radioactivity (in curies), released by nuclide during the reporting period, and based on representative isotopic analysis

Average concentration at point of release (in microcuries/cc) during the reporting period

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

b. Gaseous Waste (summarized on a quarterly basis)

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

Argon-41

Particulates with half-lives greater than eight days

c. Solid Waste (summarized on a quarterly basis)

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

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

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

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

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

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

6.7. RECORDS

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

a. Operating logs or data that shall identify:

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

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

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

4. Rod worth measurements and other reactivity measurements

b. Principal maintenance operations

c. Reportable occurrences

d. Surveillance activities required by Technical Specifications

e. Facility radiation and contamination surveys

f. Experiments performed with the reactor

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

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

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

g. Changes to operating procedures

h. Fuel inventories and fuel transfers

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

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

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

l. Records of meetings of the RRFSS

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

a. Training exams

b. Requalification records

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

a. Gaseous and liquid radioactive effluents released to the environs

b. Appropriate offsite environmental monitoring surveys

c. Radiation exposures for all personnel

d. Updated as-built drawings of the facility

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

ANSWERS TO TS-RELATED QUESTIONS
QUESTIONS 14-41

(Revised 20 October 2011)

14. TS 1.0, Definitions: ANSI/ANS-15.1-2007, Section 1 defines Reactor Secured and Reactor Shutdown. Please evaluate AFRRI TS 1.19 against the standard definition of Reactor Secured and TS 1.20 against the standard definition to Reactor Shutdown. Propose changes to meet ANSI/ANS-15.1-2007 or justify your definitions.

The definition of Reactor Shutdown is revised to match the definition in ANSI/ANS-15.1. The definition of Reactor Secured is revised to incorporate all the elements of the ANSI/ANS-15.1 definition. Subparagraph c. is worded such that no work on fuel or other reactor components or high-value experiments would occur unless sufficient fissile material is removed to prevent criticality under optimum conditions of moderation and reflection as required by ANSI/ANS-15.1 subparagraph (1).

15. TS 1.0, Definitions: ANSI/ANS-15.1-2007, Section 1 provides a definition of Reference Core Condition. TS 1.5 defines Cold Critical at 40 degrees C with no specific condition for Xenon reactivity. Please evaluate the definition of Cold Critical against the ANSI/ANS-15.1-2007 standard definition for Reference Core Condition and propose and justify a definition that can be applied to the Limiting Conditions for Operations (LCO) for Excess Reactivity and Shutdown Margin in TS 3.1.3, or justify the current definition.

The definition of Cold Critical has been removed and replaced by a definition for Reference Core Condition that complies with ANSI/ANS-15.1. The term "cold critical" will also be replaced in Section 3.1.3.

16. TS 1.0 Definitions: ANSI/ANS-15.1-2007, Section 1 provides definitions for key terminology utilized in technical specifications. Please include definitions of Excess Reactivity, Scram Time, Shall, Should and May, Secured Experiment, and Movable Experiment, in TS 1.0 Definitions, or provide a basis for not defining these items.

Equivalent definitions have been added to the technical specifications. In some cases, the wording is not identical to ANSI/ANS-15.1 but is more in keeping with AFRRI terminology. The definition of Shutdown Margin has been changed to agree with ANSI/ANS-15.1 except that AFRRI has no "nonscramable" rods.

17. TS 1.0 Definitions: ANSI/ANS-15.1-2007, Section 6.7.2 provides a schedule of events that require special reports. The Reportable Occurrence definition in TS 1.21© which addresses these special reports is not consistent with the guidance that specifies "malfunction caused by maintenance" instead of "malfunction discovered during tests." Please propose changes or justify not doing so.

All discussion of Reportable Occurrences has been moved to TS Section 6.5.2. Subsection (c) in TS 6.5.2 is changed to meet the requirements of ANSI/ANS-15.1.

18. TS 3.1.2: ANSI/ANS-15.1-2007, Section 3.1(3) provides guidance for the LCO for pulse limits. In TS 3.1.2, the LCO for pulse mode operations specifies the maximum step insertion of reactivity shall be $4.00 in the pulse mode. Analyses performed by the licensee indicate that this magnitude pulse may achieve a peak fuel temperature that exceeds the fuel vendor's recently recommended peak temperature of 830 degrees C during pulse mode operations. Please analyze and discuss how TS 3.2.1 should be revised to meet the fuel vendor recommendation.

TS 3.1.2 has been modified to change the maximum allowable pulse size to $3.50. This smaller size pulse will not affect AFRRI operations since experimental pulses over $3.50 have not been required in more than 20 years and are not expected to be needed in the future. The TS basis has been revised to indicate the current temperature analysis from the new Safety Analysis Report Chapter 4 submitted on March 4, 2010.

19. TS 3.1.3.b: ANSI/ANS-15.1-2007, Section 3.1(4) indicated that limits shall be established for core configurations. NUREG-1537, Appendix 14.1, Section 3.1(4) specifies that a special cor3e configuration should be included in the LCOs for the specifies that a special core for maintenance or other condition when control rods need to be removed from the core for maintenance or other purpose. How does AFRRI control the core configuration when removing a control rod from the core in order to maintain the shutdown margin as required by TS 3.1.3.b?

When a control rod is fully removed from the core for maintenance, the core configuration is controlled by first removing enough fuel elements so that, if both the most reactive control rod and another control rod were removed simultaneously, the core would still be subcritical by at least $0.50. In practice, three C-ring fuel elements worth about $3.00 total are removed giving a final shutdown margin of about $3.50 after both control rods are removed. This procedure is detailed in the reactor annual maintenance checklist. Two control rods are never fully removed from the core simultaneously and our procedure well exceeds the TS requirements.

20. TS 3.2.1, Table 1: ANSI/ANS-15.1-2007, Section 3.2 provides guidance for the reactor control and safety systems. In Table 1 in TS 3.2.1 identifies a Pulse Energy Integrating Channel. This instrument is not described in the SAR. Please clarify.

The channel name in Table 1 is changed to "Power Pulsing Channel." This is the name used in the revised SAR Chapter 7 that will be submitted as part of the overall relicensing application. The four relevant pages of Chapter 7 are included as Attachment 1.

7 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 SUMMARY DESCRIPTION

The reactor is operated from a Control System Console (CSC) located in the control room. The Data Acquisition Cabinet (DAC) is located in the reactor room along with wall-mounted cabinets that house the digital neutron log power channel (NM-1000) and the driver modules for the control rod stepping motors.

The operating mode of the reactor is determined by four push-button mode selector switches on the console. In Automatic and Steady-State modes, the reactor can operate at power levels up to 1 MW. In Square Wave mode, a step insertion of reactivity rapidly raises reactor power to a steady-state level up to 1 MW. In the Pulse mode, a large-step insertion of reactivity results in a short duration reactor power pulse.

The reactor instrumentation is all solid-state circuitry with a mixture of analog and digital modes of operation.

Additional ventilation system instrumentation and controls are housed in a separate cabinet near the console.

7.2 DESIGN OF INSTRUMENTATION AND CONTROL SYSTEMS

Three independent power measuring channels provide for a continuous indication of power from the source level to peak power resulting from the maximum allowed pulse reactivity insertion. Two of the channels are analog instruments and the third is a digital instrument. Trips are provided for over power and loss of detector high voltage on the two analog channels. Fuel temperature is measured for display as well. Other parameters not used by the reactor protection system are also monitored and displayed.

The instrumentation and control system is designed to provide the following:

- complete information on the status of the reactor and reactor-related systems
- a means for manually withdrawing or inserting control rods
- automatic control of reactor power level
- automatic scrams in response to over power, loss of detector high voltage, or high fuel temperatures
- automatic scrams in response to a loss of operability of the digital computer system
- monitoring of radiation and airborne radioactivity levels

7.2.1 Design-Basis Requirements

The primary design basis for the AFRRI Reactor is the safety limit on fuel temperature. To prevent exceeding the safety limit, design features, operating limitations, and automatic scrams are provided for over power conditions. Interlocks limit the magnitude of transient reactivity insertion.

7.2.1.1 Reactor Power Measurements

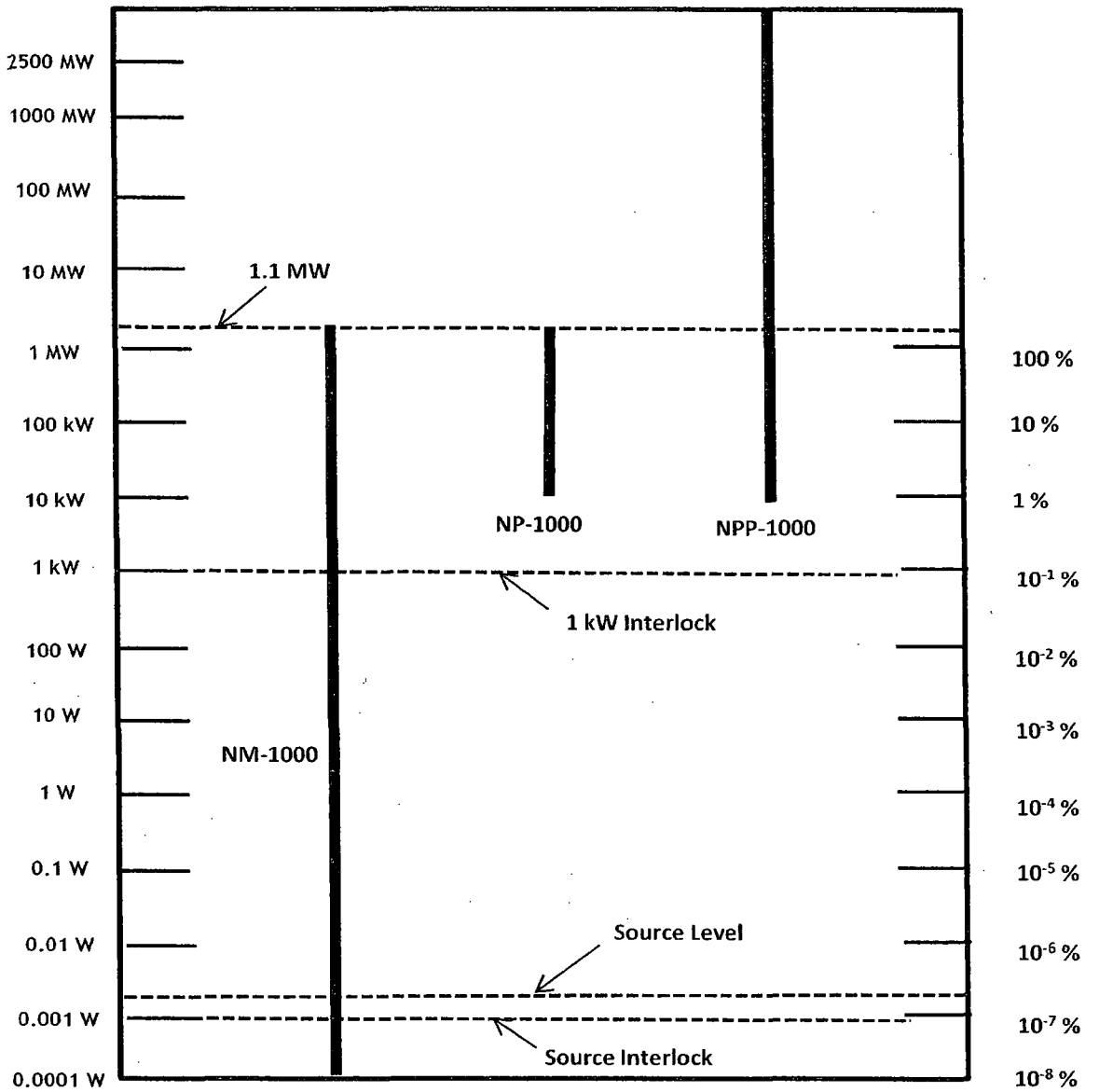
Reactor power is measured by three separate detectors; a fission chamber and two uncompensated ion chambers. The signal from the fission chamber is used by the NM-1000 to provide wide range log power from 10^{-8} % to 100 % reactor power and period indication from -30 seconds to +3 seconds. One uncompensated ion chamber is connected to the NP-1000 safety channel. A second uncompensated ion chamber is used by the NPP-1000 percent power and pulsing channel. Both the NP-1000 and NPP-1000 provide indication of linear reactor power from 0 % to 120 % steady state reactor power and the NPP-1000 also provides indication of reactor power for pulsing operations. Figure 7-1 shows the relative ranges of the channels and the detectors.

The fission chamber for the NM-1000 wide range instrument is connected to analog circuitry in a NEMA Preamplifier box mounted on the wall of the reactor room. This box contains a high voltage power supply, low voltage power supplies, preamplifier, campbelling module, counter/transmitter module and other circuitry. The high voltage power supply also monitors the high voltage to the fission chamber. If a loss of high voltage to the fission chamber is sensed, a bistable circuit will be tripped, resulting in a scram. The analog output from the preamplifier and campbelling module box described above is sent to the counter/transmitter which digitizes the signal and communicates this signal to a NEMA Microprocessor box that is also mounted on the wall of the reactor room. This box contains low voltage power supplies and microprocessor circuitry to convert the detector signal to useable digital values with software and transmit that signal to the DAC computer system. The Microprocessor box also contains circuitry for trip setpoints that provide interlock functions and indications on the console.

The NM-1000 log display provides a continuous indication from 10^{-8} % to 100 % of full power for the console display, analog bar graph display, and the console chart recorder.

The reactor period signal is generated by the microprocessor assembly of the NM-1000. Reactor period is displayed on the console display and analog bar graph display. A bistable circuit provides a visual warning and rod withdrawal interlock when the period is less than a predetermined limit. The period signal is also used by the AUTO control system.

The NP-1000 safety channel provides a linear power signal to the console display and analog bar graph display. These displays are scaled at 0 to 120 % of full power. A bistable circuit provides scram and alarm functions if the high power setpoint is exceeded. The detector input to the NP-1000 safety channel is disabled during pulse mode operations. A separate bistable circuit provides a scram signal to the reactor protection system upon a loss of detector high voltage.



NM-1000 – Fission Chamber
NP-1000 – Uncompensated Ion Chamber
NPP-1000 – Uncompensated Ion chamber

Figure 7-1 AFRRI Power Instrument Ranges

The NPP-1000 power pulsing channel functions as a second NP-1000 safety channel during steady state operations and during pulsing operations displays peak power from a pulse on the scale of 0 to 3300 MW on the analog bar graph and a scale of 0 to 3300 MW on the console display. An analog bar graph display of integrated energy is also provided with a scale of 0 to 30 MW-s. A graphical display of a pulse is available on the console display, along with text information on the pulse number, pulse time and date, full-width at half-maximum power, peak power, integrated power, minimum period, and peak fuel temperature. These data are recorded and may be stored and recalled at a later date. The pulsing channel is enabled when the pulse mode switch is pressed, as long as all interlock conditions are met. The pulse data collection is performed by the DAC computer and it begins when the pulse rod "Fire" button is depressed. This also enables the peak hold circuit and starts a one-minute timer. The peak power and energy displays are reset at the end of the one-minute period. The peak power is also recorded on the console data recorder.

The NPP-1000 channel contains bistable circuits that will produce a scram and alarm output for the conditions of the high power setpoint being exceeded and for high voltage is lost.

7.2.1.2 Temperature Measurements

As illustrated in Figure 7-2, fuel temperature may be measured by three thermocouples embedded in an instrumented fuel element. There are two fuel temperature channels in the reactor instrumentation system, so two thermocouples may be connected at one time. The two thermocouples may be from the same instrumented fuel element or from two different instrumented fuel elements. Fuel temperature is displayed on the console display and console analog bar graphs. A high temperature scram is sent to the reactor protective system for high power pulsing operations.

Temperature of the bulk pool water is measured by a resistance temperature detector (RTD) and a thermometer. The thermometer is a local readout device only. The RTD is mounted to the top of the reactor tank and the probe extends about 18 inches (45.72 cm) below the top of the tank. It sends a signal to the console for display as the pool water temperature. A temperature alarm circuit on the pool water channel will annunciate an audible and visual alarm on the console if the water temperature exceeds a preset temperature. Two additional RTDs are located in the primary piping, one on the inlet to the heat exchanger and one on the outlet of the heat exchanger. The temperature signals from these detectors are sent to the console for display as the pool water outlet temperature and the pool water inlet temperature. These primary piping RTDs may not display accurate temperatures for the primary cooling water if the primary pump is not operating.

21. TS 3.2.1: ANSI/ANS-15.1-2007, Section 3.2(1) provides guidance that the minimum number of operable control rods shall be specified for operations of the facility. Please indicate in TS 3.2.1 that the reactor shall not be operated unless the control rods specified in TS 5.2.2.b are operable or justify why this is not needed.

TS 3.2.1 has been modified to require that all four control rod drives be operable.

22. TS 3.2.2, Table 2: ANSI/ANS-15.1-2007, Section 3.2 provides guidance for the reactor control and safety systems. Table 2 in TS 3.2.2 specifies that a total of three Emergency Stop switches be provided, one at each exposure room and one in the main console. As stated in the TS, the purpose of these switches is to allow personnel in an exposure room or the reactor operator to stop actions through the interlock system. However, TS Table 2 appears to indicate that only one of the three switches is needed for reactor operation. It is not clear how the availability of only a single switch would be sufficient to satisfy the safety function of these switches. Please clarify.

TS 3.2.2 Table 2 lists the minimum required reactor safety system scrams. While there are three emergency stop buttons, they are all wired in series and connected to one input point of the reactor scram circuitry. There is only one emergency scram indication on the reactor console no matter which button is pushed, thus the table only lists one scram.

23. TS 3.2.2, Table 3: ANSI/ANS-15.1-2007, Section 3.2 provides guidance for the reactor control and safety systems. Table 3 in TS 3.2.2 identifies an interlock to prevent pulse initiation at power levels greater than 1 kW. Can this interlock be manually overridden? Is there a written procedure to instruct the operator in the function and operation of this interlock? Is there periodic surveillance on this interlock?

The interlock in TS3.2.2, Table 3 to prevent pulse initiation at power levels above 1kW is part of the proprietary reactor console software and cannot be manually overridden by the reactor operator. Since it cannot be overridden, there is no written procedure to instruct the operator in the function and operation of this interlock. This interlock is tested on each day of operation during the Daily Startup Checklist or before any operations lasting more than 24 hours by inserting a test signal into the console simulating a power level of 1kW and ensuring that the reactor cannot be placed in pulse mode.

24. TS 3.2.2, Table 3: ANSI/ANS-15.1-2007, Section 3.2 provides guidance for the reactor control and safety systems. Table 3 in TS 3.2.2 specifies the minimum reactor safety system interlocks that are required. However, three interlocks listed in the SAR are not included in Table 3, specifically the high voltage to fission detector, water temperature< 60 degrees C, and period > 3 s interlocks. Please propose the addition of these interlocks to the TSs or justify why these interlocks are not included in Table 3.

The interlocks for high voltage to the fission detector and period shorter than three seconds are added to TS 3.2.2, Table 3. The interlock for water temperature greater than 60°C applies only to the water temperature at the input to the demineralizer beds in the water purification system. This interlock is not related to the pool water temperature and thus is not related to any reactor safety system. The operator will reduce power to 5 kilowatts if the inlet temperature is greater than 60°C. Exposure of demineralizer resins to temperatures over 60°C renders them ineffective per the manufacturer's specifications. The reactor operator has a visual indication on the console that this interlock setpoint has been reached and that control rod withdrawal is no longer possible.

25. TS 3.5.2.b: 10 CFR 20.1101(d) specifies an ALARA constraint of 10-mrem TEDE for the public. Explain how the TS 3.5.2(b) exposure limit of 90 mRem complies with the 10-mrem ALARA constraint.

TS 3.5.2(b) is revised to indicate 9 mRem rather than 90 mRem.

26. TS 3.6.c: Regulatory Guide 2.2, Section C.1.c(3) states that the "materials of construction and fabrication and assembly techniques should be so specified and used that assurance is provided that no stress failure can occur at stresses twice those anticipated in the experiment." TS 3.6(c) allows explosive materials in quantities less than 25 mg to be irradiated in the reactor in a container "provided that the detonation pressure has been calculated and/or experimentally demonstrated to be less than the design pressure of the container." Please discuss how AFRRI will ensure a safety factor of two in TS 3.6(c) for this type experiment and other experiments. In addition, please provide the reference used to determine if materials are explosive materials.

TS 3.6.c is revised to limit the detonation pressure to no more than half the design pressure of the container, thereby ensuring the safety factor of two required by Regulatory Guide 2.2. For all experiments; containers, supports, and other experiment components are constructed of materials capable of withstanding at least twice the expected stresses of an experiment. If there is any doubt as to the structural stability or stress resistance of an item, that item or system is thoroughly tested before being placed into the reactor environment. The references used to determine if materials are explosive are the NIOSH Pocket Guide to Chemical Hazards, the 2008 Emergency Response Guidebook published by the U.S. Department of Transportation, the MSDS for the material, and similar reference materials.

27. TS 3.6.e: ANSI/ANS-15.1-2007, Section 3.8.1(1) guidance includes establishing a LCO for the maximum absolute value of the reactivity worth of individual experiments. TS 3.6(e) establishes a limit for the sum of all experiments in the reactor, but no limit is specified for a single experiment. Please propose and justify LCOs regarding the reactivity of individual experiments in the reactor, both secured and moveable or justify why such LCOs are not needed.

TS 3.6.e states that the sum of the absolute reactivity worths of all experiments in the reactor and in the associated experimental facilities shall not exceed $3.00 (2.1% Δk/k). The section has been revised to clarify that the absolute reactivity worth of a single secured experiment may also be $3.00 but the sum of the absolute reactivity worths of all moveable experiments shall be less than $1.00. This prevents pulsing of the reactor should all moveable experiments move at once. Failure, whether of one experiment or multiple experiments simultaneously, will not result in a reactivity insertion greater than $3.00, which is less than the authorized pulse magnitude of $3.50.

28. TS 4.1: ANSI/ANS-15.1-2007, Section 4.2(8) recommends that a thermal power calibration be done annually. Please clarify that the surveillance in TS 4.2.2.c includes a thermal power calibration.

TS 4.2.2.d has been added to clarify that a thermal power calibration is performed annually.

29. TS 4.1: ANSI/ANS-15.1-2007, Section 4.2(4) specifies surveillance for scram times of control rods. The surveillance requirement for control rod drop times is in TS 4.2.1. The 4.2.1 requires that control rod drop times by measured semiannually. This TS does not include "or after any work is done on rods or rod drive system" which is specified in the guidance. Please propose and justify TS to meet the guidance or justify why this is not needed.

TS 4.2.1 has been changed to require that the rod drop time be checked after any maintenance on the rod or the rod drive mechanical components.

30. TS 4.3: ANSI/ANS-15.1-2007, Section 4.3(4) recommends that the reactor coolant be analyzed for radioactivity at least annually, if necessary. Please propose and justify a specification to TS 4.3 for periodic analysis of the coolant for radioactivity or provide a basis for why such a surveillance specification should not be required.

The primary reactor pool water has historically been tested monthly for radioactivity by AFRRI health physics personnel using AFRRI Health Physics procedures for at least 30 years. No radioactivity above that expected in any TRIGA pool reactor has ever been detected. Any materials irradiated either in-core or in the pool are doubly encapsulated as required by TS. A specific surveillance requirement has been added as TS 4.3.c.

31. TS 4.4: ANSI/ANS-15.1-2007, Section 4.5 provides guidance for surveillances on ventilation system filter efficiency measurements and an operability check of any emergency exhaust systems. Please discuss whether the TS 4.4 is consistent with the standard guidance.

The term "emergency exhaust system" is not clearly defined in ANSI/ANS-15.1. Should an unplanned release of airborne radioactive particulate material occur in the reactor room, the air particulate monitors would automatically close the positive self-sealing dampers, thus preventing release of the radioactive material to the environment. The positive self-sealing dampers are inspected and tested at least monthly under TS 4.4 and they should be considered our "emergency exhaust system." This testing meets the recommendations of ANSI/ANS-15.1, Section 4.5(1). The integrity of the reactor ventilation system HEPA filters are verified whenever the filters are replaced. Manufacturer's literature states that HEPA filters do not have a finite shelf life. Filters are changed when the differential pressure increases to twice the initial pressure of the filters as recommended by the manufacturer. The differential pressure across the HEPA filters is recorded on the daily startup checklist by the reactor operator each day of reactor operation. The checklist also includes a requirement to notify the RFD or ROS if the pressure exceeds twice the initial pressure. The filters would then be changed and tested before reactor operations. Both of the recommendations of ANSI/ANS-15.1, Section 4.5 are covered by TS or reactor procedures.

32. TS 5.3: ANSI/ANS-15.1-2007, Section 5.4 provides guidance for the Keff in the storage of fissionable material; however, TS 5.3 does not specify a value. Please specify the Keff value to which the fuel storage racks are designed or justify why this is not needed.

All spare fuel elements stored in the reactor tank are stored under water in standard 12-element TRIGA fuel storage racks. This type of rack has been successfully used at TRIGA reactors for over 50 years without incident. AFRRI has not performed an analysis to determine the specific k-eff of a fully loaded storage rack. However, an analysis was performed in 1981 by an AFRRI nuclear engineer showing that, if a fully loaded rack should fail, the resulting configuration of the 12 fuel elements lying on the bottom of the pool could not reach criticality. In the worst-case configuration, the k-eff was shown to be no more than 0.746. This calculation was performed for the most reactive neutronic configuration possible. The standard two-level linear configuration of fuel elements in a TRIGA storage rack is certainly not the most reactive configuration possible and, thus, the k-eff of an intact fully loaded rack will be less than 0.746. A similar calculation was performed in 1966 by Fabian Fouchee of General Atomics, the designer of the TRIGA fuel storage racks (Attachments 2 and 3). A reference to k-eff <0.90 has been added to clarify the requirement.

file - reactor FE

SCIENTIFIC SUPPORT DEPARTMENT

MEMORANDUM FOR RECORD: 19 January 1981

SUBJECT: Nuclear Criticality Safety Analysis of Hypothetical AFRRI TRIGA Fuel Element Storage Rack Accidents

1. An analysis was performed to substantiate that a criticality excursion would not result in the unlikely event that a fully-loaded AFRRI fuel element storage rack were to fail.

2. For the purposes of analysis, it is conservatively assumed that when the storage rack fails, all twelve fuel elements contained in the rack escape and fall to the bottom of the pool. In addition, it is conservatively assumed that the twelve fuel elements come to rest at the bottom of the pool in the most reactive neutronic configuration possible. Moreover, it is conservatively assumed that the optimum configuration of fuel elements at the bottom of the reactor tank is fully reflected by water over a complete solid angle of 4π steradians even though only 2π steradian water reflection would actually exist.

3. Fuel elements used in the AFRRI reactor are standard stainless-steel clad TRIGA elements containing U-$ZrH_{1.7}$ with 8.5 weight percent uranium at a nominal U^{235} enrichment of 20 percent (See Figure 1). Each fuel element contains a nominal maximum [redacted] of U^{235}.

4. Figure 2, reproduced from TID-7028 (1), is based on experimental and analytical data and indicates that the minimum critical mass, $m_{crit.}$, for a heterogeneous, 20% enriched, fully water reflected U^{235} system in its most reactive configuration,* is [redacted] of U^{235}. Since our assumed twelve element configuration contains a total of (12 fuel elements) X ([redacted] U^{235}/fuel element) = [redacted] U^{235}, it would have a mass fraction critical, $m/m_{crit.}$, less than or equal to [redacted] U^{235}/[redacted] U^{235} or [redacted].

*For our assumed system, this conservative assumption not only takes into consideration an optimum reactive geometry but also neglects parasitic neutron capture in the stainless-steel clad, Sm-Al burnable poison wafers, etc. and assumes that the graphite end reflectors are replaced by water - a more effective neutron reflector.

ATTACHMENT 2 (5 pages)

Using: $k_{eff} = \sqrt[3]{m/m_{crit.}}$, (2) indicates that our assumed system would have a $k_{eff} \leq 0.746$. Therefore, even with the application of the most conservative assumptions, our assumed system would still not achieve criticality. In fact, if our assumed system had a $k_{eff} = 0.746$, then it would be subcritical by more than \$36.00 (assumes $\beta_{eff} = 0.007$).

Based on the minimum critical mass, $m_{crit.}$, value of [redacted] U^{235} obtained from Figure 2, and a U^{235} fuel loading per element of [redacted] U^{235}, a minimum of 29 AFRRI TRIGA fuel elements arranged in an optimum neutronic configuration would be required for a criticality excursion (≈\$1.09) to occur.

5. Verification of the conservatism of this analysis is provided by data in RSD 5-8[3]. That is, experience has shown that, during actual AFRRI core loading, ~ 69 stainless-steel TRIGA fuel elements ([redacted] U-235) are required to achieve criticality. Therefore, since the AFRRI core lattice arrangement is very close to the optimal neutronic geometry for TRIGA fuel elements, the results of this criticality analysis are conservative by a factor of ~ 2.4 on a fuel element as well as a U-235 mass basis for criticality.

6. In summary, a hypothetical AFRRI fuel element storage rack failure is analyzed from a nuclear criticality safety standpoint. Conservative assumptions are applied wherever possible; yet k_{eff} and $m/m_{crit.}$ for the system are found to be no greater than 0.746 and 0.415, respectively. As a result, there is no possibility of a criticality excursion in the unlikely event that a fully-loaded fuel storage rack were to fail in the AFRRI TRIGA reactor facility.

3 Encls
1. Fig. 1
2. Fig. 2
3. References

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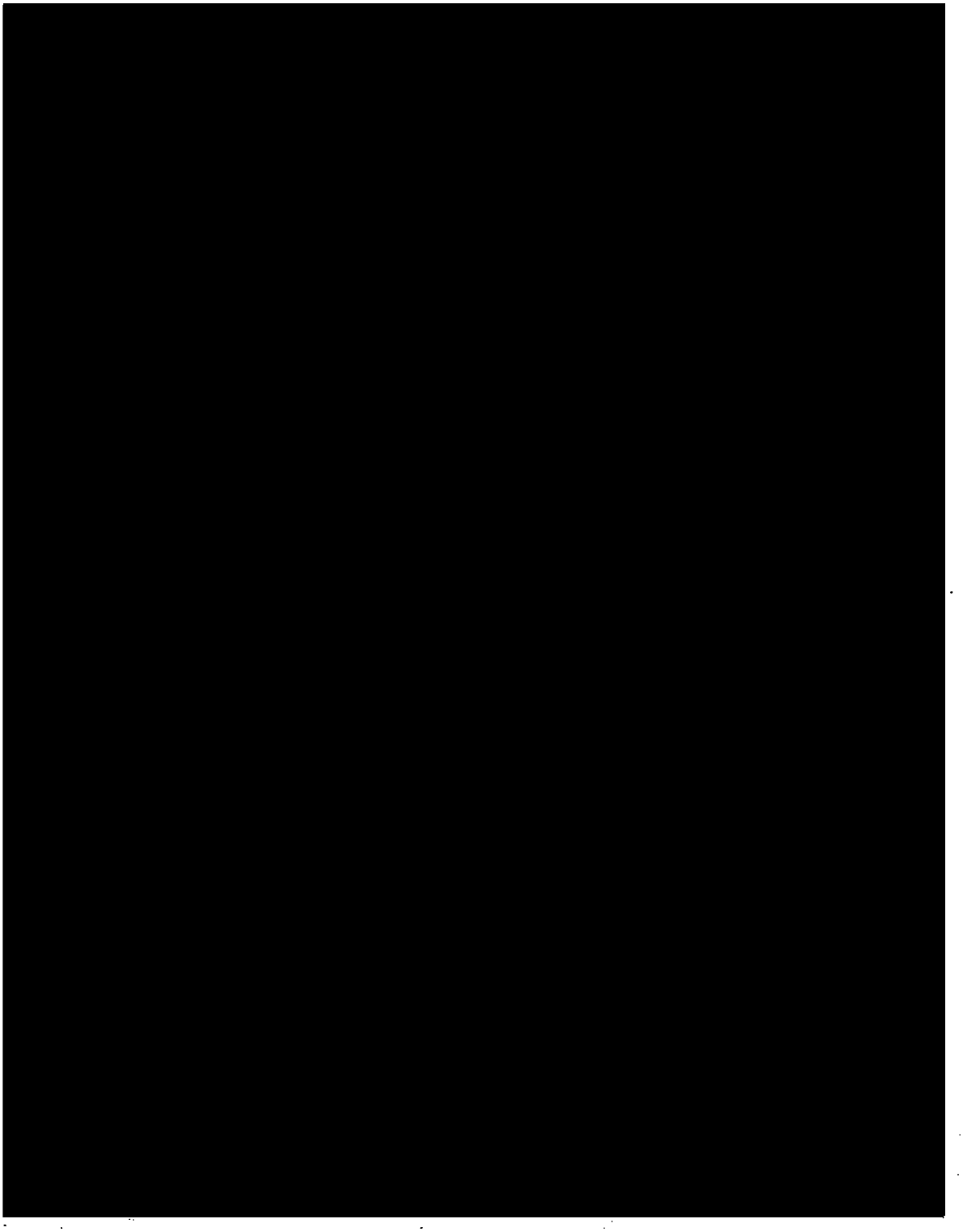
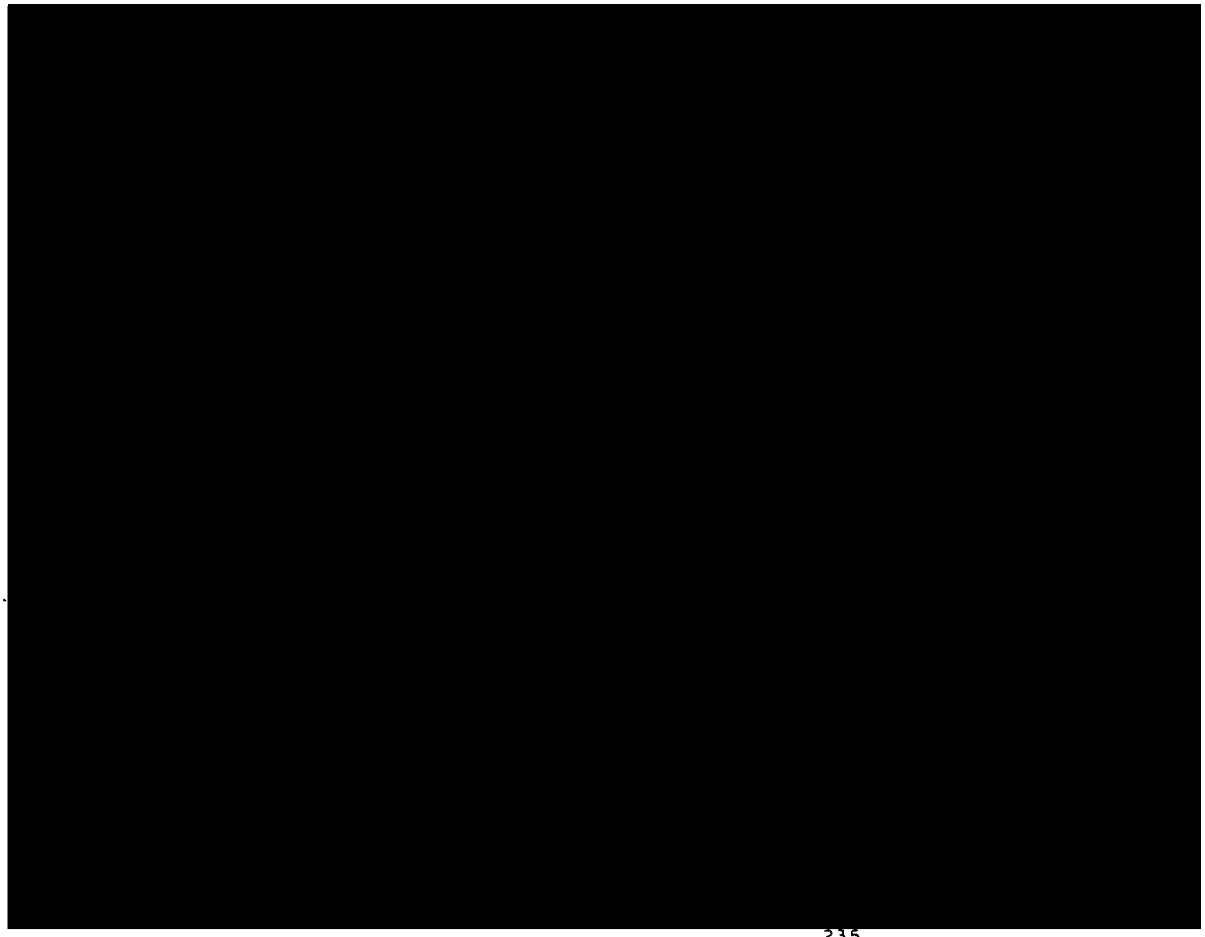


Figure 1. Standard AFRRI TRIGA Fuel Element

Encl 1



URANIUM ENRICHMENT, W.T% U^{235}

Fig. 2. Minimum critical mass as a function of U^{235} enrichment in hydrogen-moderated systems

Encl 2

REFERENCES

1. Paxton, H.C., Thomas, J.T., Callihan, D., and Johnson, E.B., Critical Dimensions of Systems Containing U^{235}, Pu^{239}, and U^{233}, TID-7028, Los Alamos Scientific Laboratory and Oak Ridge National Laboratory, Oak Ridge, TN, June 1964.

2. O'Dell, R.D. (editor), Nuclear Criticality Safety, compendium of information presented at the Biannual Nuclear Criticality Safety Short Course in Taos, NM by the University of New Mexico, May 1973, published by Technical Information Center, Office of Information Services, U.S. Atomic Energy Commission, Washington, D.C., 1973.

3. Radiation Sources Division Instruction, RSD 5-8, AFRRI/SSRS.

GENERAL DYNAMICS / GENERAL ATOMIC DIVISION

To: Distribution

Date: March 1, 1966

From: Fabian C. Foushee FCF

Subject: Storage of TRIGA Fuel Elements

Introduction:

In GA-5402, "Criticality Safeguards Guide", general limits on the various dimensions, and concentrations of fissile material that must be imposed on any unit of such material (see p. 18, Table 1) are set forth. A further, very general limitation on the storage of well moderated U-235 is given (on p. 22) as an average of [redacted] of U-235 per square foot of aspect area. This latter limitation has been cited as evidence that the TRIGA fuel storage racks located in the reactor pool are indeed safe. In these racks, 10 TRIGA fuel elements are arranged in line with 2 inches separating the axial centerline of adjacent elements. Assuming, at the very most, [redacted]s of U-235 per element, the concentration in the array is [redacted]s of U-235/ft^2. This storage system, then, is certainly safe as the concentration for this finite array is only 2/3 of the recommended maximum safe value for an array that extends to infinity in two dimensions.

Recently, it has been suggested that it would be more appropriate to calculate the multiplication of a system containing fissile material as this would be a more meaningful measure of the safety of the system. Consequently, there have been completed some GAZE calculations of the TRIGA in-pool fuel storage racks which do, indeed, show a considerable margin of safety.

In the GAZE calculations to be described a set of six group (3 fast and 3 thermal) cross-sections derived for an 8.0 wt.-% U-ZrH (20% enriched U) system was used. These cross-sections were calculated for an homogeneous system with volume fractions of the various component materials as in TRIGA. The results of the calculations are consequently conservative as lumping the fuel (as in a fuel element) will lower k_{eff} because of self-shielding. The last statement must be qualified by noting that lumping the fuel has an advantageous effect in that the effective U-238 resonance integral is reduced. To examine the effect of this parameter on the multiplication of the system, two GAZE calculations were performed that were identical except one used U-238 cross-sections appropriate to an infinite dilution. From these calculations one could infer that using an infinite dilution U-238 resonance integral decreases k_{eff} by about 1.5%.

TRIGA Spent Fuel Storage

TRIGA fuel elements are stored in the reactor pool in racks fastened to the side of the pool. Each storage rack accommodates ten elements, the axial centerlines of which are two inches apart and lie in a single

ATTACHMENT 3 (2 pages)

plane. The homogeneous model employed to describe the storage rack was an infinite plane 1.47 inches thick (the diameter of a fuel element) with 42.3% of the volume occupied by water (corresponding to the two inch separation distance). The nuclear densities of the constituents are given in Table I.

Table I

Composition of TRIGA Fuel Storage Rack System

Constituent	Nuclear Density x 10^{-24}
H-in $ZrH_{1.65}$	3.088×10^{-2} nuclei/cm^3
H-in H_2O	2.614×10^{-2}
Oxy	1.307×10^{-2}
Zr	1.939×10^{-2}
U-235	[redacted]
U-238	[redacted]
Stainless Steel	2.64×10^{-3}

Two calculations were performed, one for a single rack and one for two racks back-to-back. The two rack calculation assumed that the fuel elements touche (i.e., the mid-plane to mid-plane separation distance between adjoining racks was only 1.47 inches). Actually, the fuel storage racks, as presently construc ed, cannot come closer together than about 2.5" center-to-center. Thus, ther would be about 1 inch of water between racks providing significant de-coupling. The results of these calculations are shown in Table II.

Table II

	k_{eff}
Plane array one element thick	0.5096
Plane array two elements thick	0.7227

It should be noted that these calculations were made for 8 wt.-% uranium. Increasing the uranium content to 8.5 wt.-%, as in the Torrey Pines TRIGA Mark III, would increase k_{eff} by 2 to 3%, or to .53 and .74 for the one and two element arrays of Table II. These heavier elements contain something less than [redacted] of U-235 each. In the one element storage array there results a concentration of about [redacted] of U-235 per square foot of aspect; in the two element thick storage array, about [redacted] per square foot. Therefore one can conclude that whereas the one element thick storage array provides an areal concentration below the maximum prescribed (i.e., [redacted] - U-235/ft^2), the two element thick array results in a k_{eff} that is less than 0.8, a value found acceptable to the AEC Division of Licenses and Regulations, in the TRIGA Mark III Technical Specifications.

33. TS 6.0: 10 CFR 20.1101(a) requires that each licensee develop, document, and implement a radiation protection program. NUREG-1537, Chapter 12.1 states that the organization should meet the non-power reactor standard ANSI/ANS-15.1-2007. ANSI/ANS-15.1-2007, Section 6.3 states that the facility shall implement a radiation protection program in accordance with the guidelines in ANSI/ANS-15.11. Please discuss whether the TS meets the criteria in 10 CFR 20.1101(a)-(c) and ANSI/ANS-15.1-2007, Section 6.3.

AFRRI has developed and implemented a radiation protection program that meets the requirements of 10 CFR 20.1101(a)-(c), ANSI/ANS-15.1, and ANSI/ANS-15.11. As shown in TS 6.1.1, the AFRRI Radiation Safety Officer (RSO) is assigned responsibility for implementing the program at the reactor and throughout AFRRI. The RSO advises the reactor facility director (Level 2) and reports to the AFRRI Director (Level 1) for all matters relating to safety. All operations at the reactor are performed under the AFRRI ALARA program developed by the RSO. Written radiation protection procedures are developed as required by TS 6.3 and the program is periodically reviewed under the auspices of the radiation safety committee. The RSO's specific responsibilities are now indicated in TS 6.1.2.

34. TS 6.1.3.2.b: ANSI/ANS-15.1-2007, Section 6.1.3.3 provides guidance for events requiring the presence at the facility of the senior reactor operator. TS 6.1.3(b) only specifies "maintenance activities that could affect the reactivity of the reactor." The other items listed in ANSI/ANS-15.1-2007, Section 6.1.3.3 that require Senior Reactor Operator supervision, such as (1) initial startup and approach to power, (2) recovery from unplanned or unscheduled shutdown or significant power reduction, and (3) refueling should be specified in order to comply with the requirements of 10 CFR 50.54(m)(1).

TS 6.1.3.2(b) has been revised to include the recommended items.

35. TS 6.2.3.3: ANSI/ANS-15.1-2007, Section 6.2.2(2) provides guidance for establishing a quorum for a meeting of the Reactor and Radiation Facility Safety Committee (RRFSC). Please explain how the RRFSC quorum satisfies the ANSI 6.2.2(2) criterion that quorums are "not less than one-half of the voting membership where the operating staff...does not constitute a majority."

Under TS 6.2.1.1, the Reactor Facility Director is the only regular member who is a member of the "operating staff." Theoretically, other members of the operating staff could be appointed as special voting members. But since under TS 6.2.3.3 a majority of those present and voting must be regular members, the operating staff can never constitute a majority. Under TS 6.2.3.3, four specific people must be present for a quorum. Depending on the number of outside regular members under TS 6.2.1.1.a(2)(b), the total regular membership could be 4-6 people. In either case, the four required people would be not less than one-half of the voting membership.
In rare cases, special voting members may be appointed under TS 6.2.1.1.b. Since, as discussed earlier, a majority of those present and voting must be at least the four specific regular members, no more than three special members can be appointed for any one meeting. Here also, the four required people would be not less than one-half of the voting membership. The TS as written satisfy the ANSI/ANS-15.1 requirements.

36. TS 6.3: ANSI/ANS-15.1-2007, Section 6.4 provides guidance requiring written procedures be approved by specified personnel prior to initiating activities listed in said Section. TS 6.3 states that the Reactor Facility Director can make substantive changes to procedures and the Reactor Operations Supervisor can make temporary changes to procedures. Please clarify that both types of changes are documented and subsequently reviewed with respect to 10 CFR 50.59 by the RRFSC.

Both substantive changes and temporary changes are documented, reviewed with respect to 10 CFR 50.59, and reviewed by the RRFSC at their next scheduled meeting. Both substantive and temporary changes are first reviewed with respect to 10 CFR 50.59. Each substantive procedure change includes a cover sheet on which each licensed operator and operator trainee signs that he/she has reviewed and understands the change. Only then is the new procedure implemented with the cover sheet attached. For temporary changes, the procedure page(s) affected are copied. The change is then hand-written on the copied page(s) and initialed by either the RFD or the ROS. The change would be posted in a conspicuous place at the console to ensure operator familiarity. At the end of the temporary period, the change would be placed in the historical reactor files. In both types of changes, the documentation would be reviewed by the RRFSC at their next scheduled meeting.

37. TS 6.5.2: ANSI/ANS-15.1-2007, Section 6.7.2.1 provides guidance on licensee actions following a reportable occurrence. It is not specified in the TS who can authorize restart of the reactor after a reportable occurrence where the reactor was shut down.

TS 6.5.2 has been revised to clarify that the reactor facility director must authorize restart of the reactor after a reportable occurrence.

38. TS 6.5.2: ANSI/ANS-15.1-2007, Section 6.7.2.1 provides guidance for a follow-up written report following a reportable occurrence. TS 6.5.2 should specify the 14-day time frame for the follow-up written report.

The 14-day time frame has been added to TS 6.5.2.

39. TS 6.6: ANSI/ANS-15.1-2007, Section 6.7.2.2 provides guidance for the requirement for a written report within 30 days to the NRC for permanent changes in the AFRRI organization involving Level I or II personnel and significant changes in transient or accident analyses as described in the SAR. This requirement does not appear in TS 6.6.

The requirements have been added at the end of TS 6.6.

40. TS 6.7.3: ANSI/ANS-15.1-2007, Section 6.8.3 provides guidance for the requirement to retain records for the lifetime of the facility. Please add to records retained for the life of the facility the following: "Reviews and reports pertaining to a violation of a Safety Limit, Limiting Safety System Setting, or an LCO."

The requested item has been added to TS 6.7.3.

41. 10 CFR 55.59(c)(5)(i) requires that the facility licensee shall retain operator requalification documentation records until the operator's license is renewed. In addition, Section 6.8.2 of ANSI/ANS-15.1-2007 contains the requirement that training records for reactor operators be maintained at all times the individual is employed or until the certification is renewed. TS 7.6.2 specifies that operator requalification records are maintained for a training cycle, which usually does not coincide with the operator license renewal cycle. Please discuss whether TS 6.7.2 meets the criteria in 10 CFR 55.59(c)(5)(i) and ANSI/ANS-15.1-2007, Section 6.8.2.

TS 6.7.2 has been revised to require that requalification records be maintained until the license is renewed or cancelled, whichever occurs first.