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AMENDMENT # 17 TO THE TECHNICAL SPECIFICATIONS

FACILITY LICENSE NO. R-84

FOR THE

AFRRI TRIGA MARK F REACTOR

DOCKET NO. 50-170

8009030452

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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. Reference NRC Regulatory Guide 1.16 and ANSI N378-1974.

I. DEFINITIONS

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1.0 DEFINITIONS

REACTOR OPERATING CONDITIONS

1.1 REACTOR SHUTDOWN

The reactor is shut down when the reactor is subcritical by at least $1.00 of reactivity.

1.2 REACTOR SECURED

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

a. The reactor is shut down.

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

c. No work is in progress involving in-core fuel handling or refueling operations, maintenance of the reactor or its control mechanisms, or insertion or withdrawal of in-core experiments, unless sufficient fuel is removed to insure a $1.00 shutdown margin with the most reactive control rod removed.

1.3 REACTOR OPERATION

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

1.4 COLD CRITICAL

The reactor is in a cold critical condition when it is critical at a power level less than 100 watts.

1.5 STEADY STATE MODE

Operation in the steady state mode shall mean steady state operation of the reactor either by manual operation of the control rods or by automatic operation of one control rod (servo-control) at power levels up to 1 MW, utilizing the appropriate scrams in Table I and the appropriate interlocks in Table II.

1.6 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 I and the appropriate interlocks in Table II. The reactor may be pulsed from a critical or subcritical state.

1.7 SHUTDOWN MARGIN

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

1.8 ABNORMAL OCCURRENCE

An "Abnormal Occurrence" is defined for the purposes of the reporting requirements of Section 208 of the Energy Reorganization Act of 1974 (P.L. 93-438) as an unscheduled incident or event which the Nuclear Regulatory Commission determines is significant from the standpoint of public health or safety.

1.9 REPORTABLE OCCURRENCE

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

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

b. Operation in violation of a Limiting Condition for Operation.

c. Failure of a required reactor or experiment safety system component which could render the system incapable of performing its intended safety function.

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

e. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of a condition which could result in operation of the reactor outside the specified safety limits.

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

1.10 EXPERIMENT

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

1.11 EXPERIMENTAL FACILITIES

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

a. Exposure Room #1

b. Exposure Room #2

NOTE: Exposure room protective barriers shall be differentiated from the primary protective barrier (pool water) for purposes of placement of experiments within these barriers.

c. Pneumatic Transfer System

d. Reactor Pool

e. Portable Beam Tube

f. In-core experiment tubes.

1.12 STANDARD CONTROL ROD

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

1.13 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.

1.14 FUEL ELEMENT

A fuel element is a single TRIGA fuel rod.

1.15 CORE GRID POSITION

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

1.16 INSTRUMENTED ELEMENT

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

1.17 SAFETY LIMIT

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

1.18 LIMITING SAFETY SYSTEM SETTING

Limiting safety systems setting is setting for automatic protective devices related to those variables having significant safety functions.

1.19 OPERABLE

A system, device, or component shall be considered operable when it is capable of performing its intended functions in a normal manner.

1.20 REACTOR SAFETY SYSTEMS

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

1.21 MEASURED VALUE

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

1.22 MEASURING CHANNEL

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

1.23 SAFETY CHANNEL

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

1.24 CHANNEL CHECK

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

1.25 CHANNEL TEST

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

1.26 CHANNEL CALIBRATION

A channel calibration consists of using a known signal to verify or adjust a channel to produce an output that responds within limits specified by the channel manufacturer.

1.27 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 whereabouts, and telephone number; and

c. the individual is capable of getting to the reactor facility within thirty (30) minutes under normal circumstances.

1.28 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.

II. SAFETY LIMIT AND LIMITING CONDITIONS FOR OPERATIONS

II. SAFETY LIMIT AND LIMITING CONDITIONS FOR OPERATIONS

1.0 SAFETY LIMIT

1.1 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 build-up 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, including 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 ultimate stress provided that the temperature of the fuel does not exceed 1000°C and the fuel cladding is water cooled.

2.0 LIMITING SAFETY SYSTEM SETTINGS

2.1 FUEL TEMPERATURE

Applicability

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

Objective

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

Specification

The limiting safety system setting for fuel temperature shall not exceed 600°C. There will be two fuel temperature safety channels. One channel will 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 400°C for standard TRIGA fuel elements.

During steady state operation, the flux may peak in the "C" ring and during pulse operation, the flux may peak in the "B" ring. Flux, hence fuel temperature varies across the core depending on the rod configuration, power level, pulse size, or placement of the core within the reactor pool. By utilizing one element from each ring a worst case indication is reasonably assured.

2.2 STEADY STATE OPERATION

Applicability

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

Objective

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

Specification

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

Basis

Thermal and hydraulic calculations indicate that TRIGA fuel may be safely operated up to power levels of at least 2.0 megawatts with natural convective cooling.

2.3 PULSE MODE OPERATION

Applicability

This specification applies to the peak power achieved in the reactor as a result of a pulse insertion of reactivity.

Objective

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

Specification

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

Basis

Based upon the Fuchs-Nordheim mathematical model of the AFRRI-TRIGA reactor, an insertion of 2.8% $\Delta k/k$ results in a maximum average fuel temperature of less than 550°C thereby staying within the limiting safety setting which protect the safety limit. The 50°C margin to the Limiting Safety System Setting and the 450°C margin to the safety limit amply allows for uncertainties due to extrapolation of measured data, accuracy of measured data, and location of instrumentated fuel elements in the core.

3.0 LIMITING CONDITIONS FOR OPERATIONS

3.1 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 to guarantee that the fuel temperature safety limit will not be exceeded.

Specifications

a. The maximum available excess reactivity above cold critical with or without all experiments in place shall be \$5.00 (3.5% $\Delta k/k$).

b. The minimum shutdown margin provided by the remaining control rods with the most reactive control rod fully withdrawn or removed shall be \$1.00 (0.70% $\Delta k/k$).

Basis

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

3.2 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 prompt shutdown of the reactor to prevent fuel damage.

Specification

The time from receipt of scram signal to the control rod drive in a full up position to the time for the control rod to be fully inserted will be less than one 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 anticipated for a TRIGA reactor the specified scram time is adequate to assure the safety of the reactor.

3.3 REACTOR CONTROL SYSTEM

Applicability

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

Objective

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

Specification

The reactor shall not be operated unless the measuring channels listed in the following table are operable.

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

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

Basis

Fuel temperature displayed at the control console given continuous information on this parameter which has a specified safety limit. The power level monitors assure that the reactor power level is adequately monitored for both steady state and pulsing modes of operation. The specifications on reactor power level indication are included in this section since the power level is related to the fuel temperature.

3.4 REACTOR SAFETY SYSTEM

Applicability

This specification applies to the reactor safety system channels.

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 I and II are operable.

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 chambers, operation of the reactor without adequate instrumentation is prevented. The preset timer insures that the reactor power level will reduce to a low level after pulsing.

TABLE I - MINIMUM REACTOR SAFETY SYSTEM SCRAMS

	Channel	Set Point	Minimum Number of Channels Operable in Effective Mode: Steady State	Pulse
1.	Percent Power (High Flux)	1.1 MW	2	1
2.	Fuel Temperature	$600^{\circ}C$ (Max)	2	2
3.	Emergency Stop	Closure switch in each exposure room, one on console	1	1
4.	Pool Water Level	15 Feet or greater above pool floor	1	1
5.	Loss of High Voltage to safety channels	0 to 20% Voltage Loss	2	1
6.	Scram Bar on Console	Manual Operated	1	1
7.	Timer for Pulse Duration	0 to 15 Seconds	0	1

TABLE II - MINIMUM REACTOR SAFETY SYSTEM INTERLOCKS

	Action Prevented	Mode in Which Effective: Steady State	Mode in Which Effective: Pulse
1.	Simultaneous manual withdrawal of two standard control rods.	X	
2.	Withdrawal of any control element except transient rod.		X
3.	Pulse initiation at power levels greater than 1 kW.		X
4.	Prevent any rod from being withdrawn unless the operational channels see a count rate greater than .5 cps.	X	X

Basis

The interlock preventing the initiation of a pulse at a critical level above 1 kW assures the pulse magnitude will not allow the fuel element temperature to approach the safety limit. The interlock preventing movement of standard control rods in pulse mode will prevent the inadvertent placing the reactor on a positive period while in pulse mode. Requiring a count rate to be seen by the operational channels insures sufficient source neutrons to bring the reactor critical under controlled conditions.

3.5 FACILITY INTERLOCK SYSTEM

Applicability

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

Objective

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

Specification

Facility interlocks shall be provided such that:

a. The reactor cannot be operated unless the lead shielding 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 shielding doors are fully closed; or if the lead shielding doors are fully opened, both exposure room plug doors must be fully closed.

c. The lead shield doors cannot be opened to allow movement to an exposure room unless a warning horn has sounded in that exposure room or unless two licensed operators have visually inspected the room to insure that prior to securing the plug door that no condition exists that is hazardous to personnel.

Basis

These interlocks prevent the operation and movement of the reactor core in an area (exposure room) until there is assurance that inadvertent exposures will be eliminated.

3.6 RADIATION MONITORING SYSTEM

Applicability

This specification describes 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 describe the radiation monitoring equipment and radiation information that is available to the operator to assure safe operation of the reactor.

Specification

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

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

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

c. Air Particulate Monitor. The air particulate monitor (APM) will sample the air above the reactor pool. This unit shall be sensitive to particulate matter from decayed fission products.

d. The following will delineate the alarm and readout system for the above monitors:

MONITOR	ALARM (A = Audible, V = Visual) LOCATION	READOUT LOCATION
ARM		
1. Reactor Room	A & V - Control Room	Control Room
2. Reactor Room	V - Control Room	Control Room
3. Exposure Area 1	V - Control Room	Control Room
4. Expsosure Area 2	V - Control Room	Control Room
GSM	V - Control Room	Control Room
APM	A & V - Control Room	Control Room

Basis

This system is intended to characterize the normal operational radiological environment of the facility and aid in evaluating any abnormal operations conditions. The radiation monitors provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the 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.

3.7 POOL BULK WATER TEMPERATURE AND PURITY

Applicability

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

Objective

To insure the effectiveness of the resins in the water purification system and to prevent activated contaminants from becoming a biological hazard.

Specifications

a. The reactor will not be operated above a thermal power of 5 kW when the pool bulk water temperature exceeds $60^{\circ}C$.

b. The conductivity of the water shall be less than 2 micromhos ($.5 \times 10^{6}$ Ohms resistance) at the output of the purification system.

Basis

Manufacturer's data states that the resins in the water purification system break down with sustained operation in excess of $60^{\circ}C$. The 2 micromohs is a reasonable level of water contaminants in a system of this nature. Based on experience activation at this level is not significant.

3.8 VENTILATION SYSTEM

Applicability

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

Objective

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

Specification

The reactor shall not be operated unless the facility ventilation system is operable except for periods of time necessary to test or permit repair of the system. In the event of a substantial release of airborne radioactivity, the ventilation system will be secured automatically by a signal from the reactor deck continuous air monitor.

Basis

During normal operation of the ventilation system, the concentration of Argon-41 in unrestricted areas is below the MPC. In the event of a substantial release of airborne radioactivity, the ventilation system will be secured automatically. Therefore, operation of the reactor with the ventilation system shut down for short periods of time to test or make repairs insures 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.

3.9 LIMITATIONS ON EXPERIMENTS

Applicability

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

Objective

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

Specifications

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

a. If a 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 II, Appendix B, of 10 CFR 20, assuming 100% of the gases or aerosols escape.

b. Each fueled experiment shall be limited such that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 1.5 curies and the maximum Sr-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 to be less than the design pressure of the container.

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

e. The sum of the absolute reactivity worths of all experiments in the reactor and in the associated experimental facilities shall not exceed \$3.00 (2.1% delta $\Delta k/k$). This includes the total potential reactivity insertion which might result from experiment malfunction, accidental experiment flooding or voiding and accidental removal or insertion of experiments.

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

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

g. If a capsule fails and releases materials which 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 Physicist-in-Charge and determined to be satisfactory before operation of the reactor is resumed.

h. All experiments placed in the reactor environment will be observed by a member of the staff for mechanical stability to insure unintended movement will not cause an unplanned reactivity change or physical damage.

Bases

a. This specification is intended to reduce the likelihood that airborne activities in excess of the limits of Appendix B of 10 CFR 20 will be released to the atmosphere outside the facility boundary.

b. The 1.5-curie limitation on iodine 131 through 135 assures that in the event of malfunction of a fueled experiment leading to total release of the iodine, the exposure dose at the exclusion area boundary will be less than that allowed by 10 CFR 20 for an unrestricted area.

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

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

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

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

g. Operation of the reactor with the reactor fuel or structure damaged is prohibited to avoid release of fission products.

3.10 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.

Specifications

Any additions or modifications to the ventilation system, the core and its associated support structure, the pool, 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 Committee. A system shall not be considered operable until after it is successfully tested.

Basis

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

3.11 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 insure that the health and safety of the public is not endangered by the discharge of Argon-41 from the TRIGA reactor facility.

Specification

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

b. If a dosimeter reading for any environmental monitoring station indicates that an exposure of 400 mr above background has been reached during the year, reactor operations which generate and release measurable quantities of Argon-41 will be curtailed to 2 MW/hr per month for the remainder of the year.

c. If dosimeter reading for any environmental monitoring station indicates that an exposure of 500 mr above background has been reached during the year, reactor operations which generate and release measurable quantities of Argon-41 will be ceased for the remainder of the year.

Basis

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 immersion conditions cumulative exposure is the more appropriate quantification of this limit than MPC values of 10 CFR.

III. SURVEILLANCE REQUIREMENTS

III. SURVEILLANCE REQUIREMENTS

1.0 SAFETY LIMIT - FUEL ELEMENT TEMPERATURE

Applicability

This specification applies to the surveillance requirements of the fuel element temperature measuring channel.

Objective

The objective is to assure that the fuel element temperatures are properly monitored.

Specifications

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

b. A calibration of the temperature measuring channels shall be performed annually but at intervals not to exceed 14 months.

Basis

Operational experience with the TRIGA system gives assurance that the thermocouple measurements of fuel element temperatures have been sufficiently reliable to assure accurate indication of this parameter.

2.0 LIMITING SAFETY SYSTEM SETTING

2.1 FUEL TEMPERATURE

Applicability

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

Objective

To insure operability of the fuel measuring channels.

Specifications

a. A check of the fuel temperature scram shall be made daily whenever the reactor is operated.

b. A calibration of the temperature measuring channel shall be made annually (to 14 months) or upon replacement of any channel component.

c. A weekly channel test will be performed on fuel measuring channels.

Basis

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

2.2 STEADY STATE AND PULSE OPERATIONS

Applicability

These specifications apply to the surveillance requirements for the high flux safety channels.

Objective

To insure the operability of the high flux safety channels.

Specifications

a. A scram check of the high flux safety channels will be made daily whenever the reactor is to be operated.

b. A channel calibration will be performed annually (not to exceed 14 months) or upon replacement of any channel component.

c. A channel test will be performed weekly whenever the reactor is operated.

Basis

TRIGA system components have operational proven reliability. Daily checks insure accurate scram functions. Weekly channel testing is sufficient to insure detection of possible channel drift.

3.0 LIMITING CONDITIONS FOR OPERATIONS

3.1 REACTIVITY LIMITATIONS

Applicability

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

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 14 months.

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

c. The control rods shall be visually inspected for deterioration annually, not to exceed 14 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. Semi-annually, at intervals not to exceed eight 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 envolving the movement of control rods or prior to each continuous operation extending more than a day.

f. The power coefficient of reactivity at 100 kW and 1 MW will be measured annually not to exceed 14 months.

Basis

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 insure that no significant changes in the shutdown margin have occurred. The visual inspection of the control rods is made to evaluate corrosion and wear characteristics caused by operation in the reactor.

3.2 SCRAM TIME AND REACTOR CONTROL AND SAFETY SYSTEMS

Applicability

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

Objective

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

Specifications

a. The control rod drop times shall be measured semiannually but at intervals not to exceed 8 months.

b. A Channel Test of each of the reactor safety system channels for the intended mode of operation shall be performed prior to each days operation or prior to each operation extending more than one day, except for the pool water level channel which shall be tested weekly.

c. A Channel Calibration shall be made of the power level monitoring channels by the calorimetric method annually but at intervals not to exceed 14 months.

Bases

Measurement of the scram time on a semiannual basis is a verification of the scram system and is an indication of the capability of the control rods to perform properly. The channel tests will assure that the safety system channels are operable on a daily basis or prior to an extended run. The power level channel calibration will assure that the reactor will be operated at the proper power levels. Transient control and checks and semiannual maintenance insure proper operation of this control rod.

3.3 FACILITY INTERLOCK SYSTEM

Applicability

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

Objective

To insure performance and operability of the facility interlock system.

Specification

Functional checks will be made annually but not to exceed 14 months to insure the following:

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

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

c. The warning horn will sound in the exposure room prior to opening the lead shield door allowing the core to move to that exposure room.

Basis

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

3.4 RADIATION MONITORING SYSTEM

Applicability

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

Objective

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

Specification

The area radiation monitoring system and the continuous air monitoring system shall be calibrated quarterly but at intervals not to exceed 5 months and shall be verified to be operable by channel check daily when reactor is in operation.

Basis

Experience has shown that quarterly verification of area radiation and air monitoring system set points in conjunction with quarterly calibration is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span.

3.5 POOL WATER

Applicability

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

Objective

The objective is to assure the integrity of the water purification system, maintaining the purity of the reactor pool water, thereby eliminating possible radiation hazards from activated impurities in the water system.

Specification

a. The pool water temperature as measured near the input to the water purification system will be measured daily.

b. The conductivity of the water shall be less than 2 μmohs at the output of the purification system, and shall be measured weekly.

Basis

Manufacturer specifications on the mixed bed demineralizer state a resin breakdown occurs at water temperature greater than 60°C. the 2 μmohs is a reasonable level of water contaminants in a system of this nature. Based on experience activation at these levels is practically nonexistent.

3.6 VENTILATION SYSTEM

Applicability

This specification applies to the building containment ventilation system.

Objective

The objective is to assure the proper operation of the ventilation system in controling releases of radioactive material into the uncontrolled environment.

Specification

It shall be verified monthly by a visual check that the ventilation sytem is operable.

Basis

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

3.7 LIMITATION ON EXPERIMENTS

The surveillance requirements on experiments are self-explanatory and contained in the SPECIFICATION II, 3.9.

3.8 SYSTEM MODIFICATION

Surveillance requirements are self-explanatory and contained in SPECIFICATION II, 3.10.

4.0 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.

Specifications

All fuel elements shall be inspected visually for damage or deterioration and measured for length and bow at intervals separated by not more than 500 pulses of insertion greater than $2.00 or annually (not to exceed 14 months), which ever occurs first. Fuel elements indication an elongation greater than 1/10 inch, a lateral bending greater than 1/16 inch, or significant visible damage shall be considered to be damaged and shall not be used in the reactor core.

Bases

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

The limit of tranverse bend has been shown to result in no difficulty in disassembling the core. Analysis of the removal of heat from touching fuel elements shows that there will be no hot spots resulting in damage to the fuel caused by this touching. Experience with TRIGA reactors has shown that fuel element bowing that could result in touching has occured without deleterious effects. The elongation limit has been specified to assure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment and to assure adequate coolant flow.

IV. DESIGN FEATURES

IV. DESIGN FEATURES

1.0 REACTOR FUEL

Applicability

This specification applies to the fuel elements used in the reactor core.

Objective

The objective is to 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.

Specifications

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

a. Uranium content: maximum of 9.0 w/o enriched to a nominal 20% Uranium 235.

b. Hydrogen-to-zirconium atom ratio (in the ZrH_x): Nominal 1.7H atoms to 1.0 Zr atoms.

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

Basis

A maximum uranium content of 9 w/o in a standard TRIGA element is greater than the design value of 8.5 w/o. 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% reduces the safety margin by at most 10%. The hydrogen-to-zirconium ration of 1.7 will produce a maximum pressure within the cladding during an accident well below the rupture strength of the cladding.

2.0 REACTOR CORE

Applicability

This specification applies 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.

Specification

a. The reactor core shall consist of standard TRIGA reactor fuel elements and a minimum of two (2) thermocouple instrumented TRIGA reactor fuel elements.

b. There shall be: four single core positions occupied by the three standard control rods and the transient rod; a neutron start-up 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. Any burnable poison used for the specific purpose of compensating for fuel burnup or long term reactivity adjustments shall be an integral part of the manufactured fuel elements.

Basis

Standard TRIGA cores have been in use for years and their safe operational characteristics are well documented.

3.0 CONTROL RODS

Applicability

This specification applies 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.

Specification

a. The standard control rods shall have scram capability and contain borated graphite, B_4C powder, or boron and its compounds in solid form as a poison in aluminum or stainless steel cladding. These rods may have an aluminum or air follower.

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

Basis

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 insure mechanical stability during movement and to isolate the poison from the pool water environment. Scram capabilities are provided for rapid insertion of the control rods which is the primary safety feature of the reactor. The transient control rod is designed for use in a pulsing TRIGA reactor.

4.0 FUEL STORAGE

Applicability

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

Objective

The objective is to assure that fuel which is stored 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 the provisions of 10 CFR 70. Irradiated fuel elements and fueled devices shall be stored in an array which will permit sufficient natural convective cooling by water or air such that the fuel element or fueled device temperature will not exceed design values. Such an array will contain not more than 12 fuel elements in a single storage rack.

Basis

The limits imposed by this specification are conservative and assure safe storage and handling. Experience shows it requires approximately 67 fuel elements, of the design used at AFRRI, in a close packed array to achieve criticality. Calculations show that in the event of a storage rack failure with all 12 elements falling in a contact configuration the mass would be less than that required for critcality.

5.0 REACTOR BUILDING AND VENTILATION SYSTEM

Applicability

This specification applies to the building which 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. The effluent from the reactor building ventilating system shall exhaust through absolute filters to a stack having a minimum elevation that is 18' above the roof level 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 ventilating system air ducts to the reactor room shall be equipped with positive sealing dampers activated by controls that will close off the ventilation to the reactor room automatically upon an alarm of the reactor room continuous air monitor.

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

Bases

The facility is designed such that the ventilation system 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.

V. ADMINISTRATIVE REQUIREMENTS

V. ADMINISTRATIVE REQUIREMENTS

1.0 ADMINISTRATIVE CONTROLS

1.1 RESPONSIBILITY

The Director, AFRRI shall be responsible for overall facility operation. The Physicist-in-Charge of the Reactor (PIC) will be responsible for day to day operations and for determination of applicability of procedures, experiment authorizations, and maintenance operations.

1.2 ORGANIZATION

1.2.1 - An organizational diagram for facility management and operation shown in figure 1.2-1 is an example of the facility structure. Organization changes may occur based on facility requirements and will be depicted in internal documents.

Director
AFRRI

Chairman, Scientific
Support Department

Radiological
Safety Department

Head, Radiation
Sources Division

Reactor and Radiation
Facility Safety Committee

Physicist-in-Charge

Reactor Operations Supervisor

Reactor Staff*

* Any Reactor Staff member has direct access to the Director involving matters of reactor safety.

Figure 1.2-1 FACILITY ORGANIZATION

1.2.2 - FACILITY STAFF QUALIFICATIONS

a. Physicist-in-Charge (PIC)

At the time of appointment to this position, the Physicist-in-Charge shall have an NRC Senior Reactor Operator License for the AFRRI reactor.

b. Reactor Operation Supervisor (ROS)

At the time of appointment to this position, the ROS shall have an NRC Senior Reactor Operator License for the AFRRI reactor.

c. Reactor Operator

At the time of appointment to this position, an individual shall have an NRC Reactor Operator License for the AFRRI reactor.

d. Reactor Operator Trainee

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

1.3 OPERATIONS

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

1. Senior Reactor Operator (SRO) on call but not necessarily on site.
2. Radiation control technician on call
3. Reactor Operator (RO) at console
4. Another person within the AFRRI complex able to carry out written emergency procedures, instructions of the operator, or to summon help in case the operator becomes incapaciated.

b. At least one licensed operator shall be at the controls when the reactor is not secured.

c. Maintenance activity that could affect the reactivity of the reactor will be accomplished under the supervision of a SRO.

1.4 TRAINING

A training and retraining program will be maintained to insure adaquate levels of proficiency in subjects related to the reactor and reactor operations.

1.5 REVIEW AND AUDIT

1.5.1 - Reactor and Radiation Facility Safety Committee (RRFSC)

1.5.1.1 - Function

The Reactor and Radiation Facility Safety Committee is directly responsible to the Director, AFRRI. This committee will review all radiological health and safety matters concerning the reactor and it's associated equipment, the reactor room, the reactor console, the exposure rooms, the pneumetic transfer system, the preparation area, the fuel element shipping casks, the reactor fuel and it's storage area.

1.5.1.2 - Composition and Qualifications

a. The RRFSC, as appointed by the Director, AFRRI shall be composed of:

(1) At least two of the following: Head, Radiation Sources Division; Head, Scientific Support Department; PIC Reactor; Head other large sources. One of the two individuals must be a licensed Senior Operator.

(2) Head Radiological Safety Department

(3) A representative of the AFRRI Directorate.

(4) Radiological Safety Officers or Source Facility Operators from at least three outside facilities, one of which must be non-DoD and at least one reactor physicist or reactor health physicist

(5) Representative of the scientific staff.

(6) Other qualified individuals as designated by the Director.

b. The minimum qualification for persons on the RRFSC shall be 5 years of professional work experience in the discipline or specific field they represent. A baccalaureate degree may fulfill 4 years of experience.

c. Non-voting members may be appointed by the Director, AFRRI or the Chairman, RRFSC.

1.5.1.3 - Alternates

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

1.5.1.4 - Meeting Frequency

The RRFSC or a subcommittee thereof shall meet at least four times a calendar year. The RRFSC shall meet at least semi-annually.

1.5.1.5 - Quorum

A quorum of the RRFSC for review shall consist of the Chairman or his designated alternate and four other members, or alternate members. A majority of those present shall be regular members.

1.5.1.6 - Review

The RRFSC shall review:

a. Safety evaluations for (1) changes to procedures, equipment, or systems and (2) tests or experiments conducted without NRC approval under provisions of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment, or systems that change the original intent or use, and are non-conservative, or those that involve an unreviewed safety question as defined in Section 50.59 of 10 CFR.

c. Proposed tests or experiments which are significantly different from previous approved tests or experiments, or those that involve an unreviewed safety question as defined in Section 50.59 of 10 CFR.

d. Proposed changes in Technical Specifications or licenses.

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

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

g. Events which have been reported to the NRC.

h. Audit reports.

1.5.1.7-Audit

Audits of facility activities shall be performed, under the cognizance of the RRFSC, but in no case by the personnel responsible for the item audited, annually not to exceed 14 months. Individual audits may be performed by one individual who need not be an identified RRFSC member. These audits shall examine the operating records and encompass:

a. The conformance of facility operation to the Technical Specifications and the license.

b. The performance, training, and qualifications of the entire facility staff.

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

d. The facility emergency plan and implementing procedures.

e. The facility security plan and implementing procedures.

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

1.5.1.8-Authority

The RRFSC shall report to the Director, AFRRI and advise the PIC in those areas of responsibility specified in Sections 1.5.1.6 and 1.5.1.7.

1.6 ALARA

Applicability

This specification applies to all reactor operations resulting in significant exposures.

Objective

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

Specification

As part of the review of all operations, consideration will be given to alternative operational profiles which 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. Similarily, 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.

2.0 SAFETY LIMIT VIOLATIONS

2.1 The following actions shall be taken in the event the safety limit is exceeded:

a. The reactor will be shutdown immediately and reactor operation will not be resumed without authorization by the Commission.

b. The safety limit violation shall be reported to the Director of NRC Region I Office of Inspection and Enforcement (or his designate), the Director, AFRRI, and to the RRFSC not later than the next working day.

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

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

3.0 PROCEDURES

3.1 There shall be written instructions that cover the following activities. They shall be approved by the Head, Scientific Support Radiation Sources Division (SSRS) and reviewed by Reactor and Radiation Facility Safety Committee (RRFSC):

a. Conduct of irradiations and experiments that could effect the operation and safety of the reactor.

b. Functions and organization of the reactor facility.

c. Administrative limits and controls for reactor operations.

d. Reactor staff training program.

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

f. Pneumatic transfer system and core experiment tube operations.

g. Personnel radiation protection consistent with 10 CFR 20.

h. Implementation of the Security Plan and Emergency Plan.

i. Reactor core loading and unloading.

3.2 Though substantive changes to the above procedures shall be made only with approval by the Chief, SSRS, temporary changes to the procedures that do not change their original intent may be made by the PIC. All such temporary changes shall be documented and subsequently approved by the Chief, SSRS.

4.0 EXPERIMENTS

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

a. Physicist-in-Charge (PIC) of the reactor

b. Chief, Scientific Support Radiation Sources Division (SSRS)

c. Radiological Safety Department

d. Reactor and Radiation Facility Safety Committee (RRFSC)

4.2 Prior to its performance, an experiment must be included under one of the following types of authorizations issued by the RRFSC:

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 PIC of the reactor or his designee.

b. Routine Reactor Authorization for experiments safely performed at least once. These experiments may be performed at the discretion of the PIC and coordinated with the Radiation Safety Department when appropriate.

4.3 Substantive changes to previously approved experiments shall be made only after review by the RRFSC and approval, in writing, by the PIC or his designated alternate. Minor changes that do not significantly alter the experiment may be approved by the ROS or PIC. Approved experiments shall be carried out in accordance with established procedures.

5.0 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the appropriate NRC Regional Office unless otherwise noted.

5.1 ROUTINE REPORTS

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 plan. The report shall address each of the tests identified in the FSAR and shall, in general, include a description of the measured values of the operating conditions or characteristics obtained during the tests 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 in 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, (3) 9 months following initial criticality, which ever 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 unit during the previous calendar year should be submitted prior to October 31 of each year, covering the previous fiscal year's operation.

The Annual Operating Report made by licensee shall provide a comprehensive summary of the operating experience having safety significance that was gained 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 major surveillance tests and inspections.

(2) A tabulation showing the power generated by the reactor on a monthly basis.

(3) List of the unscheduled shutdowns, including the reasons therefore and corrective action 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 or untried experiments or tests performed during the reporting period that are not described 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 10 CFR 50.59 which clearly shows the reason leading to the conclusion that no unreviewed safety question existed and that no change to the Technical Specifications was required.

(7) A summary of the nature and amount of airborne 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.

(a) Total estimated quantity of radioactivity released (in curies) determined by an approved sampling and counting method.

(b) Total estimated quantity of Argon-41 released (in curies) during the reporting period based on data from an appropriate monitoring system.

(c) Estimated average atmospheric diluted concentration of Argon-41 released during the reporting period in terms of microcuries/cc and fraction of the applicable MPC value.

(d) An estimate of the average concentration of other significant radionuclides present in the gaseous waste discharge in terms of microcuries/cc and fraction of the applicable MPC value for the reporting period if the estimated release is greater than 20% of the applicable MPC.

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

5.2 REPORTABLE OCCURRENCES

Reportable occurrences, including causes, probable consequences, corrective actions, and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe the final resolution of the occurrence. In case of corrected or supplemental reports, an amended Licensee Event Report shall be completed and reference shall be made to the original report data.

a. Prompt Notification With Written Followup. The types of events listed below shall be reported as expeditiously as possible by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the appropriate NRC Regional Office, or his designate no later than the first work day following the event, with a written followup report within two weeks. The written followup report shall include, as a minimum, a completed copy of a Licensee Event Report Form. Information provided on the Licensee Event Report Form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Failure of the reactor protection system, or other systems subject to limiting safety system settings, to initiate the required protective function at the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the Technical Specifications or failure to complete the required protective function.

(2) Operation of the reactor or affected systems when any parameter or operation subject to a limiting condition is less conservative than the limiting condition for operation established in the Technical Specifications without taking permitted remedial action.

(3) Abnormal degradation discovered in a fission product barrier, i.e., fuel cladding, reactor coolant boundary, or reactor building.

(4) Reactivity balance anomalies involving:

(a) Disagreement between expected and actual critical positons of approximately 0.3% Δ k/k;

(b) exceeding excess reactivity limit;

(c) shutdown margin less conservative than specified in the Technical Specifications;

(d) unexpected short-term reactivity changes that cause a positive period of 10 seconds or less;

(e) if sub-critical an unplanned reactivity insertion of more than approximately 0.5% Δ k/k or any unplanned criticality.

(5) Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

(6) Personnel error or procedural inadequacy which prevents, or could prevent by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAE.

(7) 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.

(8) 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 in a manner less conservative than assumed in the analyses.

(9) 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 of 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.

6.0 RECORD RETENTION

6.1 Records to be retained for a period of at least five years or as required by 10 CFR regulations:

a. Operating logs or data which 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 or 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 reactor.

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

(1) Records of radioactive material transferred from the facility as required by license.

(2) Records required by the RRFSC for the performance of new or special experiments.

g. Changes to operating procedures.

h. Fuel inventories and fuel transfers.

i. Records of transient or operational cycles for those components designed for a 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 10 CFR 50.59.

l. Records of meetings of the RRFSC.

6.2 Records to be retained for the life of the facility:

a. Gaseous and liquid radioacitve effluents released to the environs.

b. Appropriate off-site environmental monitoring surveys.

c. Radiation exposures for all personnel.

d. Updated as-built drawings of the facility.