10CFR 50.36
DOCKET 50-170

DEFENSE NUCLEAR AGENCY
ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE
BETHESDA, MARYLAND 20814-5145

DIR

May 17, 1991

SUBJECT: Additional Information on Proposed Technical Specifications Changes for License No. R-84.

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Please find enclosed nine (9) attachments that address the issues of your request for additional information dated 1 May, 1990. If you have any questions or comments, please contact the Reactor Facility Director, Mr. Mark Moore, or the Reactor Executive Officer, 1st Lt Matt Forsbacka, at (301) 295-1290. Your prompt reply will be greatly appreciated.

Sincerely,

[signature] for

GEORGE W. IRVING, III
Colonel, USAF, BSC
Director

Enclosures:
as stated

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9105310052 910517
PDR ADOCK 05000170
P PDR

cc:
Mr. Al Adams
USNRC, Mail Stop 11B20
Washington, D.C. 20555

Mr. Marvin Mendonca
USNRC, Mail Stop 11B20
Washington, D.C. 20555

Mr. Thomas Dragoun
USNRC, Region I
King of Prussia, PA 19406

Attachment 1

Comment 1

In your amendment request you have added a new definition for Reactor Facility Director (RFD) and have amended Section 6.1.2 of the Technical Specifications (TS) to include additional information concerning the responsibilities of the RFD. The two TS Sections affected by you request repeat the same information to a large extent. However, the list of responsibilities for the RFD differs between the definition and Section 6.1.2 of the TS. Please consolidate your requested changes concerning the RFD to one place in the TS. We can find no Section 6.1.3.1. Please Correct. Section 6.1.2 of the TS also discusses brief absences of the RFD and his designee. What time interval do you mean by "brief"? Justify the time interval chosen.

Reply to Comment 1

To consolidate the changes concerning the information pertaining to the RFD, please delete the proposed definition 1.17 REACTOR FACILITY DIRECTOR. The proposed Section 6.1.2 should be changed as follows:

> The Director, AFRRI, shall have license responsibility for the reactor facility. The Reactor Facility Director (RFD) shall be responsible for administration and operation of the Reactor Facility and for determination of applicability of procedures, experiment authorizations, maintenance, and operations. The RFD may designate an individual who meets the requirements of Section 6.1.3.1.a to discharge the RFD's responsibilities in the RFD's absence. During brief absences (periods less than four hours) of the Reactor Facility Director and his designee, the Reactor Operations Supervisor shall discharge these responsibilities.

The four hour time interval was chosen to allow the RFD and his designee to be absent for short periods of time to have lunch, run errands, or attend short meetings outside of AFRRI.

Attachment 2

Comment 2

Amendment No. 20 issued on October 4, 1990 amended Section 4.2.5 of the TS. This Section new differs from your request of April 30, 1990. Please provide a revised Section 4.2.5 to reflect the addition of fuel followed control rods (FFCRs) to the reactor. Also, we note that on the second line of the basis of the TS, we introduced a typographical error by spelling cladding, "classing". Please correct this error in your request.

Reply to Comment 2

The request of April 30, 1990 included information pertaining to the FFCRs. The original request is reproduced here for your convenience:

Page 22, Section 4.2.5: Replace the Specifications in its entirety as follows to clarify the requirement for fuel element surveillance and to accommodate the fuel follower control rods:

All the fuel elements present in the reactor core, to include fuel follower control rods, shall be inspected 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 15 months), whichever occur first. Fuel elements in long-term storage need not be measured until returned to core; however fuel elements routinely moved to temporary storage shall be measured every 500 pulses of insertion greater than \$2.00 or annually (not to exceed 15 months), whichever occurs first.

Safety Analysis: Fuel in long-term storage is not subject to the rigors of the fuel used in the reactor core. Thus damage to the fuel which may manifest itself as elongation or lateral bow is not a possibility for fuel in long-term storage. The high degree of purity maintained in the AFRRI TRIGA pool assures that deterioration or corrosion of the cladding will not be a problem.

Page 22, Section 4.2.5: Replace the Basis in its entirety for clarification as follows:

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

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

Safety Analysis: This is an administrative change in the wording to clarify the passage. There are no safety implications.

Attachment 3

Comment 3

You have requested an amendment to Section 4.4 of the TS concerning verification of operation of the ventilation systems. You give as justification for the change conformance with ANS Standard 15.1. However, the wording you have chosen does not match ANS 15.1. For monthly inspections, the interval shall not exceed six weeks. Please amend you wording to conform with ANS 15.1.

Reply to Comment 3

Please replace the Specifications of Section 4.4 in its entirety as follows to conform with the latitude recommended in ANSI 15.1.4.f:

> 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).

Attachment 4

Comment 4

You have requested to add information to the basis of Section 5.2.1 of the TS to justify addition of the FFCRs to the reactor. In light of the additional information you have provided in response to requests for additional information from NRC concerning the FFCRs, please confirm the changes to this Section requested on April 4, 1990 are still valid.

Reply to Comment 4

Please replace the second paragraph for the basis of Section 5.2.1 of the TS with the following:

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

Attachment 5

Comment 5

In your submittal of April 4, 1990, you describe changes to Section 6.1.1 of the TS. However, there are additional changes to Figure 1 that you do not describe or justify. Please explain and justify all of the changes mode on Figure 1.

Reply to Comment 5

All changes to Figure 1 in Section 6.1.1 are administrative changes to reflect changes in existing terminology. There are no safety implications. Please replace "Chief, Radiation Sources Division" with "Chief, Reactor Division" and "Reactor Staff" with "Reactor Operations Staff".

Attachment 6

Comment 6

You have requested amendment of Section 6.1.3.1.b of the TS concerning the Reactor Operations Supervisor (ROS). You discuss the requirement for the ROS to have one year experience before appointment as the ROS. What type of experience is this? Licensed as a Senior Reactor Operation at AFRRI? Please clarify and justify this change.

Reply to Comment 6

Replace the requirements of Reactor Operations Supervisor in its entirety with the following:

> At the time of appointment to this position, the ROS shall have 3 years nuclear experience. Higher education in a science or nuclear engineering field may fulfill up to 2 years of experience on a one-for-one basis. The ROS shall hold a USNRC Senior Reactor Operator license on the AFRRI reactor. In addition, the ROS shall have 1 year of experience as a USNRC licensed Senior Reactor Operator at AFRRI or at a similar facility before the appointment to this position.

Attachment 7

Comment 7

In light of your request to increase authorized reactor power to 1100 Kw, do you desire changes to section 3.1.1 of the TS?

Reply to Comment 7

No. It is our stated intention to limit normal steady-state power operations to 1000 kW. The purpose of the request to increase the maximum steady-state power level allowed by the Facility Operating License is to resolve the conflict with TS 3.1.1 which allows a maximum steady-state power level of 1.1 MW for the purposes of testing and calibration.

Attachment 8

Comment 8

In our Request for Additional Information dated November 14, 1990, we requested additional information on your request to increase maximum steady-state power level. Your reply was limited to FFCRs. Please provide a safety analysis that discusses operation of the reactor at 1100 kW. For reference, we have enclosed a Safety Analysis for Oregon State University that supported an amendment raising their licensed maximum power to 1100 kW.

Reply to Comment 8

SAFETY ANALYSIS OF INCREASE OF MAXIMUM ALLOWED STEADY STATE POWER LEVEL FROM 1000 kW to 1100 kW

Introduction

The purpose of this safety analysis is to resolve the inconsistency in the maximum allowed steady- state power level reported in the Technical Specifications and the Facility Operating License No. R-84. Technical Specification (TS) 3.1.1 allows a maximum steady-state power level of 1100 kW for the purposes of testing and calibration. Amendment No. 18 of the Facility Operating License, however, states that the maximum steady-state power level shall be 1000 kW. As stated in TS 3.1.1, the normal steady-state operating power limit of the reactor shall be 1000 kW, so it is our intent to simply amend the Facility Operating License to reflect the maximum power limit allowed in the Technical Specifications.

The maximum steady-state power level of 1100 kW allows for the following processes:

- When the AFRRI TRIGA reactor is operating in the automatic mode, it is normal for the feedback effects of a properly functioning servo system to result in small (1-2%) power oscillations around the mean power. When the reactor is at 1.0 MW it is possible to briefly operate at powers slightly greater than 1.0 MW without violating the Facility Operating License.

- Table 2 of the AFRRI Technical Specifications indicates that the maximum set point of the high flux safety channel may be set at 1.1 MW. Thus if the reactor is shut down by a high power scram, there would be no violation of the Technical Specifications provided the power did not exceed 1.1 MW. Additionally, the provision to physically rather than electronically test the high flux safety channel exists.

As stated earlier, the intent of this license modification is not to allow for prolonged steady state operations at power levels of 1100 kW. Any steady-state operations that exceed 1.0 MW will be brief in nature (tens of seconds or less), so there will be no significant change in the fission product inventories reported in the design basis accidents described in the Safety Analysis Report for AFRRI TRIGA Mark-F Reactor.

Safety Limitations

The safety limit for a TRIGA reactor is the fuel element temperature. If the fuel temperature exceeds its safety limit, the disassociation of hydrogen and zirconium in the fuel-moderator could cause a hydrogen overpressure condition between the fuel-moderator and the cladding which could result in a loss of cladding integrity. The safety limit for the standard TRIGA fuel is based on data that includes a great deal 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 disassociation of zirconium hydride will remain below the ultimate stress, provided that the temperature of the fuel does not exceed 1000 C while immersed in water.

For steady-state operation, fuel temperatures are dependent upon the heat transfer characteristics of the fuel element and coolant. As established in "Maximum Temperature Calculation and Operational Characteristics of Fuel Follower Control Rods for the AFRRI TRIGA Reactor

Facility" (See correspondence to USNRC dated March 5, 1991) the maximum possible power density in the AFRRI TRIGA reactor at a steady-state power level of 1.1 MW is 72.4 W/cm^3. Equation 17 in the referenced report will allow us to determine the maximum fuel temperature for a fuel element in the cente. of the core:

$$T_i = T_f + \frac{q'''R_i^2}{4k_f}\left(\left(\frac{R_o}{R_i}\right)^2 - 2\ln\left(\frac{R_o}{R_i}\right) - 1\right) + \frac{q'''R_i^2}{2}\left(\left(\left(\frac{R_o}{R_i}\right)^2 - 1\right)\left(\frac{1}{k_c}\ln\frac{R_o+c}{R_o}\right) + \frac{1}{h(R_o+c)}\right)$$

Where
T_i = *maximum fuel temperature*
T_f = *bulk coolant temperature*
R_o = 1.82 *cm (radius of s'andard fuel element)*
R_i = 0.23 *cm (radius of Zr rod)*
c = 0.051 *cm (cladding thickness)*
k_f = 0.18 $W/cm-^\circ C$ *(thermal conductivity of U Zr H)*
k_c = 0.138 $W/cm-^\circ C$ *(thermal conductivity of SS304)*
h = 1.339 $W/cm^2-^\circ C$ *(free convective heat transfer coefficient of water)*
q''' = 72.4 W/cm^3

Solving the above equation with a bulk water temperature of 48.6 C (the condition at which h was determined) yields a maximum fuel temperature of 425.1 C. It should be noted that the reported value is hypothetical as the central thimble in the AFRRI TRIGA is occupied by a control rod rather than a fuel element. This does, however, represent the most conservative value of the maximum attainable temperature in any case of steady-state operation. As shown by this calculation, the maximum temperature of this conservative limiting case falls nearly 575 C short of the technical specifications limit of 1000 C. Furthermore, the limiting safety system setting of 600 C will remain exactly the same.

Operational Impact

The proposed change to the Facility Operating License will have no impact on either pulse or steady-state operations. No modification of operating procedures, maintenance procedures, or technical specifications will be required if the proposed change is approved. In addition, there is no impact on the DBA as reported in the SAR as the DBA assumes 100 continuous hours of operation at the maximum allowed steady state power before a fuel element is ruptured due to a mechanical impact. Because the 1.1 MW steady-state power limit is intended only for short durations of time, the saturated fission product inventory will be unchanged. Thus no changes

in the facility SAR or emergency plan or procedures will be required.

The basis for Technical Specification 3.1.1 states: "Thermal and hydraulic calculations and operation experience indicate that TRIGA fuel may be safely operated up to power levels of at least 1.5 megawatts with natural convective cooling." In addition, the AFRRI heat exchanger is rated at 1.5 MW. Since it is our stated intention to limit routine steady-state operations to 1.0 MW, we anticipate no changes in cooling capabilities.

Conclusion

There are many TRIGA reactor-years of experience which attest to the safety of this reactor type at power levels measurably in excess of 1.1 MW. We would like to reference the NRC's Safety Evaluation Reports (SERs) for the University of Texas and Oregon State University TRIGA reactors which determined a 1.1 MW peak power was found to be completely safe and acceptable. In addition, the basis for the current Technical Specification 3.1.1 establishes that the maximum safe steady-state power for a natural convection cooled TRIGA reator is 1.5 MW.

Attachment 9

Proposed Replacement Pages
for the
Technical Specifications
for the AFRRI Reactor Facility

Armed Forces Radiobiolgy Research Institute
Bethesda, MD 20889-5145

May, 1991

TECHNICAL SPECIFICATIONS FOR THE
AFRRI REACTOR FACILITY
LICENSE NO. R-84
DOCKET #50-170

TABLE OF CONTENTS

c. Reactor Pool

d. Core Experiment Tube

e. Portable Beam Tubes

f. Pneumatic Transfer System

g. Incore Locations

1.9 FUEL ELEMENT

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

1.10 INSTRUMENTED ELEMENT

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

1.11 LIMITING SAFETY SYSTEM SETTING

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

1.12 MEASURED VALUE

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

1.13 MEASURING CHANNEL

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

1.14 ON CALL

A person is considered on call if

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

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

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

1.15 OPERABLE

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

1.16 PULSE MODE

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

1.17 REACTOR OPERATION

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

1.18 REACTOR SAFETY SYSTEMS

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

1.19 REACTOR SECURED

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

a. The reactor is shut down.

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

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

1.20 REACTOR SHUTDOWN

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

1.21 REPORTABLE OCCURRENCE

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

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

b. Operation in violation of any Limiting Condition for Operation, Section 3.

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

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

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

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

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

1.22 SAFETY CHANNEL

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

1.23 SAFETY LIMIT

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

1.24 SHUTDOWN MARGIN

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

1.25 STANDARD CONTROL ROD

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

1.26 STEADY STATE MODE

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

1.27 TRANSIENT ROD

The transient rod is a control rod with scram capabilities that can be rapidly ejected from the reactor core to produce a pulse. It is activated by applying compressed air to a piston.

Specification

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

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

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

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

Basis

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

Specifications

All the fuel elements present in the reactor core, to include fuel follower control rods, shall be inspected 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 15 months), whichever occur first. Fuel elements in long-term storage need not be measured until returned to core; however fuel elements routinely moved to temporary storage shall be measured every 500 pulses of insertion greater than $2.00 or annually (not to exceed 15 months), whichever occurs first.

Basis

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

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

4.3 COOLANT SYSTEMS

Applicability

This specification applies to the surveillance requirements for monitoring th   ool 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   t to the water purification system, shall be measured daily, whenever operation are planned.

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

Basis

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

4.4 VENTILATION SYSTEM

Applicability

This specification applies to the facility ventilation system isclation.

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.

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.

Basis

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 insures an adequate dilution of effluents well above ground level. The separate ventilation system branch insures a dedicated air flow system for reactor effluents.

5.2 REACTOR CORE AND FUEL

5.2.1 REACTOR FUEL

Applicability

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

Objective

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

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

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

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

Basis

A maximum uranium content of 9 weight percent in a standard TRIGA element is greater than the design value of 8.5 weight percent, and encompasses the maximum probable variation in individual elements. Such an increase in loading would result in an increase in power density of less than 6%. An increase in local power density of 6% in 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. Because the volume of the fuel in a fuel follower fuel element is 56% of the volume of a standard TRIGA fuel element, however, the actual power produced in a fuel follower fuel element is 33% less that the power produdced 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 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. 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

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, B4C 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 the Specifications of 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.

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 by the rapid insertion of the control rods, which is the primary operational safety feature of the reactor. The transient control rod is designed for use in a pulsing TRIGA reactor.

5.3 SPECIAL NUCLEAR MATERIAL STORAGE

Applicability

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

Objective

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

Specification

All fuel elements not in the reactor core shall be stored and handled in accordance with applicable regulations. Irradiated fuel elements and fueled devices shall be stored in an array that will permit sufficient natural convective cooling by water or air, so that the fuel element or fueled device temperature will not exceed design values. Storage shall be such that groups of stored fuel elements will remain subcritical under all conditions of moderation.

Basis

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

6.0 ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION

6.1.1 STRUCTURE

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

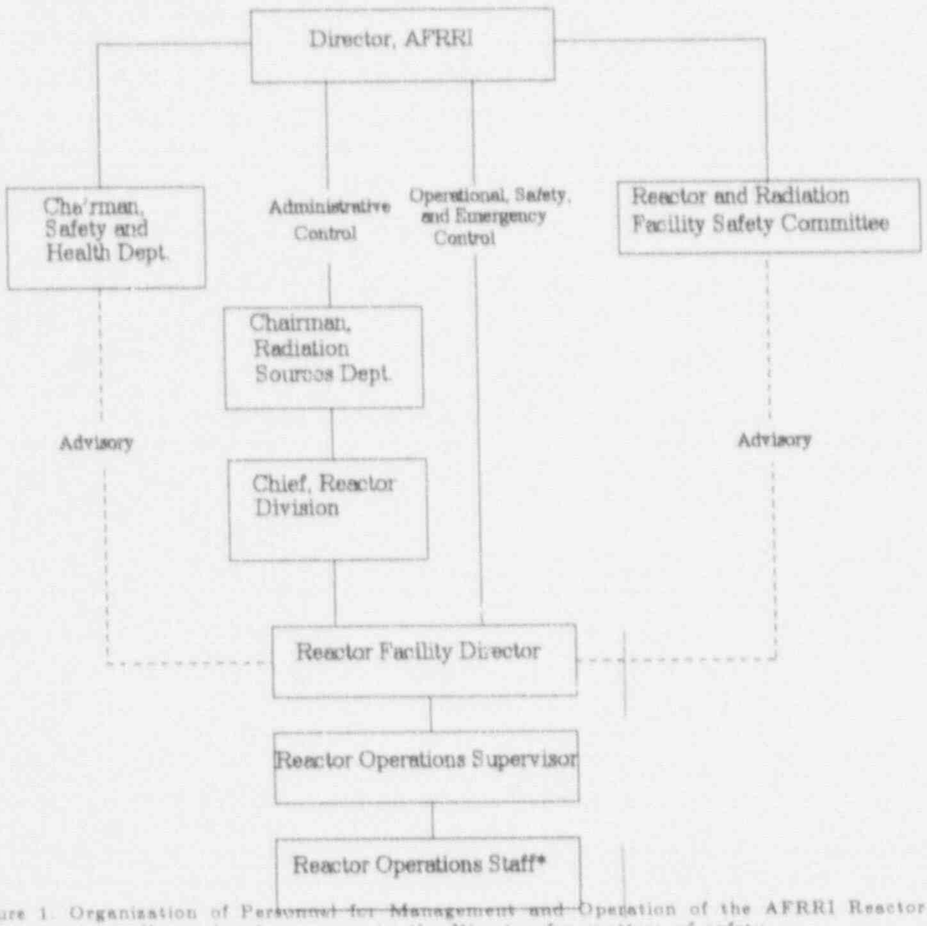

Figure 1. Organization of Personnel for Management and Operation of the AFRRI Reactor Facility.
* Any reactor staff member has access to the Director for matters of safety.

6.1.2 RESPONSIBILITY

The Director, AFRRI, shall have license responsibility for the reactor facility. The Reactor Facility Director (RFD) shall be responsible for administration and operation of the Reactor Facility and for determination of applicability of procedures, experiment authorizations, maintenance, and operations. The RFD may designate an individual who meets the requirements of Section 6.1.3.1.a to discharge the RFD's responsibilities in the RFD's absence. During brief absences (periods less than four hours) of the Reactor Facility Director and his designee, the Reactor Operations Supervisor shall discharge these responsibilities.

6.1.3 STAFFING

6.1.3.1 Selection of Personnel

a. Reactor Facility Director

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

b. Reactor Operations Supervisor (ROS)

At the time of appointment to this position, the ROS shall have 3 years nuclear experience. Higher education in a science or nuclear engineering field may fulfill up to 2 years of experience on a one-for one basis. The ROS shall hold a USNRC Senior Reactor Operator license on the AFRRI reactor. In addition, the ROS shall have 1 year of experience as a USNRC licensed Senior Reactor Operator at AFRRI or at a similar facility before the appointment to this position.

c. Reactor Operators/Senior Reactor Operators

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

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

6.1.3.2 Operations

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

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

2. Radiation control technician on call

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

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.