TECH SPECS
DOCKET 50-170


**DEFENSE NUCLEAR AGENCY**
ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE
BETHESDA, MARYLAND 20889-5145


RSDR

SUBJECT: Submission of Annual Report

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Dear Sir:

Attached please find the 1991 Annual Report for the AFRRI TRIGA reactor facility, submitted as required by license R-84, facility docket 50-170.

Should you need any further information, please contact the undersigned at (301) 295-1290.

Mark Moore

Attachment
as stated

MARK MOORE
Reactor Facility Director

CY Furn:

U.S. Nuclear Regulatory Commission
ATTN: Mr. Marvin Mendonca, Mail Stop 11H10
Washington, DC 20555

U. S. Nuclear Regulatory Commission, Region I
ATTN: Mr. Thomas Dragoun
475 Allendale Road
King of Prussia, PA 19406

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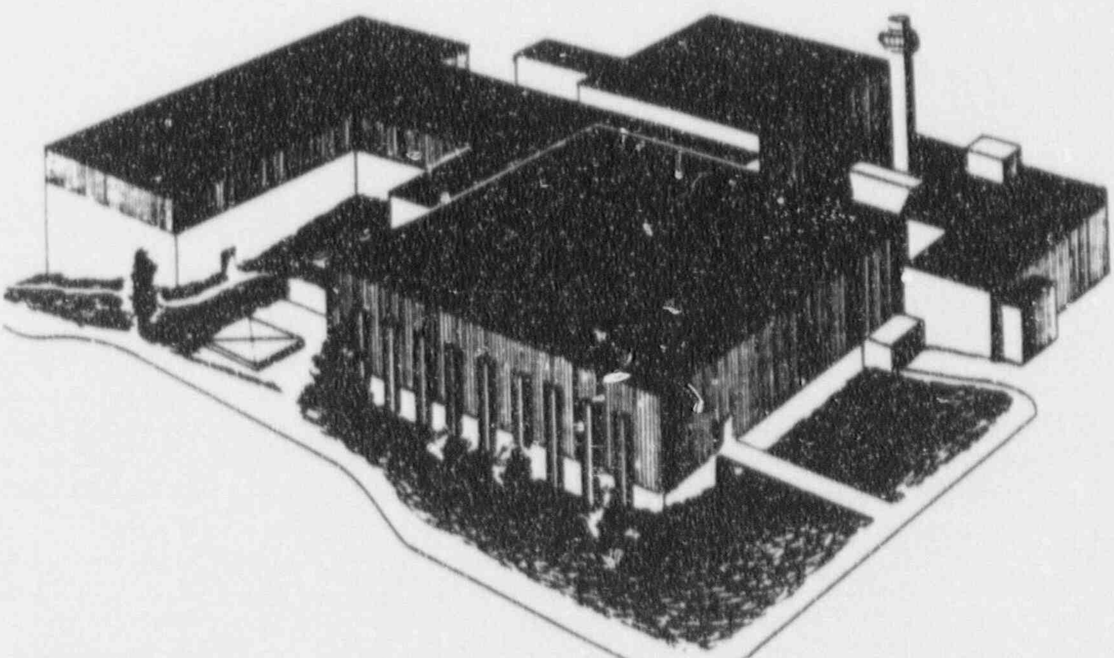
# 1991

# ANNUAL REPORT

# OF

# AFRRI TRIGA REACTOR



# 1991

# ANNUAL REPORT

# OF

# AFRRI TRIGA REACTOR


Submitted by:

MARK MOORE
Reactor Facility Director


Docket 50 - 170
License R - 84

Reviewed and Approved

Mark L. Moore 23 March 92
MARK L. MOORE Date
Reactor Facility Director

Approved for Release

Robert L. Bumgarner 23 MAR 92
ROBERT L. BUMGARNER Date
Captain, MC, USN
Director

1991 ANNUAL REPORT

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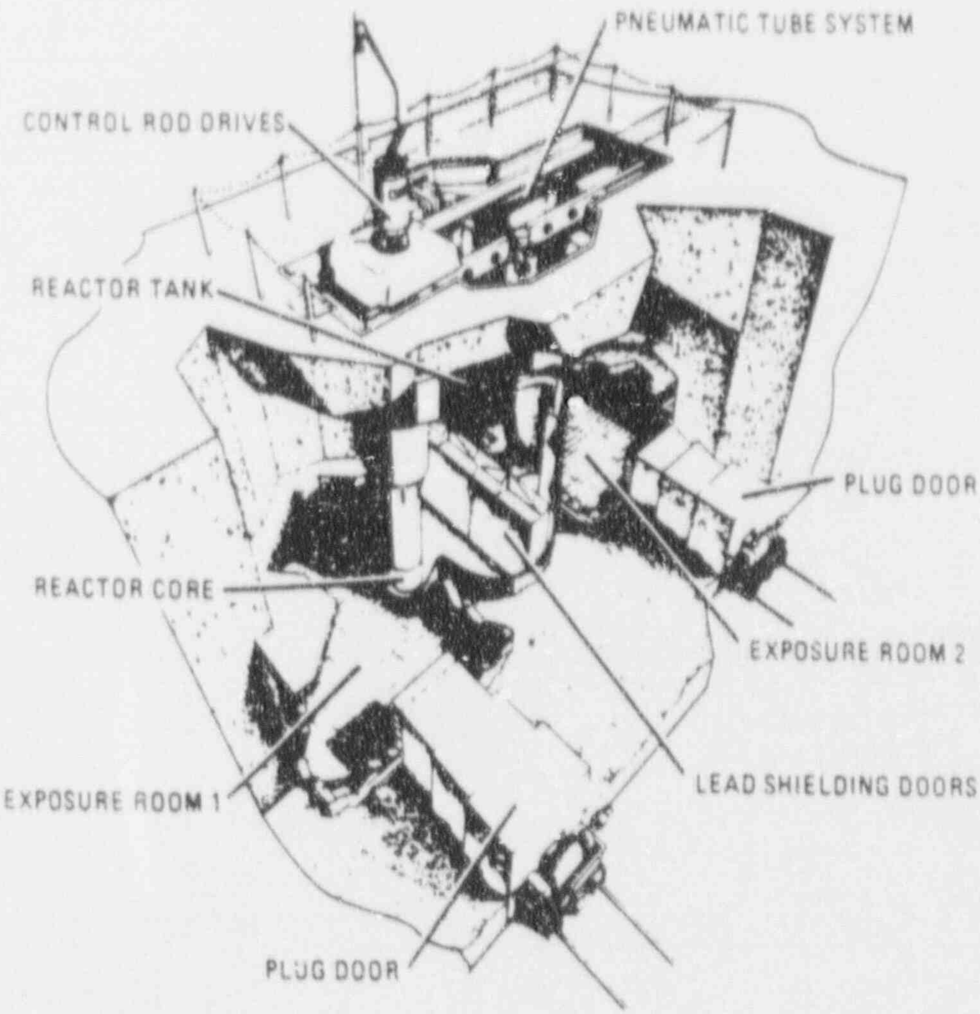
# INTRODUCTION




Cutaway View of AFRRI TRIGA Reactor

1991 ANNUAL REPORT

INTRODUCTION:

In 1991, the AFRRI reactor staff accomplished two important milestones in continuing efforts to improve the operational capabilities and reliability of the reactor facility. After Nuclear Regulatory Commission (NRC) approval of the installation through Amendment 21 to the facility operating license (Attachment C), the reactor staff installed three new fuel-follower standard control rods, a new longer air-follower transient control rod, and related equipment during an extended maintenance shutdown period of November-December 1991. Significant post-installation testing and calibration was performed to verify the validity of the original safety analysis and refueling plan (Attachment A) and the reactor was returned to regular steady-state operation on 16 December 1991. Approval for resumption of pulsing operations was withheld by the Reactor Facility Director pending completion of an testing program of incremental sized pulses in early 1992. That testing program will include pulses up to $2.50, but the current administrative pulse size limit of $2.00 remains in effect for normal operations.

Also, the microprocessor-based instrumentation and control console installed in 1990 completed its first full year of successful operation. Two modifications made to the console to improve its capabilities are discussed fully in Sections I and V and Attachment E.

The Reactor Facility was inspected by the Defense Nuclear Agency Inspector General from 25 to 27 September 1991. The inspection found that the AFRRI TRIGA Reactor Facility received an overall grade of SATISFACTORY with no significant discrepancies noted in organization, operations, material handling, and safeguards. There were no reportable deficiencies found in the maintenance records, operational log books, or training records. During the inspection, four minor deficiencies were documented. None of the cited deficiencies were considered significant and, neither singly nor in aggregate, impaired the performance or degraded the safety of the TRIGA nuclear reactor operations. Concurrently with that inspection, the reactor facility also received a Nuclear Reactor Security Inspection by representatives of the Assistant to the Secretary of Defense, Atomic Energy. That inspection also gave the reactor facility an overall rating of SATISFACTORY.

The Reactor Facility was also inspected by NRC personnel from Region I on 23-25 April 1991. No violations were identified during this inspection.

Changes were made to the procedures and facilities during 1991. These changes were supported by an extensive safety review process in accordance with the provisions of 10 CFR 50.59. The changes will be discussed fully in Sections I and V.

No trainees were added to the reactor staff during 1991. Two former trainees obtained Senior Reactor Operator licenses and one former trainee received his Reactor Operator license. One Senior Reactor Operator, Mr. Thomas Wright, departed during the year. Also, one Senior Reactor Operator, Mr. Stephen Holmes, whose license was terminated on 29 November 1990 was relicensed in 1991. Requests from non-AFRRI investigators continued to supplement the substantial inhouse experimental work load. These experimenters included representatives from the National Institutes of Health (NIH), Smithsonian Institution, National Institute of Standards and Technology (NIST), Naval Medical Research Institute

(NMRI), and the University of Maryland at Baltimore. The reactor staff was also tasked with providing personnel to assist in conducting inspections of the Fast Burst Reactor facilities at Aberdeen Proving Grounds, Maryland and White Sands Missile Range, New Mexico. At the request of Cornell University, an operations audit of their reactor facility was conducted.

The AFRRI documentation on financial assurance for decommissioning was accepted by the NRC on 11 February 1991.

No Licensee Event Reports were submitted during the year, however there were several malfunctions and unplanned shutdowns that are discussed in Sections III and IV.

The remainder of this report is written in a format to include notification items required by the AFRRI TRIGA Reactor Technical Specifications. Items not specifically required but of general informational value are presented in the General Information section. Each section following the general information corresponds to the required section as listed in Section 6.6.1.b of the AFRRI TRIGA Reactor Technical Specifications.

# GENERAL

# INFORMATION

Key Personnel

Reactor and Radiation Facility
Safety Committee.

GENERAL INFORMATION:

All personnel held their positions as listed throughout the entire year unless otherwise specified.

1. Current key AFRRI personnel (as of 31 December 1991) are as follows:

Director - CAPT Robert L. Bumgarner, USN (effective 30 August)

Chairman, Radiation Sources Department - CAPT C. B. Galley, USN

Chairman, Safety and Health Department - Mr. Thomas J. O'Brien and AFRRI Radiation Protection Officer (effective 1 June)

2. Reactor Facility Director - Mr. Mark Moore (SRO)

3. Current key Reactor Operations Personnel:

Reactor Operations Supervisor - Capt Matthew Forsbacka, USAF (SRO) (effective 31 May)

Training Coordinator - CPT Christopher Owens (SRO effective 7 May)

Maintenance/Procurement - Mr. Robert George (SRO)

Administration - MSG Harry Spence (SRO)

Other Senior Reactor Operators - Mr. John Nguyen (SRO effective 7 May)
Mr. Stephen Holmes (SRO effective 15 Oct)

Reactor Operator - SFC Michael Laughery (RO effective 7 May)

4. Senior Reactor Operator Candidates: None

5. Departures during CY 1991:

Mr. Thomas Wright (SRO license terminated 31 May)

6. There were several changes to the RRFSC during the 1991 calendar year. Mr. Thomas O'Brien replaced Mr. Douglas Ashby as Chairman, Safety and Health Department effective 1 June. CDR Joseph E. DeCicco of the Navy Dosimetry Center served as a regular member from 1 January until his retirement on 30 November. Dr. Samuel Levine was appointed as a special member effective 9 December to assist in review of the fuel-follower control rod installation. Mr. John Dickson and Ms. Leslie Moore of the Calvert Marine Museum were appointed as special members on 10 September to perform special projects related to reactor pool water quality.

The 1991 RRFSC consisted of the following membership in accordance with AFRRI Reactor Technical Specifications (as of 31 December):

Chairman: Col. Nicholas Manderfield, USAF (Director's Representative)

Regular Members:

Mr. Thomas O'Brien (Chairman, Safety and Health Department, AFRRI)
Mr. Mark Moore (Reactor Facility Director, AFRRI)
Dr. Marcus Voth (Director, Breazeale Reactor and Professor of Nuclear Engineering, Pennsylvania State University)
Mr. Ron Luerson (Safety Directorate, Naval Research Laboratory)

Special Members:

CAPT C.B. Galley, USN (Chairman, Radiation Sources Dept)(Certified HP)
Dr. Samuel Levine (SHL Nuclear Associates)(Nuclear Engineer-Reactor Specialist)
Mr. John Dickson (Calvert Marine Museum)(Water Quality Specialist)
Ms. Leslie Moore (Calvert Marine Museum)(Water Quality Specialist)

Non-voting Observer:

Mr. James Caldwell (EPA, Montgomery County, MD)

Recorder:

Ms. Carol King

Meetings of the RRFSC were held:

20 February 1991

27 June 1991 (Subcommittee)

24 September 1991

17 December 1991

# SECTION I

Changes to the Facility Design
Performance Characteristics
and Operational Procedures.
Results of Surveillance Tests
and Inspections.

SECTION I

Changes to the facility design, performance characteristics, operational procedures, and results from surveillance testing are contained in this section.

A. DESIGN CHANGES:

1. As previously discussed in the Introduction section, three new fuel-follower standard control rods, a new air-follower transient control rod, and associated equipment were installed during the year. Also, the control rod drive support structure was modified to permit easier access to the transient rod drive mechanism for maintenance. (Attachment E-1)

2. The pool water gamma activity monitoring system was redesigned to replace vacuum tube technology with a more reliable solid state system (Attachment E-2). Also, the primary water conductivity readout modules in the control room were replaced by newer models. The sensors and overall system design remain the same and no changes to the flow of the primary coolant system were made. (Attachment E-3)

3. The air particulate monitor (CAM) automatic damper closure system was modified by adding an override circuit to allow an operator to open the reactor room air dampers while a CAM is alarming. This temporary modification would allow controlled venting of the reactor room should a cladding failure occur when it would be necessary to pull air through the absolute filters, for example, during pulse testing of the new fuel-follower control rods. (Attachment E-4)

4. The steady-state timer originally installed on the new reactor console was a count-down timer. That timer was replaced with a count-up timer to facilitate determination of steady-state run lengths since runs are often terminated based on dosimetry measurements and not at predetermined elapsed times. (Attachment E-5)

5. The lead-acid battery back-up system that previously provided standby power to the criticality monitor and radiation area monitors (RAMs) was replaced by a larger uninterruptible power supply (UPS) necessary for providing more stable current and decreasing maintenance requirements. Replacement of the lead-acid batteries also eliminates a hazardous chemical problem and hydrogen explosion hazard.

6. Hardware and software modifications were made to the control console to install the new low pulse display option and new pulse mode scram timer (Attachment E-6). The pulse timer allows the operator to adjust the amount of time the transient rod remains up during a pulse as per earlier preapproved console installation plan. The new small pulse display option allows high sensitivity pulse data acquisition and a more readable display of pulse characteristics on the console CRT.

B. PERFORMANCE CHARACTERISTICS:

Installation of the fuel-follower control rods and readjustment of the reactor core loading resulted in an increase in the nominal control rod worths by an average of \$0.40 (0.0028 $\Delta k/k$) and increased the k-excess measurement at infinite water by \$0.67 (0.0047 $\Delta k/k$). The negative temperature coefficient of reactivity and the reactor tank constant were not affected. Specific details of the installation and testing program will be presented in the Startup Report to be submitted as required by the reactor Technical Specifications in early 1992.

C. ADMINISTRATIVE PROCEDURES:

Several changes to the Reactor Administrative Procedures were approved and implemented during the year. A complete set of current administrative procedures is included at Attachment B.

Administrative Procedure A1 was changed to clarify the recording requirements when an operator is using over-the-counter medications by requiring operators to check possible side effects in the Physician's Desk Reference, notify the RFD, and log the medication to a Drug Log.

Administrative Procedure A2 was changed to delete the escorted access roster for the exposure room prep area. All persons allowed in the area now receive training in radiation safety to allow them unescorted access.

Administrative Procedure A3 was revised to simplify the 10 CFR 50.59 worksheets to implement recommendations from various NRC inspections.

Administrative Procedure A4 is a completely new procedure developed in conjunction with NRC inspection recommendations on inventory procedures for special nuclear material.

D. OPERATIONAL PROCEDURES:

Numerous changes were made to the operational procedures to improve clarity and to account for the new fuel-follower control rods, new low pulse display, and other facility modifications. The changes made during the major revision in May 1991 at listed at Attachment F and subsequent 1991 changes are summarized below. A complete set of current operational procedures is at Attachment B.

1. Procedure 4, Personnel Radiation Protection, was revised to specifically indicate that AFRRI Instruction 6055.8, Occupational Radiation Protection Program, is the radiation protection program followed by the reactor.

2. Procedure 8, Reactor Operations, was extensively revised as follows:

a. Basic: The basic procedure was revised to clarify that respirator equipment is intended for use only during emergency conditions, not on a routine basis. Changes were also made to record the SRO on-call at the beginning and end of each day or if a change occurs during the operational day instead of on each logbook page. Also, a new designation, physicist in charge (PIC), is added to clarify the individual in charge physically present at the reactor.

b. Tab A: The logbook entry checklist was changed to show that SRO on-call and PIC entries would be in black ink.

c. Tab C: The nuclear instrumentation setpoints were adjusted for the new reactor control console and to delete references to the obsolete stack particulate monitoring system.

d. Tab D: The k-excess procedure was revised to change the standard power level for performance from 15 watts to 5 watts. This allows better readability of the power level on the reactor console linear power indicator.

E. RESULTS OF SURVEILLANCE TESTS AND INSPECTIONS:

All required maintenance and surveillance items were accomplished as required. Malfunctions discovered are detailed in Section IV.

# SECTION II

Energy Generated by Current
Reactor Core and Number of Pulses
$2.00 or Larger.

SECTION II

Energy generated by the reactor core:

| MONTH | KwHrs |
|---|---|
| JAN | 1883.3 |
| FEB | 3115.1 |
| MAR | 4417.6 |
| APR | 5993.6 |
| MAY | 1747.0 |
| JUN | 4715.9 |
| JUL | 1575.3 |
| AUG | 2303.6 |
| SEP | 2078.8 |
| OCT | 1343.0 |
| NOV | 1509.4 |
| DEC | 1370.9 |
| TOTAL | 32053.5 |

| | |
|---|---|
| Total energy generated this year: | 32053.5 KwHrs |
| Total energy on fuel elements: | 734168.6 KwHrs |
| Total energy on FFCRs: | 1370.9 KwHrs |
| Total Pulses this year ≥ $2.00: | None |
| Total Pulses on fuel elements ≥ $2.00: | 4112 |
| Total Pulses on FFCRs ≥ $2.00: | None |
| Total Pulses on fuel elements: | 9765 |
| Total Pulses on FFCRs: | None |

# SECTION III

Unscheduled Shutdowns

SECTION III

Unscheduled Shutdowns:

There were four unscheduled shutdowns during this reporting period.

1. During a 1 MW steady-state run on 7 March, a reactor scram occurred when a high voltage drop in the NP-1000 channel exceeded the trip point of 10%. Investigation showed that the trip point had been set more conservatively than the required 20% limit. The trip point was readjusted to 20% and operations continued.

2. An "NPP High Voltage Low" scram occurred during a 1 MW steady-state run on 1 August. The voltage circuit was tested and no problem could be found. The run was resumed and the reactor operated normally.

3. During a 1 MW steady-state run on 25 October, a reactor scram occurred because of a momentary loss of site electrical power affecting a broad local area. All required safety instrumentation continued to operate on back-up uninterruptible power supplies and normal power was restored in only a few seconds.

4. During control rod calibrations on 4 December at a power level of 5 watts, a reactor scram occurred with a "Timer Scram" indication on the control console CRT screen. At the time, the timer scram was not enabled nor was the steady-state timer running. Testing of both the related console hardware and software did not indicate any cause for the scram and calibrations resumed with no further problems.

# SECTION IV

Safety-Related Corrective Maintenance

SECTION IV

Safety-related corrective maintenance:

The following are excerpts from the malfunction logbook during the reporting period. The reason for the corrective action taken, in all cases, was to return the failed equipment to its proper operational status.

17 Jan 91 Problem: While performing k-excess, the console scrammed when the Auto mode button was pushed. A "Console Scram Button" message appeared on the CRT.

Solution: System was tested but the malfunction could not be made to reoccur. Reactor operated normally thereafter.

18 Jan 91 Problem: While moving the core from position 700 to position 250, a "Console Scram Button" message appeared on the CRT. The reactor was not at power at the time.

Solution: System was tested but the malfunction could not be made to repeat. Reactor operated normally thereafter.

05 Feb 91 Problem: While raising control rods to begin a steady-state run, the transient rod scrammed while being raised. No message appeared on the CRT , no other rods scrammed, and the transient rod drive did not drive down until the DOWN button was pressed.

Solution: System was tested but the malfunction could not be made to reoccur. Reactor operated normally thereafter.

04 Mar 91 Problem: While preparing for a steady-state run, an NM-1000 high power trip occurred and would not allow the rods to be raised. The CRT scram message could not be cleared.

Solution: NM-1000 scram clearing sequence was performed, but the scram would not clear until a complete console prestart check sequence was run. Reactor operated normally thereafter.

22 Apr 91 Problem: The console would not pass the automatic prestart check sequence due to a "DIS064 Timeout" message on the CRT.

Solution: A defective power supply in the CSC computer expansion chassis was replaced, the system was tested, recalibrated, and returned to full service.

25 Apr 91 Problem: Criticality monitor R-5 alarmed at a steady-state power level of 70 KW, lower than expected.

Solution: The calibration was checked and appeared to have drifted slightly. The monitor was recalibrated with a known source and returned to service. The calibration was checked periodically to ensure no further drift was observed.

06 May 91 Problem: During morning startup procedures, multiple scram messages, DOS error messages, and timeouts occurred.

Solution: The problem was traced to a possibility of excessively high humidity in the control room. The ventilation system was adjusted and when the humidity was lowered below 80% all systems returned to normal.

15 May 91 Problem: The secondary air particulate monitor (CAM) exhibited a very erratic trace while the primary CAM showed no fluctuation in effluent radiation concentration.

Solution: The problem was traced to a faulty chart lamp. The lamp was replaced, the CAM was tested with a known source, and returned to service.

31 May 91 Problem: While raising control rods for k-excess measurements, the three standard control rods scrammed but the rod drives did not drive down. The transient rod did not scram and no message appeared on the CRT.

Solution: A loose relay contact in the NP/NPP high voltage power supply was secured and the reactor operated normally.

10 Jun 91 Problem: A malfunction of the reactor area security system computer caused the computer to not accept keyboard commands and to not respond to alarms. Reactor Facility Director notified as well as AFRRI security and logistics sections.

Solution: The reactor area backup system was turned on so that reactor-related alarms would appear on security monitors at the front desk which is manned at all times. The primary system was repaired by a contractor and returned to full service on 12 July. No loss of security effectiveness occurred while the backup system was operational.

17 Jun 91 Problem: During the daily startup procedures, a large volume of air began escaping around the transient rod anvil. The RFD was notified and an investigation begun.

Solution: The reactor staff disassembled the transient rod system and determined that a teflon seal on the piston had worked loose due to the lack of a required locking pin in the retaining collar. This pin was left out by the contractor during rebuilding of the drive as part of the fuel-follower control rod installation program. A new teflon seal was machined and a new locking pin inserted. Cleaned and lubricated piston and barrel. Removed, inspected, and replaced O-ring seal at bottom of barrel. Reassembled drive and performed rod drop tests and recalibrations as required. Notified General Atomics that locking pin had not been installed during rebuilding of drive assembly at General Atomics factory.

02 Jul 91 Problem: When the console CRT screen was reconfigured after displaying the graph of a pulse, the digital linear power readout showed 2 milliwatts while the CRT bargraph had a scale of 0-1.2 milliwatts.

Solution: True reactor power determined to be 2 milliwatts based on independent monitoring of counts out of fission chamber. As soon as the power was raised enough to require a change of bargraph scale (to 10-120 milliwatts) the system operated normally. General Atomics notified of possible software problem in CRT reconfiguration program.

05 Jul 91 Problem: After completing automatic console prestart check sequence, a "Pulse Power High" scram message would not clear. While repeating prestart sequence in an attempt to clear the scram, the safe and shim rod drives started to drive up. There was no magnet power to the drives and all control rods remained in the core.

Solution: The console was tested and the malfunction could not be made to reoccur. The reactor operated normally thereafter.

09 Jul 91 Problem: None of the console keyboard function keys used for alternate CRT displays (pulse graph, accumulated operator time, etc) would work.

Solution: The console computer was reinitialized and all keys then worked normally.

16 Jul 91 Problem: While the reactor was secured, an "NPP High Voltage Low" scram message appeared on the console CRT screen.

Solution: When the key was inserted into the console the scram message cleared. No cause for the scram could be found and the reactor operated normally.

03 Oct 91 Problem: While closing the ER1 plug door, the door stopped moving and a loud grinding noise was apparent in the gear reduction box.

Solution: The ER2 gear box was moved to ER1 and the door operated with no more problems. The defective gear box was returned to the factory for repairs.

07 Oct 91 Problem: During prestart checks, a malfunction of the DRIVE UP microswitch caused the safe rod drive to drive up instead of down when the reactor was scrammed. The magnet released and the rod itself dropped correctly. When the drive continued past the upper limit, the position indicator belt and flexible wire guide broke.

Solution: The drive was removed; belt, wire guide, and DRIVE UP microswitch replaced; drive assembly inspected, tested, and reinstalled. Rod travel was adjusted, rod drop time checked, and rod worth curves checked.

NOTE: The reactor was out of service for the annual maintenance shutdown period 12-27 September. No malfunctions occurred during that time. The reactor was also taken out of service for installation of the fuel-follower control rods on 8 November. No malfunctions occurred after the reactor returned to steady-state operation on 16 December.

# SECTION V

Facility Changes and Changes to Procedures as Described in the Safety Analysis Report. New Experiments or Tests during the Year.

SECTION V

Changes to the facility and procedures as described in the Safety Analysis Report and new experiments or tests performed during the year are contained in this section.

A. As previously discussed in the Introduction section, three new fuel-follower standard control rods, a new longer air-follower transient control rod, and related equipment were installed during the year. Also, the control rod drive support structure was modified to permit easier access to the transient rod drive mechanism for maintenance. (Attachment E-1)

B. The pool water gamma activity monitoring system was redesigned to replace vacuum tube technology with a more reliable solid state system (Attachment E-2). Also, the primary water conductivity readout modules in the control room were replaced by newer models. The sensors and overall system design remain the same and no changes to the flow of the primary coolant system were made. (Attachment E-3)

C. The air particulate monitor (CAM) automatic damper closure system was modified by adding an override circuit to allow an operator to open the reactor room air dampers while a CAM is alarming. This temporary modification would allow controlled venting of the reactor room should a cladding failure occur when it would be necessary to pull air through the absolute filters, for example, during pulse testing of the new fuel-follower control rods. (Attachment E-4)

D. The steady-state timer originally installed on the new reactor console was a count-down timer. That timer was replaced with a count-up timer to facilitate determination of steady-state run lengths since runs are often terminated based on dosimetry measurements and not at predetermined elapsed times. (Attachment E-5)

E. The lead-acid battery back-up system that previously provided standby power to the criticality monitor and radiation area monitors (RAMs) was replaced by a larger uninterruptible power supply (UPS) necessary for providing more stable current and decreasing maintenance requirements. Replacement of the lead-acid batteries also eliminates a hazardous chemical problem and hydrogen explosion hazard. The installation of these items was approved along with two other UPSs that were installed in 1990. (See Attachment G-5, 1990 Annual Report)

F. Hardware and software modifications were made to the control console to install the new small pulse display option and new pulse mode scram timer (Attachment E-6). The pulse timer allows the operator to adjust the amount of time the transient rod remains up during a pulse as per earlier preapproved console installation plan. The new small pulse display option allows high sensitivity pulse data acquisition and a more readable display of pulse characteristics on the console CRT.

G. There were no new experiments or tests performed during the reporting period that are not encompassed in the Safety Analysis Report. However, all previously approved Routine Reactor Authorizations were replaced by a revised set approved by the RRFSC on 24 September 1991. The revised Authorizations do not include any experiments that were not in the previous authorizations. The revision served primarily to update format and references and to obtain approval from the current RRFSC membership. A set of the Authorizations is at Attachment D.

Attachments E-1 through E-6 are a summary of safety evaluations made for changes not submitted to the NRC pursuant to the provisions of 10 CFR 50.59. Each modification was described and qualified using Administrative Procedure A3, Facility Modification. This procedure utilizes a step-by-step process to document the fact that there were no unreviewed safety questions and no changes required to the Technical Specifications.

# SECTIONS VI through VIII

Summary of Radioactive Effluent Released.

Summary of Radiological Surveys.

Exposures Greater Than 25% of 10 CFR Limits.

SECTION VI

Summary of radioactive effluent released:

A. Liquid Waste - The reactor produced no liquid waste during CY 1991.

B. Gaseous Waste - There were no particulate discharges in CY 1991. The total Ar-41 discharges in CY 1991 were 13.488 Curies.

On a quarterly basis:

| | | |
|---|---|---|
| Jan - Mar 1991 | 4969.9 mCi |
| Apr - Jun 1991 | 3139.5 mCi |
| Jul - Sep 1991 | 4195.4 mCi |
| Oct - Dec 1991 | 1183.4 mCi |

C. Solid Waste - All solid material was transferred to the AFRRI byproduct license; none was disposed of under the R-84 license.

SECTION VII

Environmental radiological surveys:

A. The environmental sampling of soil, water, and plant growth reported radionuclide levels that were not above the normal range. The radionuclides that were detected were those normally expected from natural background and from long-term fallout.

B. The environmental monitoring (dosimetry) program reported the following results for CY 1991.

1. The average background of 19 thermoluminescent dosimeters (TLD) located outside a 15 mile radius from the AFRRI site was determined to be 84.40 ± 1.69 millirem.

2. The average reading of approximately 30 environmental stations located on the AFRRI site was determined to be -0.45 ± 0.31 millirem above background.

3. The single highest environmental station reading was 14.39 ± 6.07 millirem above background. This station is approximately 500 meters from the AFRRI reactor building.

4. The above results are expressed at a 95% confidence level.

C. The in-plant surveys, including analysis of effluent filters, showed no measurable activity (except as reported in this section) in all areas outside the restricted-access areas.

D. There were no special environmental studies conducted during the year.

SECTION VIII

Exposures greater than 25% of 10 CFR 20 limits:

There were no exposures to staff or visitors greater than 25% of 10 CFR 20 limits.

# ATTACHMENT A

10 CFR 50.59 Analysis and
Refueling Plan for Fuel-Follower
Control Rods

Facility Modification Worksheet 1

10 CFR 50.59 Analysis

Proposed Change Install Fuel Follower Control Rods

Submitted by: Capt Forsbacka Date 23 OCT 91

1. Description of change:

Replace current standard control rods with fuel follower control rods (FFCRs). A complete description of the FFCRs is included in the attached report "Maximum Temperature Calculation and Operational Characteristics of Fuel Follower Control Rods for the AFRRI TRIGA Reactor Facility"(AFRRI TR91-1).

2. Reason for change:

FFCRs will be installed to overcome long-term burnup effects and increase the excess reactivity of the reactor core.

3. Verify that the proposed change does not involve a change to the Technical Specifications or produce an unresolved safety issue as specified in 10 CFR 50.59(a)(2). Attach an analysis to show this.

This modification DOES INVOLVE A TECHNICAL SPECIFICATIONS CHANGE. See attached approval from NRC. Full documentation is in Reactor File 605.06.

Analysis attached? Yes X

AFRRI TR91-1 is the safety analysis report regarding FFCR installation and use.

4. The proposed modification constitutes a changes in the facility or an operational procedure as described in the SAR. Describe which (check all that apply).

Procedure X Facility X Experiment ____

5. Specify what sections of the SAR are applicable. In general terms describe the necessary updates to the SAR. Note that this description need not contain the final SAR wording.

The following sections in the SAR will require modifcation to reflect the installation of FFCRs:

- Section 4.9, Fuel Elements
- Section 4.10, Reactor Control Components

6. For facility modifications, specify what testing is to be performed to assure that the systems involved operate in accordance with their design intent.

See attached plan, "AFRRI TRIGA Refueling Plan".

7. Specify associated information.

New drawings are: Attached X (Blueprints of FFCRs are in the reactor blueprint file)
Not required ____

Does a drawing need to be sent to Logistics? Yes ____ No X
Are training materials effected? Yes X No ____
Will any Logs have to be changed? Yes ____ No X
Are other procedures effected? Yes X No ____

List of items affected:

Training Materials:
- New SRO Training Package will need modification to reflect presence of FFCRs in the core.

Maintenance Procedures:
- Annual shutdown checklist will have to be modified because FFCRs will not be brought out of the pool once they have been irradiated.
- Elongatation and lateral bend will have to be measured on the FFCRs.

8. Create an Action Sheet containing a list of associated work specified in item # 7, attach a copy, and submit another to the RFD.

Action Sheet: Submitted X Not Required ____

Reviewed and approved by RFD [signature] Date 23 Oct 91

RRFSC Concurrence ____________________ Date 17 DEC 1991

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

October 8, 1991

Docket No. 50-170

Colonel George W. Irving, III, BSC, USAF
Director
Armed Forces Radiobiology Research Institute
Bethesda, Maryland 20814-5415

Dear Colonel Irving:

SUBJECT: ISSUANCE OF AMENDMENT NO. 21 TO FACILITY OPERATING LICENSE NO. R-84 - ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE (AFRRI)

The Commission has issued the enclosed Amendment No. 21 to Facility Operating License No. R-84 for the AFRRI TRIGA Research Reactor. The amendment consists of changes to the Technical Specifications in response to your submittal of April 30, 1990, as supplemented on December 17, 1990, March 5, 1991, May 17, 1991, August 16, 1991, and September 10, 1991.

The amendment (1) corrects errors in typography and grammar, (2) increases the maximum licensed steady state reactor power to 1100 kilowatts, (3) authorizes installation of fuel follower control rods, (4) clarifies the transfer of Reactor Facility Director (RFD) responsibilities in the absence of the RFD, and (5) allows operational flexibility in performing surveillance testing of the ventilation system for the reactor facility.

Enclosure 2 is a copy of the related Safety Evaluation supporting Amendment No. 21.

Sincerely,

Richard F. Dudley, Jr. for

Alexander Adams, Jr., Project Manager
Non-Power Reactors, Decommissioning and
Environmental Project Directorate
Division of Advanced Reactors
and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 21
2. Safety Evaluation

cc w/enclosures:
See next page

# AFRRI TRIGA Refueling Plan

## Receiving and Storage of FFCRs Prior to Installation

Prior to receiving the FFCRs, the Hot Cell will be prepared for fuel storage. The room will be reasonably free of dust and debris, a gamma detector (criticality monitor) will be in place, a high security lock will be installed, and cradles for holding the fuel will be available.

The FFCRs will enter AFRRI through the shipping and receiving department (LOGS). A memorandum will be prepared (See attachment 1) to instruct LOGS personnel to not open the containers holding the FFCRs. LOGS personnel will be instructed to call RSDR and SHD immediately upon receipt of the FFCRs. SHD will perform a radiological survey in accordance with HPP-0-3 of external radiation and external contamination of the packaging in the loading dock area prior to moving the FFCR containers to the Hot Cell.

Once the FFCRs have been cleared by SHD to be moved to the Hot Cell, RSDR staff will conduct a series of tests within the Hot Cell to determine if any uranium is on the outer surface of the FFCR cladding. The testing methodology is similar to series of measurements outlined in 10 CFR 70.39 which deals with the certification of calibration or reference radiation sources. The following tests will be conducted:

1. Dry wipe test. The entire surface of the FFCRs will be wiped with filter paper using moderate finger pressure. Any radioactivity on the filter paper will be determined by measuring the radiation levels using SHD's counting lab.

2. Wet wipe test. The entire surface of the FFCRs will be wiped with filter paper, moistened with water, using moderate finger pressure. Any radioactivity on the filter paper will be determined by measuring the radiation levels using SHD's counting lab.

3. Water Boil Test. The FFCRs will be completely immersed in boiling water for one hour. The residue obtained by evaporating the water will then be monitored for the presence of uranium using the SHD counting lab.

If measurable quantities of radioactivity are present following any of the above tests, the FFCRs will be thoroughly cleaned and the tests will be repeated. In the event that radioactivity is found after repeated tests and cleanings, the shipment will be rejected and returned to the manufacturer.

Following the cumulation of successful tests showing no contamination of the FFCR exterior cladding with uranium or other radioactive contaminants, the FFCRs may be moved to storage in the Reactor Room.

**Core Preparation**

Prior to shutdown and FFCR installation, neutron activation foils/wires in will be used in ER1 with the bare core configuration at a fixed position one meter from the core. Determine neutron energy spectrum, save data for later comparison with FFCR loaded core. Next, completely unload all fuel from the reactor core. Following the removal of fuel, the standard control rods and transient control rod will be removed. The rod drive support structure will then be modified as required to allow for the ease of transient rod drive maintenance. Next, the new transient rod will be installed.

The FFCR connecting rods will be set up to allow for the removal of FFCRs without bringing them out of the reactor pool. Once the FFCRs have been irradiated, they will become highly radioactive due to their fission product inventories, so it will be important to keep them under water for shielding purposes.

**Installation of FFCRs into AFRRI TRIGA Reactor Core**

Measurements of the required connecting rod lengths will be made by measuring the entire length of the standard rods connected to the barrel. Correcting for differences between the FFCRs and the standard rods, the connecting rod lengths for the FFCRs will be determined. The connecting rods will then be attached to the FFCR and the entire unit will be installed into the reactor core. The connecting rods will then be attached to the barrel assembly to complete the installation of the FFCRs.

Following installation, the FFCRs will centered in their grid locations using the set screws for the barrel assemblies. The goal is to minimize any rubbing of the FFCR as it travels up and down. Once the FFCRs have been centered, drop time tests will be performed to insure compliance the technical specifications.

**Refueling AFRRI TRIGA Reactor Core**

A conservative approach to refueling the reactor core will be taken. These instructions will supplement Reactor Operational Procedure VII. Once the critical loading has been achieved, excess reactivity will be estimated using the transient rod until there is enough excess reactivity to perform rod worth curves. Excess reactivity will be determined after each fuel loading step until the operational configuration is achieved. Care will be taken to ensure that the \$5.00 maximum allowed excess reactivity is not exceeded.

Following Reactor Operational Procedure VII, install the thermocoupled fuel elements and load the B-ring. Place neutron source in source holder and $BF_3$ neutron detectors in F-7 and F-18. Then load elements for grid locations C-1, C-5, C-6, C-9, C-10, C-11, D-14, D-15, E-18, E-19, F-22, and F-23. Loading these elements will allow for the neutronic coupling of the FFCRs and the neutron source. At this point the rods will be withdrawn as described in step 2.a. of the procedure, and the first subcritical multiplication measurements will be taken.

Complete loading the C-Ring, load D-ring elements 2, 6, 8, 12, and 18, and repeat the subcritical multiplication measurements. Complete loading the D-ring, and load E-ring elements 5, 6, 14, 15; perform subcritical multiplication measurements. Load E-ring elements 1, 2, 8, 9, 10, 16, 17, and 24; perform subcritical multiplication measurements. Load E-ring elements 2, 23, 7, 11, 13, and 20; perform subcritical multiplication measurements. Load the following sets of elements and perform subcritical multiplication measurements after each step (note: this load pattern may be modified to accommodate instrumentation or other items that may obstruct fuel loading): E-4 and E-12; E-21 and E-22; F-1 and F-2; F-3 and F-30; F-4 and F-29; F-5 and F-28; F-6 and F-27; F-26 and F-16; F-15 and F-17; F-14 and F-18; F-13 and F-19; F-12 and F-20; F-11 and F-21; F-10; F-9; and F-8. This loading pattern allows for the FFCRs to exercise a high influence over the neutron population while the core is still very subcritical.

**Core Calibration**

Once the core has been loaded to the operational excess reactivity, core calibrations will proceed in the same manner as following an annual shutdown. Differential and integral reactivity worth curves will be generated for each rod in core positions 250, 500, and 750. Up to this point all testing has been done at very low powers, so the fission product inventory in the FFCRs will be low. Since the probability that the FFCR cladding may fail is highest shortly after installation, a series of pulses will be performed to stress the FFCRs before the fission product inventory has much of a chance of building up. During this pulsing operation the water will be closely monitored for any fission fragments. In the event that fission fragments are found in the pool water, all activities will stop, the NRC and GA will be notified, and the leaking element(s) will be found and isolated.

A thermal power calibration will be performed in the usual manner. Next the power coefficient of reactivity curve will be generated followed by the reflection coefficient measurements in positions 250, 500, and 750. The neutron energy spectrum experiment in ER1 and ER2 will be repeated to ensure that the character of the radiation field has not been modified.

**Refueling Checklist**

Pre-shipment

1. Clean Hot Cell.
2. Inspect Hot Cell, ensure area is reasonably dust free.
3. Install high security lock on Hot Cell.
4. Ensure radiation monitor in hot cell is functional.
5. Prepare fuel cradles, install in Hot Cell.
6. Send memoranda to SHD and LOG on receipt of FFCR shipment.

Receipt of shipment

7. SHD performs radiological surveys of exterior of package in accordance with their procedures.
8. FFCR packages are transferred to Hot Cell.
9. Open FFCR shipping packages in Hot Cell.
10. Place FFCRs in prepared cradles.

Hot cell testing

11. Perform dry wipe test.
12. Perform wet wipe test.
13. Perform water boil test.
14. Repeat wet wipe test.
15. If all tests are successful, FFCRs may be moved to the Reactor Room for storage.

Core preparation

16. Use neutron activation wire set from Reactor Experiments to establish base line neutron energy spectrum using Ledney's set up.

17. Unload all fuel from the core in accordance with Procedure VII.

18. Modify the rod drive support structure.

FFCR installation

19. Install new transient rod.

20. Fabricate connecting rods for FFCRs.

22. Install FFCRs, insure that they are not "bottomed out" with they are fully down.

23. Measure FFCRs against FFCR standard, record results.

24. Reinstall FFCRs, install rod drive motors, and center FFCRs in their core grid locations.

25. Perform drop time tests to insure compliance with Technical Specifications.

Core refueling

26. Place neutron source in its holder, and place $BF_3$ or fission detectors in core grid locations F-7 and F-18.

27. Load the thermocoupled elements into core grid locations B-5 and C-6.

28. Complete loading the B-ring and load C-1, C-5, C-6, C-9, C-10, C-11, D-14, D-15, E-18, E-19, F-22, and F-23. Perform subcritical multiplication measurements.

29. Complete loading the C-ring. Load D-ring locations 2, 4, 6, 8, 10, 12, 14, 16, and 18. Perform subcritical multiplication measurements.

30. Complete loading of D-ring, and load E-ring elements 5, 6, 14, and 15. Perform subcritical multiplication measurements.

31. Load E-ring elements 1, 2, 8, 9, 10, 16, 17, and 24. Perform subcritical multiplication measurements.

32. Load E-ring elements 15, 19, F-22, and F-23, place neutron source into its holder, and perform subcritical multiplication measurements.

33. Load E-ring elements 2, 23, 7, 11, 13, and 20. Perform subcritical multiplication measurements.

34. Load E-ring elements 4 and 12, and perform subcritical multiplication measurements.

35. Load E-ring elements 21 and 22, and perform subcritical multiplication measurements.

**** NOTE ****
When critical configuration is achieved,
estimate excess reactivity using the transient
rod. Continue loading until there is enough excess
reactivity to perform a control rod worth measurement
**** ****

36. Load F-ring elements 1 and 2, perform subcritical multiplication/excess reactivity measurements.

37. Load F-ring elements 3 and 30, perform subcritical multiplication/excess reactivity measurements.

38. Load F-ring elements 4 and 29, perform subcritical multiplication/excess reactivity measurements.

39. Load F-ring elements 5 and 28, perform excess reactivity measurements.

40. Load F-ring elements 6 and 27, perform excess reactivity measurements.

41. Load F-ring elements 16 and 26, perform excess reactivity measurements.

42. Load F-ring elements 15 and 17, perform excess reactivity measurements.

43. Load F-ring elements 14 and 18, perform excess reactivity measurements.

**** NOTE ****

Do not exceed $5.00 excess reactivity!

....... Perform thermal power calibration at 500 to insure proper placement of operational channel. Perform rod worth curves for all rods in position 500 when operational loading is achieved

****      ****

44. Load F-ring elements 13 and 19, perform excess reactivity measurements.

45. Load F-ring elements 12 and 20, perform excess reactivity measurements.

46. Load F-ring elements 11 and 21, perform excess reactivity measurements.

47. Load F-ring elements 10, perform excess reactivity measurements.

48. Load F-ring elements 9, perform excess reactivity measurements.

49. Load F-ring elements 8, perform excess reactivity measurements.

Core calibration

50. Install all core instrumentation into their permanent positions.

51. Perform rod worth curves for all rods in core positions 250, 500, and 750.

52. Perform thermal power calibration and set safety chambers.

53. Fire $1.10 pulse, take pool water sample, monitor for fission fragments.

54. Fire $1.20 pulse, take pool water sample, monitor for fission fragments.

55. Fire $1.30 pulse, take pool water sample, monitor for fission fragments.

56. Fire $1.40 pulse, take pool water sample, monitor for fission fragments.

57. Fire $1.50 pulse, take pool water sample, monitor for fission fragments.

58. Fire $1.60 pulse, take pool water sample, monitor for fission fragments.

59. Fire $1.70 pulse, take pool water sample, monitor for fission fragments.

60. Fire $1.80 pulse, take pool water sample, monitor for fission fragments.

61. Fire $1.90 pulse, take pool water sample, monitor for fission fragments.

62. Fire $2.00 pulse, take pool water sample, monitor for fission fragments.

63. Fire $2.00 pulse, take pool water sample, monitor for fission fragments.

64. Fire $2.00 pulse, take pool water sample, monitor for fission fragments.

65. Fire $2.00 pulse, take pool water sample, monitor for fission fragments.

66. Fire $2.00 pulse, take pool water sample, monitor for fission fragments.

67. Fire $2.00 pulse, take pool water sample, monitor for fission fragments.

68. Fire $2.00 pulse, take pool water sample, monitor for fission fragments.

69. Fire $2.00 pulse, take pool water sample, monitor for fission fragments.

70. Generate power coefficient of reactivity curve, take pool water sample and monitor for fission fragments.

71. Fire $2.00 pulse, take pool water sample, monitor for fission fragments.

72. Operate reactor at 1.0 MW for 30 minutes, take water sample and monitor for fission fragments.

73. Fire $2.00 pulse, take pool water sample, monitor for fission fragments.

74. Fire $2.00 pulse, take pool water sample, monitor for fission fragments.

75. Repeat neutron energy spectra measurements in ER1 and ER2.

76. Perform measurements of reflector coefficients in positions 250, 500, and 750.

# Action Sheet

| | | |
|---|---|---|
| **Subject** Installation of FFCRs | **Control Number** | |
| | **Office Symbol** RSDR | **Suspense** N/A |
| | **Date** 23 OCT 91 | |

**Action Required** RFD APPROVAL AND CONCURRENCE

**Memorandum for Record.** (Describe briefly the requirements, background and action taken or recommended. Must be sufficiently detailed to identify the action without recourse to other sources.)

1. Request that we plan to install FFCRs in mid to late November, 1991 in accordance to fuel reloading plan attached.

2. Training materials will require an audit to ensure that they will reflect the reactor configuration with FFCRs installed.

3. Formal maintenance procedures involving FFCR surviellance should be developed in the next few months.

(Continue on plain bond)

**Coordinations** / **Approvals**

| Office | Name | Phone | | Initials | Date |
|---|---|---|---|---|---|
| RSDR | Mark Moore, Reactor Facility Director | 5-1290 | MMoore | | 23Oct91 |
| | | | | | |
| | | | | | |
| | | | Dispatched (Dtg) | | |

Show Additional Coordination on Reverse Side or Continuation Sheet

**Action Officer** (Name, grade, phone and signature)

Matt Forsbacka, Capt USAF, 5-1290 Matt Forsbacka

DNA Form 547
4 Jan 83

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AFRRI TR91-1

# AFRRI TECHNICAL REPORT

# Maximum Temperature Calculation and Operational Characteristics of Fuel Follower Control Rods for the AFRRI TRIGA Reactor Facility

**M. Forsbacka**

**M. Moore**

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ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE
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## Contents

## INTRODUCTION

Operational requirements of the Armed Forces Radiobiology Research Institute (AFRRI) TRIGA reactor facility necessitate the implementation of fuel follower control rods (FFCR's). Fuel follower control rods are like the standard TRIGA control rods as described in section 4.10.1 of the AFRRI TRIGA Safety Analysis Report (SAR) except that they have a fuel-filled follower rather than an air or aluminum follower. The primary purpose of the FFCR's is to offset the long-term effects of fuel burnup.

The Code of Federal Regulations (CFR; Title 10, Part 50.59) requires that modifications of a portion(s) of a licensed facility, as described in the facility SAR, be documented with a written safety analysis. The SAR ensures that all safety issues associated with the implementation of FFCR's have been reviewed. This technical report will show that implementing FFCR's will allow the standard control rods to function in their intended purpose and will restore core reactivity economically. FFCR's have been implemented in approximately a dozen TRIGA reactors and have been used for over 20 years without reported failure.

This report has been submitted to the AFRRI Radiation Facility Safety Committee to ensure that all safety questions have been reviewed before submission to the U.S. Nuclear Regulatory Commission (NRC), as required under 10 CFR 50.59.

## GENERAL DESCRIPTION OF FUEL FOLLOWER CONTROL RODS

The current AFRRI TRIGA standard control rods were installed in 1964. The standard control rod consists of a sealed aluminum tube (0.065 inch thick) approximately 1.25 inches in diameter and 31 inches long. The upper 15.25 inches of the tube contain a compacted borated graphite rod ($B_4C$ with 25-percent free boron or other boron compounds), which functions as a neutron absorber or poison. The lower end of the tube contains a 15.25-inch long and 1.125-inch diameter solid aluminum rod called the aluminum follower. The follower functions as a mechanical guide for the control rod as it is withdrawn from or inserted into the reactor core.

The proposed FFCR's differ from the current standard control rods in the following respects:

- The aluminum cladding is replaced by smooth stainless steel (SS304) cladding with a wall thickness of 0.020 inch. The inner and outer diameters are 1.085 inches and 1.125 inches, respectively.
- The length of the control rod is increased to 37.75 inches; the absorber and fuel follower section are both nominally 15 inches long.
- The outer diameter of the absorber section and the fuel follower are both 1.085 inches.
- The fuel follower has a solid zirconium rod as its central core with an outer diameter of 0.225 inch.

The absorber or poison material of the proposed FFCR's is, however, identical to the standard control rods presently installed.

The fuel contained in the FFCR consists of a fuel-moderator element in which zirconium hydride is homogeneously mixed with partially enriched uranium. The FFCR fuel element contains 12 percent uranium by weight and has a nominal enrichment of 20 percent in the $^{235}U$ isotope. The FFCR fuel element contains about 30.0 grams of $^{235}U$--this is 79% of the $^{235}U$ loading of a standard AFRRI TRIGA fuel element. The nominal hydrogen-to-zirconium ratio in the FFCR fuel element is 1.7 with a range between 1.6 and 1.7. The FFCR fuel element contains no burnable poison. The stainless steel cladding on the FFCR fuel element has a hardness greater than the aluminum control rod guide tubes, so wearing will occur on the guide tubes rather than on the FFCR fuel elements.

## FFCR MAXIMUM FUEL TEMPERATURE CALCULATION

A thermal-hydraulic analysis of the FFCR fuel element to determine the maximum fuel temperature uses the following model:

- The neutron mean free path for neutrons of all energies is smaller than the diameter of the TRIGA fuel rods, so the reactor must be treated as a heterogeneous reactor. Thus, the active volume of the core is taken to be the volume of fuel contained within the reactor core.
- The ratio of power in a fuel element with 12 wt-% uranium versus 8.5 wt-% uranium is 1.21. This is determined by General Atomics design calculations.[1]
- The reactor is operating at a steady-state power level of 1.0 MW, and the heat flux across the fuel element is described by Fourier's law of thermal conduction:[2]

$$q''(r) = -k\nabla T(r) \tag{1}$$

where
$q''(r)$ = heat flux at position $r$
$k$ = thermal conductivity
$T(r)$ = temperature at position $r$

For steady-state heat transport, the heat production rate and the rate of energy loss due to heat transport are equal. This can be generally expressed as

$$q'''(r) = \nabla\cdot q''(r) \tag{2}$$

where
$q'''(r)$ = volumetric heat rate (heat production rate) at position $r$.

Substituting equation (1) into equation (2) yields the time-independent equation of thermal conduction:

$$q'''(r) = -\nabla\cdot k\nabla T(r) \tag{3}$$

Equation (3) is, thus, the second-order ordinary differential equation that must be solved to determine the maximum temperature attained in the fuel portion of the FFCR.

Using this model to determine the maximum fuel temperature divides the analysis into two separate tasks: determining the power density in the FFCR in a D-ring grid position and solving equation (3) for the given power density.

## Power Density in FFCR Fuel Element

The anticipated fuel loading for the AFRRI TRIGA reactor core with FFCR's installed will consist of 77 standard TRIGA fuel elements and the three FFCR fuel elements. Presuming that the control rods are fully withdrawn to achieve a power level of 1.1 MW, the total active fuel volume will be 30,597.9 $cm^3$. Thus, the average power density at 1.1 MW will be 36.0 $W/cm^3$.

The maximum fuel temperature is the important parameter, so only the radial variation of the core centerline power density is considered. To determine the maximum power density in the D-ring location of the FFCR fuel element, the following calculations are made:

For the AFRRI TRIGA, the radial and axial peak-to-average power ratios are 1.55 and 1.30, respectively.[3] Thus the maximum power density (heat rate) will be

$$q'''_{max} = (1.55)(1.30)q'''_{ave} = 72.4\ W/cm^3 \tag{4}$$

To determine $q'''_{D\text{-}ring}$ relative to $q'''_{max}$, it is useful to compute a scaling factor from the gross variation of thermal neutron flux in the radial direction (thermal flux and power density are directly proportional). The normalized radial flux distribution for the AFRRI TRIGA core is best represented by a Bessel function of the first kind of order zero:

$$\phi_{therm} = J_0\left(\frac{2.405r}{R_e}\right) \tag{5}$$

where

$R_e$ = 21.78 cm, the extrapolated core radius

r = 11.99 cm, radial position of D-ring element

and the Bessel function scaling factor is $J_o$ (1.3240) = 0.6074.

The power density for the D-ring is thus computed to be

$$q'''_{D\text{-}ring} = (0.6074)\ q'''_{max} = 44.0\ W/cm^3 \tag{6}$$

Because the FFCR fuel element differs from the standard fuel element in concentration of uranium, the power density in an FFCR fuel element is greater than the power density in a standard fuel element by a factor of 1.21.[1]

Taking the above scaling factor into account, the power density of an FFCR fuel element is found to be

$$q'''_{FFCR} = (1.21)q'''_{D\text{-}ring} = 53.2\ W/cm^3 \quad (7)$$

Note that the calculation of $q'''_{FFCR}$ takes into account the most limiting condition for power peaking in a 12 w/o fuel element versus an 8.5 w/o fuel element. As developed in equation (7), $q'''_{FFCR}$ is still considerably less than the theoretical maximum $q'''$ as determined in equation (4). A less conservative approach would have also accounted for the reduced volume in an FFCR fuel element.

## Maximum Temperature in FFCR Fuel Element

Equation (3) takes the following form for cylindrical geometry with axial and azimuthal symmetry (see Figure 1):

$$\frac{1}{r}\left[\frac{d}{dr}kr\frac{dT}{dr}\right] + q''' = 0 \quad (8)$$

The boundary conditions required that constrain equation (8) are as follows:

$$\frac{dT}{dr} = 0 \text{ at } r = R_i \text{ and } T = T_i \text{ at } r = R_i \quad (9)$$

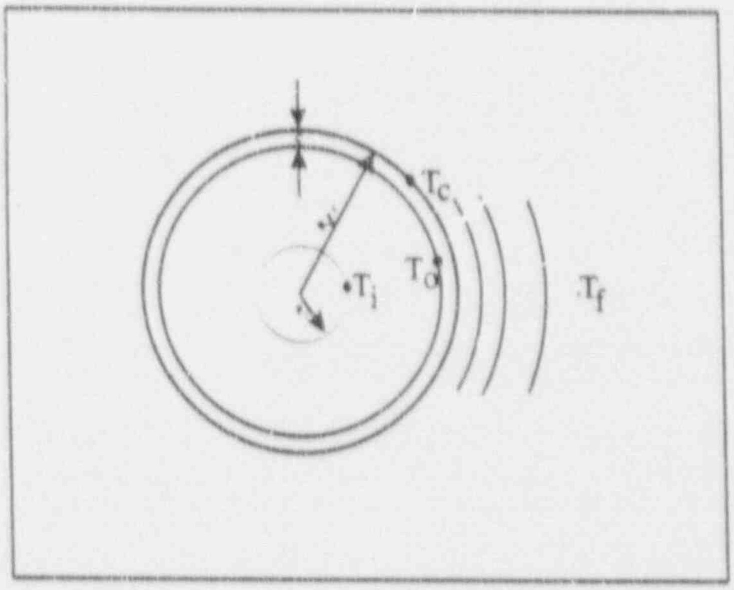


Figure 1. Cross-section of FFCR fuel element.

Integrating equation (8) twice yields a general solution of the form

$$T(r) = -q''' \frac{r^2}{4k_f} + C_1 \ln(r) + C_2 \tag{10}$$

Applying the boundary conditions to solve for the temperature difference between the outer edge of the zirconium rod and the inner surface of the cladding,

$$C_1 = \frac{q''' R_i^2}{2k_f} \tag{11}$$

$$C_2 = T_i - \frac{q''' R_i^2}{2k_f} (\ln(R_i) - \frac{1}{2})$$

$$T_i - T_o = \frac{q''' R_i^2}{4k_f} [(\frac{R_o}{R_i})^2 - 2\ln(\frac{R_o}{R_i}) - 1] \tag{12}$$

To account for the transfer of heat from the fuel through the cladding to the coolant, we must consider the heat conduction between the inner and outer surfaces of the cladding, $q_{clad}$, and the heat conduction from the outer surface of the cladding to the coolant, $q_{fluid}$. We make the assumption that no heat is produced in the cladding or the coolant, so the heat conduction from the outer surface of the fuel, $q_{fuel}$, must be equal to $q_{clad}$ and $q_{fluid}$. The heat conduction leaving the fuel is given by

$$q_{fuel} = \pi(R_o^2 - R_i^2)Lq''' = \pi((R_o/R_i)^2 - 1)LR_i^2 q''' \tag{13}$$

where
L = length of fuel element

$$q_{clad} = -k_c A \left.\frac{dT}{dr}\right|_{clad} = \frac{2\pi k_c L(T_o - T_c)}{\ln(\frac{R_o + c}{R_o})} \tag{14}$$

$$q_{fluid} = hA(T_c - T_f) = 2\pi(R_o + c)Lh(T_c - T_f)$$

Note that the area, A, used in computing $q_{clad}$ is the logarithmic mean area of the cladding. Recall that

$$q_{fuel} = q_{clad} = q_{fluid} \tag{15}$$

So we can solve for the temperature differences in the above equations in terms of q''':

$$T_o - T_c = \ln\left(\frac{R_o + c}{R_o}\right)\left[\left(\frac{R_o}{R_i}\right)^2 - 1\right]\frac{R_i^{\,2}}{2}q'''$$

$$T_c - T_f = \frac{1}{(R_o+c)h}\left[\left(\frac{R_o}{R_i}\right)^2 - 1\right]\frac{R_i^{\,2}}{2}q''' \qquad (16)$$

Adding equations (16) with equation (12) and solving for $T_i$ gives us the expression for the maximum temperature in the FFCR.

$$T_i = T_f + \frac{q'''R_i^{\,2}}{4k_f}\left[\left(\frac{R_o}{R_i}\right)^2 - 2\ln\left(\frac{R_o}{R_i}\right) - 1\right] + \frac{q'''R_i^{\,2}}{2}\left[\left[\left(\frac{R_o}{R_i}\right)^2 - 1\right]\left[\frac{1}{k_c}\ln\left(\frac{R_o + c}{R_o}\right) + \frac{1}{h(R_o + c)}\right]\right] \qquad (17)$$

where
$T_i$ = maximum fuel temperature
$T_f$ = bulk coolant temperature
$R_o$ = 1.38 cm (radius of FFCR fuel element)
$R_i$ = 0.286 cm (radius of Zr rod)
c = 0.051 cm (cladding thickness)
$k_f$ = 0.18 W/cm-°C (thermal conductivity of UZrH)[4]
$k_c$ = 0.138 W/cm-°C (thermal conductivity of SS304)[2]
h = 1.339 W/cm$^2$-°C (free convective heat transfer coefficient of water)
q''' = 53.2 W/cm$^3$ (from equation (7))

Note that the free convective heat transfer coefficient, h, was an experimentally derived quantity. The method by which h was determined is presented in Appendix A. Solving equation (17) using a volumetric heat rate of 53.2 W/cm$^3$ and a bulk water temperature of 48.6°C (the conditions at which h was determined) yields a maximum fuel temperature of 210.2°C. The maximum temperature achieved in the FFCR is nearly 180°C less than the normal temperature of 390°C in a standard fuel element in the B-ring during a 1.0 MW steady-state power operation.

### Fuel Temperature in Pulse Mode Operation

The Nordheim-Fuchs model predicts the maximum fuel temperature achieved in a pulse mode operation.[5] The fundamental assumptions of this model are as follows:

- The neutron flux in the reactor is separated into a spatial component (shape factor) and a time-dependent component (amplitude factor), such that

$$\Phi(r,t) = vn(t)\Psi(r) \qquad (18)$$

where
$v$ = neutron velocity
$n(t)$ = neutron density (amplitude factor), proportional to power
$\Psi(r)$ = shape factor

The shape factor is assumed to remain constant during a pulse. This is called the point-reactor model.

- The production of delayed neutrons and the effects of source neutrons are neglected.

- The pulse from a thermodynamic standpoint is adiabatic, so

$$\frac{dT}{dt} = Kn(t) \tag{19}$$

where
$T$ = fuel temperature
$K$ = reciprocal of heat capacity

From the first and second assumptions we can write the time-dependent neutron density as

$$\frac{dn}{dt} = \frac{\rho - \beta}{\ell} n \tag{20}$$

where
$\ell$ = mean lifetime of neutrons in the reactor
$\beta$ = delayed neutron fraction
$\rho$ = reactivity

To account for a step insertion of reactivity, we can write

$$\rho = \rho_o - \alpha \Delta T \tag{21}$$

where
$\alpha$ = negative of the temperature coefficient of reactivity
$\rho_o$ = step insertion of reactivity

Taking the derivative with respect to time of the above equation and substituting the result from equation (18),

$$\frac{d\rho}{dt} = -\alpha K n \tag{22}$$

Applying the chain rule to equation (20) yields

$$\frac{dn}{d\rho} = -\frac{(\rho - \beta)}{\alpha K 1} \tag{23}$$

Integrating equation (23) and solving for the constant of integration gives us the result

$$n = \frac{1}{2\alpha K l} \left[ (\rho_o - \beta)^2 - (\rho - \beta)^2 \right] \tag{24}$$

The pulse is terminated when n becomes negligibly small. This occurs when

$$\rho = 2\beta - \rho_o \tag{25}$$

Equation (25) gives us the condition for the total energy release from the pulse, which manifests itself as a temperature rise in the fuel element when it is substituted into equation (21).

$$\Delta T_{core,ave} = \frac{2(\rho_o - \beta)}{\alpha} \tag{26}$$

Calculations by General Atomics show that a complete core of 12 w/o fuel would have a temperature coefficient of reactivity, $\alpha$, that is 75% of the value for an 8.5 w/o fueled core.[1] The value for $\alpha_{8.5\ w/o}$ is taken to be -\$0.0118/°C. (This value was experimentally verified with a series of 23 pulses ranging from \$1.30 to \$2.00 that resulted in an average $\alpha_{8.5\ w/o}$ of -\$0.0128/°C--within 8.5% of the published value.) The effect of adding three 12 w/o FFCR's would, however, have a negligible effect on the overall temperature coefficient of reactivity for the entire core.[1]

Applying the Nordheim-Fuchs model for self-limiting power excursions, we can determine the maximum average increase in temperature for the entire core using equation (26). For a maximum allowed reactor pulse with a \$4.00 step insertion of reactivity, the maximum attained average temperature rise is calculated to be 333°C. Applying the power-peaking factors from the previous section, the maximum calculated temperature rise in an FFCR would be $1.48 \cdot \Delta T_{core,ave}$; the temperature rise for an FFCR for a \$4.00 pulse is calculated to be 493°C. Assuming an initial temperature of 25°C, the maximum temperature value would be 518°C. Note that even in the limiting case, neither the technical specification safety limit of 1000°C nor the limiting safety system setting of 600°C is violated.

## FFCR OPERATIONAL CHARACTERISTICS

FFCR's are a standard design offered as a stock item by General Atomics and have been used in several TRIGA reactors for over 20 years. FFCR's are currently implemented in approximately a dozen TRIGA reactors. There has been no reported evidence of fuel failure as a result of FFCR use in the United States. The operational issues to be resolved are the effects of burnup on the FFCR and the influence of FFCR's on the temperature coefficients of reactivity, shut-down margin, and rod worth.

FFCR control rod worth curves will be generated the same way that standard control rod worth curves are. Since the poison section of the FFCR will be the same as that of the currently installed standard control rods, the worth of the poison section of the FFCR will be that of the currently installed control rods. Measurements made by AFRRI reactor staff of control rod worth of the currently installed standard control rods yielded a nominal rod worth of $1.90. The fuel follower is expected to add at least $0.70 of reactivity[6] when the control rod is fully withdrawn, so the total rod worth for an FFCR is estimated to be $2.60. The transient control rod and its follower have been measured and have a total nominal worth of $4.01. The shutdown margin, as established by ANS/ANSI 15.1, is computed as follows:

| | | |
|---|---|---|
| Total rod worth | | $11.81 |
| $k_{excess}$ (maximum) | - | $ 5.00 |
| | | $ 6.81 |
| Worth of TRANS rod | - | $ 4.01 |
| Shutdown margin | | $ 2.80 |

The shutdown margin with the most reactive control rod removed from the reactor is $2.80--well in excess of $0.50 minimum allowed value.

Once operational rod worth curves are established and power monitoring channels have been calibrated by the thermal power calibration method, power coefficient of reactivity curves will be generated. The issues regarding the measurement of shut-down margin and excess reactivity are addressed in Appendix B, Reactor Core Loading and Unloading.

Structural changes in the FFCR's will be monitored on an annual basis as part of the annual shutdown and maintenance. Specific effects to be monitored are the elongation and lateral bending of the fuel. FFCR fuel elements that have an elongation greater than 0.100 inch or a lateral bend greater than 0.0625 inch will be removed from service.

## CONCLUSION

The analysis in this report shows that installing FFCR's in the AFRRI TRIGA reactor core will not result in an unsafe condition or violation of technical specifications. The primary parameter of interest, the maximum fuel temperature, was computed to be 210°C in the limiting case for steady-state operation and 518°C in the limiting case for pulse operation. Operational issues regarding maximum excess reactivity, shutdown margin, and burnup have also been addressed, and it has been determined that sufficient surveillance capabilities exist to prevent any unsafe or illegal condition.

## REFERENCES

1. General Atomics, letter to M. Moore on fuel follower control rods, 28 October 1988.

2. El-Wakil, M. M., *Nuclear Heat Transport*, The American Nuclear Society, Lagrange Park, IL, 1978.

3. Defense Atomic Support Agency, *AFRRI/USAEC Facility License R-84, Complete with Applications and Amendments*, Bethesda, MD, 1962.

4. Wallace, W. P., and Simnad, M. T., *Metallurgy of TRIGA Fuel Elements*, GA-1949, General Atomics, San Diego, CA, 1961.

5. Hetrick, D. L., *Dynamics of Nuclear Reactors*, The University of Chicago Press, 1971.

6. DNA Contract DNA001-89-R-0030 to General Atomics for fuel follower control rod construction.

7. Jaluria, Y., *Natural Convection Heat and Mass Transfer*, Pergamon Press, 1980.

# APPENDIX A: DETERMINATION OF FREE CONVECTIVE HEAT TRANSFER COEFFICIENT

## Introduction

We can measure the bulk water temperature within the AFRRI TRIGA core to determine the average free convective heat transfer coefficient of the cooling water. This experiment involves inserting a temperature-measuring probe between the B- and C-ring fuel elements while the reactor is operating at a steady-state power level of 1.0 MW and measuring the water temperature at various axial positions. Once the bulk water temperature has been determined, Newton's law of cooling can be used to calculate the average free convective heat transfer coefficient.

## Experimental Apparatus and Procedure

The equipment used in this experiment consists of two approximately 18-foot lengths of chromal-alumel thermocouple wires fused together at one end, encased in a 16-foot-long, 0.375-inch-diameter aluminum (Al) tube, and the thermocouple display readout on the AFRRI computerized reactor control console (Figure A-1).

The potential difference generated at the thermocouple junction as the water is heated by the reactor is amplified and displayed by the thermocouple circuitry in the AFRRI computerized reactor control console. The thermocouple is initially inserted into the core to correspond to position I. The thermocouple resides in each region for several minutes to allow it to attain thermal equilibrium. Once thermal equilibrium is attained, ten temperature readings are taken at 10-second intervals. After each temperature measurement, the thermocouple is withdrawn to the next position, and the temperature measuring procedure is repeated.

Figure A-2 shows that the temperature is measured in five axial positions: (I) 3 inches below midpoint (14 inches of thermocouple wire inserted into the core); (II) midpoint in axial dimension; (III) halfway between midpoint and bottom of graphite slug; (IV) at top of fuel region; (V) 1.5 inches above top of fuel region.

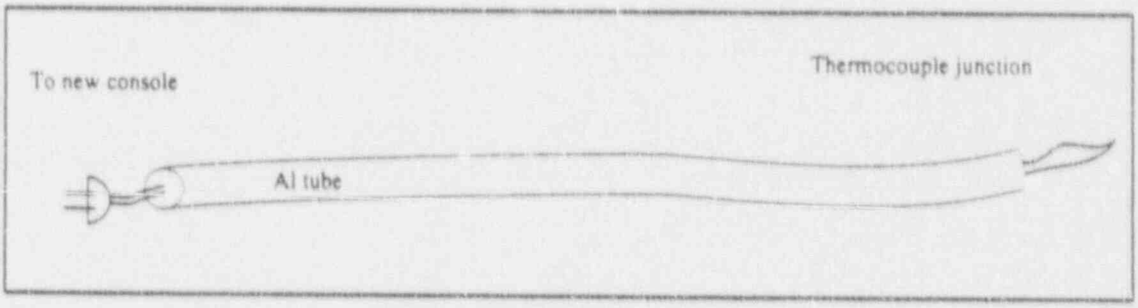


Figure A-1. Experimental apparatus.

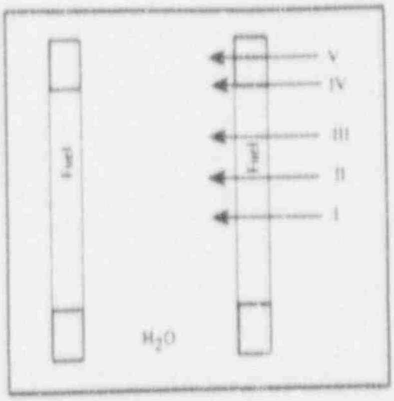


Figure A-2. Axial measuring points.

## Safety Considerations

There are two safety considerations associated with this experiment: radiation streaming and an unintentional positive change in reactivity if the thermocouple wires are rapidly withdrawn from the reactor core while it is at power. Radiation streaming is avoided by flooding the aluminum tube with water and bending the tube so that it is at an angle not normal to the top of the core. The thermocouple wire displaces only 0.043 inch$^3$ of water when it is fully inserted in the core, so using the void coefficient of reactivity, the thermocouple wire represents a negative reactivity insertion of only 0.001 cents. If we were to estimate conservatively that the thermocouple wire had the same neutron-absorbing properties of a control rod, the maximum negative reactivity would be only 0.01 cents. Thus, there is no possibility of a reactivity accident associated with the apparatus used in this experiment.

## Data

Table A-1 summarizes the data gathered during a 1.0 MW steady-state run of the AFRRI TRIGA reactor. The variation in the temperature measurements is most likely due to variance in the radial position of the temperature probe in the channel.

Table A-1. Bulk Water Temperature at Each Axial Position in the AFRRI TRIGA Reactor Core

| Axial position | Inlet temp (°C) | Measured core bulk water temp (°C) |
|---|---|---|
| I | 22 | 72.9 |
| II | 24 | 65.0 |
| III | 25 | 48.6 |
| IV | 26 | 51.6 |
| V | 27 | 59.7 |

## Analysis and Conclusion

The purpose of this experiment is to determine the bulk water temperature within the core shroud; thus, it is the lowest measured value of the water temperature that is sought. Figure A-3 illustrates the temperature variation within a cooling channel.

Table A-1 shows that the measured value that most closely represents the bulk water temperature within the core shroud is 48.6°C.

The free convective heat transfer coefficient, h, is found by solving equation (8) for boundary conditions given by a standard TRIGA fuel element. Equation (A-1) gives the solution in terms of h.

$$h = \left(\frac{1}{r_o + c_o}\right)\left[\frac{(T_i - T_f) - \dfrac{q'''r_i^2}{4k_f}\left[\left(\dfrac{r_o}{r_i}\right)^2 - 2\ln\left(\dfrac{r_o}{r_i}\right) - 1\right]}{\dfrac{q'''r_i^2}{2}\left[\left(\dfrac{r_o}{r_i}\right)^2 - 1\right]} - \frac{1}{k_c}\ln\left(\frac{r_o + c_o}{r_o}\right)\right]^{-1} \quad \text{(A-1)}$$

where

$T_i$ = measured fuel temperature at 1.0 MW
$T_f$ = measured bulk coolant temperature in the core
$r_o$ = fuel outer radius, 1.816 cm
$r_i$ = fuel inner radius, 0.229 cm
$c_o$ = cladding thickness, 0.051 cm
$k_f$ = thermal conductivity of fuel, 0.18 W/cm-°C
$k_c$ = thermal conductivity of clad, 0.138 W/cm-°C
$q'''$ = volumetric heat rate.

The measured fuel temperature in the B-ring at 1.0 MW steady-state power level is 390°C, and the calculated volumetric heat rate is 65.9 W/cm$^3$. Using the measured value of the bulk coolant temperature of 48.6°C yields a value of 1.339 W/C°-cm$^2$ for the free convective heat transfer coefficient.

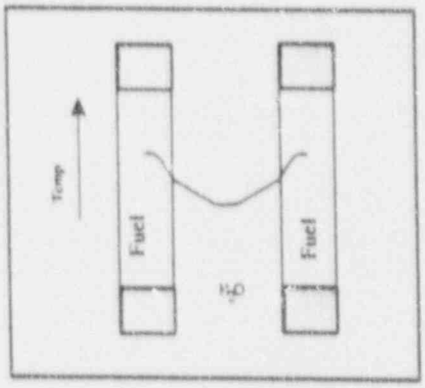


Figure A-3. Temperature variation within a cooling channel.

Newton's law of cooling expresses the linear relationship between the heat transfer rate, Q, and the temperature difference between the clad surface temperature, $T_c$, and the bulk water temperature, $T_f$, as

$$Q = hA(T_f - T_c)$$

where h is the overall convective heat transfer coefficient and A is the area of the fuel element.[7] The value of h determined for the AFRRI TRIGA is unique in that it takes into account the flow configuration, fluid properties, and the dimensions of the fuel elements. Assuming that the dependence of Q on the temperature difference $T_f - T_c$ is roughly linear, then the value of h, computed using data from B-ring elements with their higher heat transfer rate and local temperature difference, will be close to the value for h for D-ring positions.

# APPENDIX B: REACTOR CORE LOADING AND UNLOADING

## General

Loading and unloading of the reactor core shall be performed under the supervision of the Reactor Facility Director or the Reactor Operations Supervisor.

## Specific

1. Setup

a. Ensure that at least one nuclear instrumentation channel is operational.

b. Ensure that the source is in the core.

c. Ensure that an operator monitors the reactor console during all fuel movements.

d. Check new FFCR's before insertion into the core; this includes cleaning, visual inspection, and length and bow measurements.

e. Install all control rods.

f. If irradiated fuel elements are to be removed unshielded from the pool, obtain a Special Work Permit (SWP) from the Safety and Health Department (SHD); do not remove fuel elements with a power history (greater than 1 KW) in the previous 2 weeks from the reactor pool.

2. Core Loading

a. After each step of fuel movement perform the following:

(1) Record detector readings.

(2) Withdraw control rods 50%; record readings.

(3) Withdraw control rods 100%; record readings.

(4) Calculate 1/M.

(5) Plot 1/M versus number of elements (and total mass of $^{235}U$).

(6) Predict critical loading.

(7) Insert ALL rods; continue to next step.

b. Load elements in the following order:

(1) Load the B-ring and C-ring thermocouple elements.

(2) Connect thermocouple outputs to reactor control console display.

(3) Install any other thermocouple elements.

(4) Complete loading of B- and C-ring elements (total of 18 standard elements plus 3 FFCR's).

(5) Load D-ring (total of 33 standard elements plus 3 FFCR's).

(6) Load the following E-ring elements in order:

16, 17, 18, 20, 6, 8, 9, 10 (total of 41 standard elements plus 3 FFCR's).

(7) Complete the E-ring by loading the following elements in order:

15, 21, 11, 5, 14, 22, 4, 12, 13, 1 (total of 51 standard elements plus 3 FFCR's).

(8) Load the following F-ring elements in two elements per step until criticality is achieved, using the following loading order:

22, 23, 24, 21, 20, 25, 26, 27, 28, 29, 30, 1, 2, 3, 4, 5, 19, 18, 17, 16, 15, 14, 13, 6, 12, 7, 11, 8, 10, 9.

Once criticality has been achieved, perform control rod worth measurements at core position 500 by rod drop technique. Calculate shutdown margin (SDM):

$$SDM = \text{total control rod worth} - K_{excess} - \text{TRANS rod worth}$$

(9) Load core to \$2.00 excess reactivity by loading two elements per step using the loading order in instruction 8.

(10) Verify control rod worth using rod drop techniques; calculate SDM.

(11) Load the core to achieve a $K_{excess}$ that will allow calibration of the TRANS rod based on the last available worth curve of the TRANS rod (approximately \$4.00). Calculate the reactivity value of each element as it is added.

(12) Calibrate all control rods.

(13) Calculate SDM.

(14) Estimate $K_{excess}$ with a fully loaded core (must not exceed \$5.00).

(15) Load core to fully operational load using loading order in instruction 8, and recalibrate all control rods. Calculate SDM.

(16) Adjust the core loading pattern to meet operations requirements if necessary. Recalibrate all control rods. Calculate SDM.

3. Core Unloading

a. Unload the reactor core starting with the F-ring and ending with the B-ring.

b. Remove the fuel elements individually from the reactor core, identify them by serial number, and place them in the fuel storage racks or a shipping cask.

c. If elements are to be loaded into a shipping cask, clean the cask completely, and check for radiological contamination before placing the cask in or near the pool. Load cask in accordance with procedures specific to the cask.

d. Once the cask is loaded, perform an air sample and survey; check temperature and pressure inside cask, if necessary.

e. If elements are placed in temporary storage away from core monitoring, ensure that criticality monitoring in accordance with 10 CFR 70 is in place.

# ATTACHMENT B

Current Reactor Administrative and Operational Procedures

# PROCEDURES

For

# The AFRRI Reactor Facility

# TABLE of CONTENTS

This procedure has been approved by the Reactor Facility Director

APPROVED

29 May 91

Reactor Facility Director Date

Reviewed by the Reactor Staff

| | |
|---|---|
| T Wright | 29 May 91 |
| Wright | Date |
| R George | 29 may 91 |
| George | Date |
| | 29 May 91 |
| Forsbacka | Date |
| | 29 May 91 |
| Spence | Date |
| Michael Laughery | 29 May 91 |
| Laughery | Date |
| John Nguyen | 29 may 91 |
| Nguyen | Date |
| | 29 MAY 91 |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991

Date

# FITNESS FOR DUTY

## GENERAL

The AFRRI Reactor Facility is a drug-free work-place. The use of illicit drugs by any RSDR staff member is prohibited. Personnel using over-the-counter or prescription drugs which cause drowsiness or otherwise alter one's state of consciousness will not be permitted to operate the AFRRI TRIGA reactor. In addition, reactor operators, operators-in-training, and management will be monitored for attitude and behavioral changes that may impact an individual's reliability.

## SPECIFIC

1. RSDR staff members shall participate in drug-free awareness programs sponsored by AFRRI. Military and civilian staff members shall submit to drug screening programs conducted by their respective services. If a staff member's drug screening test yields a positive result, that staff member shall not be permitted to operate the reactor pending verification of the test. The Reactor Facility Director (RFD) is required to ensure that the cutoff levels for alcohol or controlled substances as established in 10 CFR 26 are not exceeded by NRC licensed personnel. Any staff member determined to be a drug user will be terminated.

2. Personnel are instructed to inform their physician of their job description and requirements prior to being issued a prescription medication. They are instructed to inquire about any medication side effects expected and the physician's opinion regarding interference with safe job performance. This information shall be relayed to RFD as soon as possible.

Personnel are encouraged to minimize their use of non-prescription over-the-counter drugs for self-medication purposes. Specifically, sedatives, cough and cold preparations, appetite suppressants, and pain relievers have central nervous system side effects. If these medications are used in any quantity, an operator must inform the RFD or ROS and be relieved from operating on that day or until any side effects have resolved once the medication has been discontinued.

Personnel are instructed to read the information in the Physician's Desk Reference (PDR) concerning medication they are taking. If the PDR indicates that the medication will adversely affect an operator's ability to safely perform his/her duties, he/she must inform the RFD or Reactor Operations Supervisor (ROS) that he/she must be relieved from operating on that day.

3. The RFD shall continuously monitor the reliability of individuals under his/her command by the following criteria:

- Any court-martial or civil conviction of a serious nature. Minor traffic violations are not a consideration.
- Negligence or delinquency in duty performance.
- Significant mental or character traits, or aberrant behavior, sustained by medical authority, that might affect the reliable performance of duties.
- Behavior patterns that show or suggest a contemptuous attitude toward the law or regulations
- Drug abuse or alcohol misuse.
- Poor attitude, lack of motivation toward assigned duties, or financial irresponsibility.

The RFD will be observed by his superiors to ensure his/her adherence to reliability criteria. Individuals who exhibit any of the listed behaviors or actions will be removed from licensed activities.

APPROVED

This procedure has been approved by the Reactor Facility Director

28 May 91
Reactor Facility Director Date

27 May 91
Chairman, Safety and Health Department Date

Reviewed by the Reactor Staff

Wright 5/29/91
Wright Date

5/29/91
George Date

29 May 91
Forsbacka Date

Spence 29 May 91
Spence Date

McLaughery 29 May 91
Laughery Date

John T. Nguyen 29 May 91
Nguyen Date

29 May 91
Owens Date

Date

Date

Reviewed by RRFSC 24 SEP 1991
Date

# PERSONNEL PASSAGE THROUGH THE PREP AREA

## GENERAL

Access to the Reactor Prep Area is limited to personnel who are granted access in accordance with the reactor physical security plan and Operational Procedure 1. The Reactor Facility Director is responsible for maintaining an unescorted access roster for the Reactor Prep Area and for providing a Prep Area briefing to all persons listed on that roster. This administrative procedure does not recapitulate the operational procedure. Rather, it presents specific guidelines for Reactor Prep Area passage for individuals who are authorized access.

## SPECIFIC

There are three specific conditions when Reactor Prep Area passage is considered appropriate.

1. ROUTINE PASSAGE: (EXPOSURE ROOM DOORS CLOSED)

   a. Personnel who are authorized unescorted access to the Reactor Prep Area may pass through the Prep Area but are required to radiologically frisk themselves when exiting the Prep Area.

   b. Personnel who are being escorted through the Reactor Prep Area may pass through the Prep Area with their escort. Both individuals must radiologically frisk themselves when exiting the Prep Area.

   c. Only appropriate personal dosimetry that has been issued by SHD or AFRRI Security is required for routine passage through the Reactor Prep Area. There is no requirement to wear a pocket chamber in addition to the AFRRI issued TLD for routine passage.

2. CONTROLLED PASSAGE: (EXPOSURE ROOM DOOR OPEN)

   a. Only personnel associated with the experiment/operation being performed are normally authorized access to the Reactor Prep Area during an exposure room opening. These personnel will be required to wear the AFRRI TLD dosimeter (issued with their AFRRI badge), pocket chamber (if dose rate at face of door is 2 mr/hr or more), and the following special dosimeters if they enter the exposure room:

      AFRRI wrist dosimeter or finger ring dosimeter.

All personnel who enter an exposure room will log their pocket chamber reading in the pocket chamber log prior to entering the room for the first time that day and will enter the final pocket chamber reading following their exit from the exposure room at the end of the day. Each individual who enters an exposure room is responsible for monitoring his accumulated dose throughout the day to ensure the he/she does not exceed the AFRRI daily exposure limits of 50 mR/day or 100 mR/week. Extremity dosimetry is required only if work is to be performed on an experimental array or within 1 meter of the core projection.
b. Personnel authorized unescorted access to the Reactor Prep Area or personnel being escorted through the Reactor Prep Area may pass through the Prep Area when an exposure room is open with permission from the Reactor Staff person in charge during the opening if the following guidelines are met:

(1) The person desiring passage must stop just inside the Prep Area door upon seeing that an exposure room door is open and request permission from the Reactor Staff member in charge before proceeding. At that time, the Reactor Staff and Safety Staff members monitoring the Exposure Room opening will determine if the radiation level at the outside entrance to the exposure room in direct line of sight with the core projection is less than or equal to 2 mR/hr. If this reading is less than or equal to 2 mR/hr, the Reactor Staff Member may grant passage permission. If the reading is greater than 2 mR/hr, passage will be denied.

(2) All personnel who pass through the Reactor Prep Area must radiologically frisk themselves before exiting the Prep Area. There is no requirement for these individuals to wear a pocket chamber just to pass through the Prep Area.

3. OPEN PASSAGE: (NON-ROUTINE)

The prep area may be opened for passage by personnel traveling between buildings at AFRRI when maintenance is being performed on the normal connecting hallway. This is not a routine occurrence and warrants written approval from the Reactor Facility Director with concurrence from the Chairman, Safety and Health Department. In addition, the Prep Area must be monitored at all times by appropriately trained personnel. Prior to the opening of the Reactor Prep Area for open passage, the Safety and Health Department shall conduct a radiological survey of the area and certify that no radiation areas exist within the Prep Area and that the non-painted areas of the Prep Area floor are free of contamination. There is no requirement for personnel who pass through the Prep Area to wear pocket chambers or frisk themselves during periods of open passage. Finally, open passage will be suspended during exposure room openings.

This procedure has been approved by the Reactor Facility Director

APPROVED

29 May 91

Reactor Facility Director Date

Reviewed by the Reactor Staff

| | |
|---|---|
| 5/24/91 | |
| Wright | Date |
| 5-24-91 | |
| George | Date |
| 24 May 91 | |
| Forsbacka | Date |
| 29 May 91 | |
| Spence | Date |
| 29 May 91 | |
| Laughery | Date |
| John T. Nguyen 29 May 91 | |
| Nguye. | Date |
| 29 May 91 | |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991

Date

# FACILITY MODIFICATION

## GENERAL

Changes to the Reactor Facility and operational procedures must comply with requirements specified in the Reactor License, and 10 CFR 50.59. It is required that modifications to the facility or procedures as described in the Safety Analysis Report (SAR) be documented with a written safety analysis. Under 10 CFR 50.59, a licensee may make changes to the facility provided there are no changes made to the Technical Specifications, there are no unreviewed safety questions, and that a proper safety analysis is carried out, documented, and reviewed.

Applicability:

- The Facility Modification Procedure applies to proposed facility changes or changes in the operating procedures.
- The referenced procedure will not cover routine replacement of parts or components with equivalent parts or components.

## DESCRIPTION

This administrative procedure consists of these instructions, the Facility Modification Worksheet Guide, and two worksheets to facilitate a 10 CFR 50.59 review of modifications and to determine if a detailed safety analysis is necessary. The instructions in the Facility Modification Worksheet Guide are used determine which worksheet must be completed for the modification. One of three conclusions regarding the proposed facility modification will be reached:

1. The modification requires prior approval or a license amendment from the USNRC,

2. The modification may be made according to the provisions of 10 CFR 50.59(a)(1) (Facility Modification Worksheet # 1), or

3. The modification does not require a 10 CFR 50.59 safety analysis (Facility Modification Worksheet # 2).

1. Technical Specification Change: If the proposed modification requires a changes in the Technical Specifications, a license amendment is required prior to making the change. NRC approval is required; do not implement the change without this approval.

2. Unreviewed Safety Question: If an unreviewed safety question is created by the proposed change as defined in 10 CFR 50.59(a)(2) such that the change increases the probability of occurrence or severity of an accident described in the SAR, can malfunction in a manner that can cause an accident of a different type than described in the SAR or can decrease safety margins as defined in Technical Specifications, then NRC approval is required. Do not implement the change without this approval.

2. If the proposed modification makes a change in the facility as described in the SAR or changes a procedure as described in the SAR, the change can be performed under a 10 CFR 50.59 analysis with a safety review, if there are no unreviewed safety issues (10 CFR 50.59(a)(2)). The change may be made following a review by the RRFSC. Go to Facility Modification Worksheet # 1.

3. If the proposed modification does not make a change to the facility as described in the SAR or to a procedure as described in the SAR and does not pose an unreviewed safety issue, a 10 CFR 50.59 analysis is not required. Go to Facility Modification Worksheet # 2.

Facility Modification Worksheet 1

10 CFR 50.59 Analysis

Proposed Change ______________________________

______________________________

Submitted by: ______________________________ Date ____________

1. Description of change:

2. Reason for change:

3. Verify that the proposed change does not involve a change to the Technical Specifications or produce an unresolved safety issue as specified in 10 CFR 50.59(a)(2). Attach an analysis to show this.

Analysis attached? Yes ____

4. The proposed modification constitutes a changes in the facility or an operational procedure as described in the SAR. Describe which (check all that apply).

Procedure ____ Facility ____ Experiment ____

5. Specify what sections of the SAR are applicable. In general terms describe the necessary updates to the SAR. Note that this description need not contain the final SAR wording.

6. For facility modifications, specify what testing is to be performed to assure that the systems involved operate in accordance with their design intent.

7. Specify associated information.

New drawings are: Attached ____
Not required ____

Does a drawing need to be sent to Logistics? Yes ____ No ____
Are training materials effected? Yes ____ No ____
Will any Logs have to be changed? Yes ____ No ____
Are other procedures effected? Yes ____ No ____

List of items affected:

8. Create an Action Sheet containing a list of associated work specified in item # 7, attach a copy, and submit another to the RFD.

Action Sheet: Submitted ____ Not Required ____

Reviewed and approved by RFD ______________________ Date __________

RRFSC Concurrence ____________________________ Date __________

Facility Modification Worksheet 2

No 10 CFR 50.59 Analysis Required

Proposed Change ______________________________________________

______________________________________________

Modification to: Procedure ____ Facility ____ Experiment ____

Submitted by: ______________________________ Date __________

1.Description of change:

2. Verify that the proposed change does not involve a change to the Technical Specifications, the facility as described in the SAR, or procedures as described in the SAR, and does not produce an unresolved safety question as defined in 10 CFR 50.59(a)(2).

3. If change involves a facility modification, attach a drawing if appropriate. If structural facility drawings need updating, forward a copy of changes necessary to Logistics.

4. Determine what other procedures, logs, or training material may be affected and record below.

5. List of associated drawings, procedures, logs, or other materials to be changed:

6. Create an Action Sheet containing the list of associated work specified above, attach a copy, and submit it to the RFD.

Action Sheet: Submitted ____ Not Required ____

Reviewed and approved by RFD ____________________ Date ____________

RRFSC Notified ______________________________ Date ____________

This procedure has been approved by the Reactor Facility Director

M. L. Moore 16 May 91
Reactor Facility Director Date

APPROVED

Chairman, Safety and Health Department 16 May 91
Date

Special Concurrance

17 May 91
Director, AFRRI Date

Reviewed by the Reactor Staff

| | |
|---|---|
| Wright 5/29/91 | |
| Wright | Date |
| R George 5-29-91 | |
| George | Date |
| 29 May 91 | |
| Forsbacka | Date |
| 29 May 91 | |
| Spence | Date |
| M Laughery 29 May 91 | |
| Laughery | Date |
| John T. Nguyen 29 May 91 | |
| Nguyen | Date |
| 29 May 91 | |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991
Date

# SPECIAL NUCLEAR MATERIAL ACCOUNTABILITY

## 1. PURPOSE

This procedure prescribes the responsibility for all work involving the receipt, use, and disposal of Special Nuclear Material (SNM) at AFRRI.

## 2. REFERENCES

a. 10 CFR 70 Domestic Licensing of Special Nuclear Material
b. 10 CFR 73 Physical Protection of Plants and Materials
c. 10 CFR 74 Material Control and Accounting of Special Nuclear Material
d. NUREG/BR-0006/7 Instructions for Material Balance Reports and Transaction Reports
e. AFRRI Health Physics Procedures

## 3. RESPONSIBILITY

a. Overall responsibility for all SNM at AFRRI rests with the Reactor Facility Director (RFD). The RFD designates individuals to be in charge of supervising the use of and accountability for specific types of SNM.
b. Members of the reactor staff will use SNM procured under license R-84 in accordance with the reactor Technical Specifications and Reactor Operating Procedures. Users of SNM listed under the byproduct material license will be governed by Health Physics Procedures (HPP 5-X series). The RFD will designate a member of the reactor staff (Inventory Officer) to conduct inventories and maintain accountability records for all SNM as required by Reference d, this procedure, and HPPs.
c. The inventory officer will receive training in fuel records, accountability procedures, and inventory procedures prior to assignment as inventory officer. This will be documented on the appointment memo. (Appendix A)

## 4. RECEIPT AND TRANSFER OF SNM

Receipt of SNM at AFRRI and transfer of SNM from AFRRI are regulated by 10 CFR and 49 CFR and performed in accordance with HPPs 0-3 and 0-5. When any SNM is received or transferred, the responsible inventory officer will ensure that DOE/NRC Form 741 is submitted as required by Reference d and 10 CFR 74.15.

The RFD must authorize any receipt or transfer of SNM. Approval is also required from the RXSC for any SNM on the byproduct material license.

## 5. ACCOUNTABILITY AND INVENTORY PROCEDURES

a. Safety and Health Department (SHD) personnel will conduct leak tests as required by HPP 5-2. A physical inventory of all sealed SNM sources on the byproduct license will be conducted as of March 31 and September 30 of each year. This semiannual inventory will be conducted to ensure annual compliance with 10 CFR 70. The specific inventory method will follow ALARA considerations. Reports required by References c and d (DOE/NRC Form 742/742C) will be prepared and submitted within 30 days (10 CFR 74.13).

b. A current reactor fuel inventory sheet (Appendix B) will be maintained. It will give the core loading (core position) for all in-core fuel elements and FFCRs. During each inventory, the inventory officer will sign the inventory sheet verifying loading. This document will be updated if changes occur and at least 10% of the in-core elements will be verified by serial number as to location annually.

c. Inventory procedures for reactor fuel elements and fuel-follower control rods will be as follows (conducted as of March 31 and September 30 each year by the inventory officer):

(1) Perform K-excess to verify the core is loaded as per the fuel inventory sheet.

(2) Sign the fuel inventory sheet to certify correct loading and attach to summary accountability sheet. (Appendix C)

(3) Inventory by serial numbers fuel elements in storage except for entombed damaged elements (previously recorded)

(4) Account for any fuel elements or FFCRs received or shipped out during the reporting period

(5) Ensure number of elements and FFCRs present matches the number of fuel element record sheets and that Fuel Element Record (AFRRI Form 255) sheets are current

(6) Transfer data from fuel inventory sheet to summary accountability sheet and verify correct total number of fuel elements and FFCRs

(7) Calculate fuel burnup in grams based on kilowatt-hours generated during the reporting period and subtract from the previous material balance

(8) Complete and submit DOE/NRC Forms 741, 742, and 742C using procedures in Reference d.

(9) File the fuel inventory sheet, the summary accountability sheet, and copies of all DOE/NRC forms in reactor files.

d. Fission chambers and other SNM on the R-84 license will be inventoried by piece count in conjunction with the fuel inventory, recorded on the summary accountability sheet, and reports submitted as in (8) above. All smear testing/handling will be in accordance with Reference e.

## 6. RECORDS

All records and reports pertaining to any SNM held at AFRRI will be maintained in reactor facility files by the inventory officer. Duplicate receipt and transfer files, radiological survey documents and leak test results will be maintained in SHD files.

## 7. REPORTS OF SNM INCIDENTS

Reports of loss, theft, attempted theft, or diversion of SNM will be submitted as required by 10 CFR 74.11. Other incidents involving SNM may be reportable under reactor Technical Specifications or 10 CFR 20.402-20.405

## 8. SIGNATURE AUTHORITY

All reports and other outgoing correspondence dealing with SNM will be signed only by the Reactor Facility Director or his designee (see paragraph 3a).

## INVENTORY OFFICER

____________________ is appointed Inventory Officer to perform duties as per Administrative Procedure A4. He has received proper training in accountability procedures, inventory procedures, and fuel records.

________________________________________

**Reactor Facility Director      Date**

APPENDIX B - FUEL INVENTORY WITHDRAWN FROM PUBLIC DISCLOSURE

## Summary Accountability Sheet

| | | Number | Weight |
|---|---|---|---|
| Fuel Elements In Core | | ______ | |
| Fuel Elements In Tank Storage | | ______ | |
| Total Fuel Elements | | ______ | ______ |
| Fuel Follower Control Rod | | ______ | ______ |
| | (Location) | | |
| Chambers | ______ | ______ | ______ |
| | ______ | ______ | ______ |
| | ______ | ______ | ______ |
| Foils | ______ | ______ | ______ |
| | ______ | ______ | ______ |
| Sources | ______ | ______ | ______ |

This procedure has been approved by the Reactor Facility Director

M L Moore APPROVED 29 May 91
Reactor Facility Director Date

Douglas M. Kelly 28 May 91
Chairman, Safety and Health Department Date

Reviewed by the Reactor Staff

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|---|---|
| T Wright | 5/29/91 |
| Wright | Date |
| R George | 5-29-91 |
| George | Date |
| [illegible] | 29 May 91 |
| Forsbacka | Date |
| [illegible] | 29 May 91 |
| Spence | Date |
| M Laughery | 29 May 9 |
| Laughery | Date |
| John Nguyen | 29 May 91 |
| Nguyen | Date |
| [illegible] | 29 May 91 |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991
Date

# PROCEDURE CHANGES

## GENERAL

This establishes procedures for permanently or temporarily changing reactor operating procedures.

## SPECIFIC

1. Permanent changes are made by revising the entire procedure. The revised procedures will be approved by the Reactor Facility Director (RFD) and reviewed by the Reactor and Radiation Facility Safety Committee (RRFSC).

2. Temporary changes may be made in pen and ink on the current procedure when initialed by the RFD or Reactor Operations Supervisor (ROS). These changes must be documented, approved by the RFD, and reviewed by the RRFSC at the next scheduled meeting.

3. Temporary procedures may be established by the RFD for a specific situation.

4. All procedures (temporary or permanent) will have an signature block for all operators and reactor staff members. Operators will review new or modified procedures and sign the signature block prior to operating the reactor console When the block is completed, the procedure will be placed in the Reactor Procedures Binder and kept available for operator review.

5. All changes will be accomplished under the following guidelines:

   a. The change will result in no decrease in the safety of the actions being addressed.

   b. The change will result in no decrease in the efficiency of procedure performance.

   c. The change will not affect the ability of the procedure to perform its intended function.

7. All changes will be staffed to the following:

  a. Chairman, Safety and Health Department (SHD)*

  b. Reactor and Radiation Facility Safety Committee (RRFSC)

  c. AFRRI TRIGA Reactor Facility staff

Procedures that may affect other areas such as building changes, security, etc., will be staffed to the appropriate office(s) prior to routing to Chairman, SHD

* NOTE: Procedural changes that do not deal specifically with health physics procedures or radiation safety issues need not be staffed through Chairman, SHD.

This procedure has been approved by the Reactor Facility Director

APPROVED

[signature] 28 May 91
Reactor Facility Director Date

[signature] 28 May 91
Chairman, Safety and Health Department Date

Reviewed by the Reactor Staff

| | |
|---|---|
| TWright | 5-29-91 |
| Wright | Date |
| R George | 5-29-91 |
| George | Date |
| [signature] | 29 May 91 |
| Forsbacka | Date |
| [signature] | 29 May 91 |
| Spence | Date |
| M. Laughery | 29 May 91 |
| Laughery | Date |
| John Nguyen | 29 May 91 |
| Nguyen | Date |
| [signature] | 29 May 91 |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991
Date

# CONDUCT OF EXPERIMENTS

## GENERAL

1. All experiments will be observed during irradiation with the exception of CET experiments or those in which no movement is possible. The closed-circuit televisions (CCTVs) in the exposure rooms and over the reactor pool can be used to meet this requirement.

2. All experiments will be set up so as to preclude movement unless the experiment apparatus is designed for movement (such as rotators, etc.).

3. All animal experimental arrays (shielding) in the exposure rooms that are set-up on wooden tables or on styrofoam will have an absorbent pad placed over the wood or styrofoam surface to prevent sanitation problems from the animal waste.

4. The Reactor Staff will conduct a thorough inspection of all experiments to determine that no unauthorized materials are irradiated.

5. ALARA will be practiced during all experiments.

## SPECIFIC

1. A Reactor Use Request (RUR) is required for any experiments included under authorizations outlined in the Technical Specifications, section 6.4.2.a. and section 6.4.2.b.. RURs are not required for reactor parameter measurements as outlined in the Technical Specifications, section 6.4.2.c. Any experiment performed by the reactor staff (except T.S. 6.4.2.a) for the purpose of determining information to be used to enhance, define, ascertain, or develop methods to expand the performance of the reactor will not require an RUR. Facility tours will not require an RUR but will require verbal approval of either the Reactor Facility Director (RFD) or the Reactor Operations Supervisor (ROS).

2. Experiment Review (Processing of RURs):

   a. Check the RUR for completeness (Section I should be filled out).

   b. Forward the RUR to the Radiation Biophysics Department, Operational Dosimetry Division (BRPD) if dosimetry support is required.

   c. Forward it to the Safety & Health Department (SHD) for radiological safety coordination.

d. Check experiment protocol against reactor authorizations.

e. Fill-in Section II of RUR with special instructions, as appropriate. Assign an RUR sequence number. Write in estimated or measured experiment worth and the core position of the experiment facility to be utilized in the appropriate block (lower left-hand corner of form).

f. Have the RFD, acting RFD, ROS or acting ROS review and sign the form.

g. Ensure the RUR form is placed in the reactor control room prior to the irradiation.

3. Conduct of Experiments. Perform setup and irradiation of experiments in accordance with the following procedures:

a. Exposure Room Entry - TAB A.

b. Core Experiment Tube (CET) - TAB B.

c. Extractor System - TAB C.

d. Pneumatic Transfer System (PTS) - TAB D.

e. In-pool/In-core Experiments - TAB E.

4. Complete the RUR by filling out Section IV with the appropriate information.

5. Attach form to clipboard in the control room.

This procedure has been approved b the Reactor Facility Director

MLMoore 29 May 91
Reactor Facility Director APPROVED Date

28 May 91
Chairman, Safety and Health Department Date

Reviewed by the Reactor Staff

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|---|---|
| Wright 5-29-91 | |
| Wright | Date |
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| M Laughery 29 May 91 | |
| Laughery | Date |
| 29 May 91 | |
| Nguyen | Date |
| 29 May 91 | |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991
Date

# REACTOR EXPOSURE ROOM ENTRY

1. REFERENCES

a. 10 CFR 20, "Standards for Protection Against Radiation"

b. USNRC licenses: R-84, 19-08330-02

c. AFRRI Instruction 6055.8B

2. GENERAL

a. PURPOSE: This procedure specifies all safety and security procedures for activities involving entry into the AFRRI TRIGA Reactor exposure rooms, currently designated exposure rooms 1 and 2 (rooms 1123 and 1122).

b. AUTHORIZED ENTRY: Both AFRRI picture badge and U-badge personnel, may enter a reactor exposure room under the supervision of the Reactor Facility Director (RFD) or his representative. Visiting personnel (V badge) require special authorization by both the Chairman, Safety and Health Department (SHD) and RFD to enter either exposure room. In general, permission to enter the exposure rooms will be granted personnel whose duties require such entry; however, permission may be denied to personnel for serious or repeated safety or security violations, or for safety reasons emanating from conditions in the exposure rooms. All personnel who are granted unescorted access to the prep area or warm storage will receive a special prep area safety briefing prior to being granted access. Only personnel who have been granted unescorted access will be given the combination to the prep area . The RFD is responsible for maintaining a roster in the prep area for personnel who have been granted unescorted access. Other personnel requiring unescorted access to the prep area or warm storage for a specific purpose or time period may be granted special access in writing by the RFD with concurrence of SHD. However, these personnel who are granted special access from the RFD will not be given the combination to the prep area.

c. ER ENTRY INSTRUCTIONS - All personnel will:

(1) Know the Reactor staff representative is in charge of all operations in the prep area. Obtain permission to enter either exposure room from the Reactor staff representative.

(2) Wear AFRRI TLD whole body badge and pocket dosimeter.

(3) Wear wrist or finger dosimeter if work is to be performed on an experimental array or within one meter of the core projection.

(4) Wear booties, eye protection, gloves and coat.

(5) Check and log pocket dosimeter reading on log in prep area prior to entry.

(6) Familiarize themselves with approximate radiation levels in the room, based on radiological surveys performed and data obtained by SHD.

(7) Ensure that all materials removed from the exposure room are properly labeled and entered on the exposure room entry log (AFRRI FORM 130) and the activated materials control log.

(8) Glove and coat requirements may be waived by the Reactor Representative on an individual basis for personnel who will not be touching anything in the exposure room. There must be a specific reason for waiving such requirements.

d. DEPARTURE FROM REACTOR EXPOSURE ROOM ENTRY PROCEDURES: Any departure from the following procedures will require a special work permit (SWP). Exceeding any radiation dose limits will require a written justification from the supervisor of the research project which must be approved by the Head, SHD.

3. SHD EXPOSURE ROOM SURVEY

a. EXPOSURE ROOM CAM: Prior to opening either exposure room, the respective CAM must read 2000 cpm or less, above background. If the CAM reads 2000 cpm or greater above background, change the filter of the CAM. If 10 minutes or more have elapsed since the end of the reactor run, the door may be opened to the first step to facilitate radioeffluent clearance in the room. Then check the CAM after 1 minute and if the reading is below 2000 cpm above background, proceed with the exposure room opening. If it is above, change the filter and wait another minute. If the CAM alarms during or immediately after a run, change the filter and reset the CAM.

b. DOSE RATE AT FACE OF DOOR: If the dose rate at the face of the plug door in the direct line of sight of the reactor tank bulge reads greater than 100 mr/hr, the door will be closed sufficiently to preclude access. The plug door will be reopened upon agreement of the SHD and RFD representatives for reevaluation of radiation levels.

c. DOSE LEVELS IN ROOM: Exposure rates will be measured at specific sites in the rooms. These measurements will be given to both the reactor representative and the personnel entering the room. Additionally the readings will be entered in the room entrance log (AFRRI FORM 130) and kept in the prep area. The levels will be measured at:

(1) The reactor door face in the direct line of sight of the reactor tank bulge

(2) At the contamination line in the entrance of the room

(3) The middle of the room

(4) One meter from the tank wall or shield

(5) Contact with the tank wall or shield

(6) The area(s) where individual(s) will be working for an extended period of time and any other place deemed necessary by the SHD or reactor representatives.

d. ROUTINE ENTRY: Entry is routinely permitted only when the maximum reading in any occupiable area is 1 R/h or less. Entry may be permitted if levels are 1-5 R/h, but no work will be permitted in fields over 1 R/h. If personel are working in a specific area for an extended period of time , the dose rate in that area will be measured.

(1) Readings over 100 mR/hr (closed window) will be reported to the Reactor representative by the SHD monitor. These areas of the exposure room will be identified to the Reactor representative and entry personnel. When appropriate, after consultation with the SHD and Reactor representatives, stay times will be assigned for entry personnel. All personnel entering will be assigned a stay time if they will be working in the high radiation area. AFRRI limits of 100 mR/week and 50 mR/day are to be used as the basis of stay time determinations.

(2) All exposure room entries will be checked by the SHD monitor for compliance with radiation safety aspects of applicable Reactor Use Requests (RURs). If not, non-compliance will be reported to the RFD and to SHD.

e. FILLING OUT THE SURVEY OF EXPOSURE ROOM OPENING LOG: The exposure room opening log sheet must be filled out completely for each opening of an exposure room. Care must be taken to fill out each blank on the entry log sheet. If a section is not applicable to the particular opening, N/A should be filled in the blank.

4. NON MONITORED OPENING:

a. Personel may enter the exposure rooms without a SHD monitor present if ALL the following conditions hold:

(1) The reactor has not been to power in that ER since the last survey.

(2) Survey meter readings at the door indicate safe entry conditions (should be less than 1 mR/hr).

(3) The ER CAM should be observed, and its reading (net) should be less than 200 cpm above background.

b. An entry will be made in the exposure room log by a reactor staff member, with a note that the survey has been waived.

c. SHD must be notified if any radioactive materials or equipment are to be removed from the prep area.

## 5. PERSONNEL PROTECTION PROCEDURES

a. Dosimetry and protective clothing requirements are given in paragraph 2.c, *ER Entry Instructions.*

b. Entry is permitted only after the SHD monitor has completed the survey and reported results to those about to enter (excluding non-monitored openings - Reference Paragraph 4, above).

c. All personnel shall record initial dosimeter reading in the prep area dosimeter log prior to entering the exposure room for the first time each day. Personnel shall read dosimeters when leaving the exposure room and record a final dosimeter reading in the prep area log at completion of daily operations. Net doses over 10 mrem must be reported to the SHD Monitor.

d. Protective clothing will be removed in such a way as not to contaminate "clean" areas by items from "dirty" areas.

e. All personnel will "frisk" themselves before leaving the prep area.

## 6. SPECIFIC ACTIONS TO OPEN EXPOSURE ROOM DOORS

a. Turn up exposure room lights (this can be waived for experiment needs).

b. Check plug door tracks for obstructions; ensure all obstacles are clear of the door (including ropes).

c. Ensure that only authorized personnel (see 2.b.) are present in the reactor prep area during exposure room openings.

d. When facility safety interlocks and opening procedures have been satisfied, insert key into exposure room door key panel and open door. DO NOT LEAVE KEY IN LOCK UNATTENDED.

e. Open door in accordance with entry procedures. Ensure all required data is logged in entry log.

f. Ensure that individuals who will be moving lead, bismuth, or other heavy materials are wearing steel-toed shoes.

g. Limit exposure times of all personnel entering the exposure rooms based on the results of the radiation survey.

## 7. ACTIVATED MATERIALS

a. PLACING MATERIAL IN EXPOSURE ROOM: Before placing any equipment or material in an exposure room for irradiation the following will be observed:

(1) Equipment tagged as AFRRI property: a DF must be sent to both the RFD and the AFRRI property officer. The DF must state that the equipment is knowingly being irradiated and therefore request that it be removed from the property books. It must also state that should the material remain byproduct material after a reasonable amount of time it will be disposed of as radioactive waste. The DF must contain all nomenclature as well as an adequate description of the equipment in order for it to be identified on the property book.

(2) Non-tagged AFRRI equipment or material (to be returned): a DF or statement on the reactor RUR must be sent to the RFD giving the kinds and amounts of byproduct material expected to be produced (that is the material that the experimenter wishes to be returned) and a copy or number of their radionuclide authorization number. The DF or RUR statement must be specific and contain an accurate description of the material being exposed (converted to byproduct). Other information will be required from personnel before any material is allowed to be removed from the prep or warm storage areas (see next section of this procedure 7.b. and 7.c.)

(3) Non-tagged equipment or material (not to be returned): A DF or statement on the RUR that the experimenter understands that byproduct material produced as a result of their irradiations will be disposed of as radioactive waste, and additionally any material not specifically requested to be held, will be disposed of as radioactive waste in the next shipment.

(4) Non-AFRRI owned equipment/material: A signed memorandum from the responsible property owner that they understand that byproduct materials generated in excess of their license will be disposed of as radioactive waste unless prior arrangements have been made with the reactor/SHD staffs for storage. Any mate-

rial not removed within a reasonable amount of time will automatically be disposed of as radioactive waste.

b. SURVEY OF MATERIALS COMING OUT OF EXPOSURE ROOM

(1) All material leaving the exposure rooms must be surveyed for activation or contamination. Survey meter readings will be used to determine dose levels. Smear surveys may be used, if the SHD representative deems them necessary. All materials will be labeled appropriately in accordance with HPP 0-2.

(2) All special equipment that has been activated such as chambers, rotators, motors, meters, etc., will be stored under the control of the reactor license or the AFRRI byproduct license in warm storage or the prep area. Removal of items from the prep area will only be allowed in accordance with HPP 3-1

c. DISPOSITION OF ACTIVATED MATERIALS

All activated or contaminated materials will be under the control of the reactor license while such materials remain in the reactor controlled area. Removal of any radioactive materials from the reactor controlled area will be done in accordance with HPP 3-1.

## 8. COMPLETION OF ENTRY

a. The Reactor Staff Representative will check to see that all personnel have left the exposure room before the plug door is closed. In the event that the warning horn in either exposure room is disconnected, for testing or experiment requirements, the exposure room plug door shall not be closed until at least two (2) licensed reactor operators visually inspect the room to ensure that no personnel remain in the room. To ensure compliance with the reactor Technical Specifications, the names of these licensed operators present at the exposure room closing shall be entered into the reactor operations logbook and on AFRRI FORM 130. At the completion of the test or experiment, the warning horn shall be reconnected and tested. All actions regarding the warning horn shall in entered in GREEN ink in the reactor operations logbook.

b. The SHD monitor will not leave the area while the plug door is open without notifying the Reactor Staff Representative.

c. Lock the exposure room door control panel; reset lights, if appropriate.

d. Resecure the prep area on departure.

This procedure has been approved by the Reactor Facility Director

APPROVED

Reactor Facility Director Date 29 May 91

Chairman, Safety and Health Department Date 29 May 91

Reviewed by the Reactor Staff

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| Wright 5-29-91 | |
| Wright | Date |
| R George 5-29-91 | |
| George | Date |
| 29 May 91 | |
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| Spence | Date |
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| Laughery | Date |
| John Nguyen 29 May 91 | |
| Nguyen | Date |
| 29 May 91 | |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991

Date

# CORE EXPERIMENT TUBE (CET)

## GENERAL

ALARA principles will be practiced during CET operations.

## SPECIFIC

1. CET Insertion into the core:

   a. Ensure a reactor operator is monitoring the reactor console.

   b. Ensure a reactor staff member is present in the reactor room.

   c. Establish communications between the reactor room and the control room.

   d. Test fuel-handling tool for operability.

   e. Lower the fuel-handling tool into the core and attach to the desired element. Notify operator on the console that you are prepared to lift fuel element. When acknowledged, lift fuel element from the core.

   f. Transfer element to a storage rack location and secure fuel-handling tool cable.

   g. Loosen CET bracket bolts and remove CET bracket.

   h. While the CET is held down, cut cable ties from around the CET.

   i. Lift CET from the storage rack location and transfer to the reactor carriage.

   j. Notify the console operator that you are prepared to lower the CET into the core; when acknowledged, lower the CET into the core ensuring that it is properly seated in the lower grid plate.

   k. With a downward pressure on the CET to keep it seated, secure the CET bracket with the two bolts.

   l. Ensure appropriate entries are made in the operations logbook and the fuel book, and that the reactor core pegboard is updated.

2. Irradiation:

   a. Clean the rabbit(s) using alcohol and water.

b. Once clean, do NOT handle the rabbit except with gloves, Kimwipes, or handling tools.

c. Ensure that the rabbit cap is secured tightly.

d. The rabbit may be inserted into the CET either before the run begins or while the reactor is at power .

e. If the rabbit is to be inserted while the reactor is at power, then after notifying the reactor operator on console, drop or lower the rabbit into the core WITH THE CAP UP. Ensure that you spend a minimum amount of time in the vicinity of the carriage. Do NOT lower the rabbit with the extractor tool while at power.

f. Complete irradiation and shut down reactor.

g. Ensure appropriate entries are made in the operations logbook and the CET logbook.

3. Rabbit Retrievals:

a. Ensure that a reactor staff member and a Safety & Health Department (SHD) monitor are present in the reactor room. Any staff member who will be handling the sample following the irradiation may be required to wear a pocket chamber and appropriate extremity dosimetry depending on the radiation levels of the irradiated sample. If the CET is in the core, a reactor operator must monitor the console during the retrieval.

b. Test the rabbit extractor ("fishing pole") for operability.

c. Insert the extractor head mechanism into the CET and reel out cable until you reach the low end indicator painted on the cable.

d. Drop the extractor head firmly on the rabbit.

e. Ensure the SHD monitor has a teletector positioned near the CET top to monitor the rabbit.

f. If the CET is in the core, notify the reactor operator that the rabbit is being pulled and continue when acknowledged.

g. Reel in the cable at a rate commensurate with radiation levels; lower the rabbit back into the CET if the rabbit is excessively hot.

h. Stop when the rabbit is visible at top of CET; have SHD take an accurate radiation reading.

i. If radiation levels are acceptable, swing rabbit away from carriage and have another individual grab it with a handling tool. If the radiation levels are not acceptable, lower the rabbit back into the CET. The rabbit will again be withdrawn for reevaluation of radiation levels when the SHD and RFD representatives concur on an acceptable radiation level in accordance with ALARA and mission requirements.

j. Release extractor head and detach rabbit from head.

k. Unless working with the rabbit or radiation levels are low, store rabbit or irradiated material in a lead pig or storage cask.

l. Make appropriate entries in the operations and CET logbooks.

4. CET Removal from Core:

a. Complete steps 1a-c above.

b. Loosen the CET bracket bolts while holding the CET down; remove the CET bracket.

c. Notify the console operator that you are prepared to remove the CET from the reactor core.

d. When acknowledged, transfer the CET to the storage rack, ensuring that it is kept as low in the water as possible.

e. Secure the CET with cable ties.

f. Secure the CET bracket with the two bolts.

g. Remove the fuel element from the storage rack and transfer to core. Notify the console operator and receive acknowledgment prior to insertion of the element .

h. Ensure the element is properly seated in the lower grid plate by listening for the "double clicks".

i. Make appropriate entries in the operations and fuel logbooks and update the reactor core pegboard.

This procedure has been approved by the Reactor Facility Director

29 May 91
Reactor Facility Director APPROVED Date

28 May 91
Chairman, Safety and Health Department Date

Reviewed by the Reactor Staff

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| 5-29-91 | |
| Wright | Date |
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| George | Date |
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| 29 May 91 | |
| Spence | Date |
| M. Laughery 29 May 91 | |
| Laughery | Date |
| John Nguyen 29 May 91 | |
| Nguyen | Date |
| 29 May 91 | |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991
Date

# EXTRACTOR SYSTEM

## GENERAL

The extractor system will be tested for operability prior to the initial experiment for the day.

## SPECIFIC

1. Assembly of the extractor system:

   a. Inside the exposure room:

      (1) Move the inside receiver section into position in front of the core; screw tube supports to the floor .

      (2) While holding the appropriate connecting tube in position, tie the strings in the tube to the two ends coming out of the exposure room wall and to the two ends in the receiver section.

      (3) Align the ends of the tubes and slide the clamp over each joint.

      (4) Place the alignment tools into the appropriate holes to check the tube alignment; tighten down the clamps.

      (5) Remove the alignment tools.

   b. Outside the exposure room:

      (1) Remove tube plug.

      (2) Move the receiver section close to the tube projecting from the wall.

      (3) While someone else is pushing the table toward the wall, insert two screws into the holes on the securing bracket (beneath the table).

      (4) Tie the string from the end of the small tube to the end of the wire cable.

      (5) Pull the string in the large tube slowly while having someone inside the room guide the string.

      (6) When the cable is all the way through both tubes, thread the cable through the receiver tube while moving the receiver table into final position against the wall (if necessary, add an additional length of cable to the take-up reel).

(7) Position and tighten clamp over the joint; position carrier in tube and connect cable to each end; remove the tape on the take-up reel.

(8) Pull back on the drive motor assembly until there is no slack in the cables; tighten the adjustment bolts on the drive assembly.

(9) Connect the electrical cables to the motor, control unit, and limit switches.

2. Disassembly:

a. Reverse the order of the above with the following changes:

(1) Before loosening the motor assembly, place tape on the cable drum to keep the cable from moving (ensure the carrier is in the receiver section).

(2) Before pulling the cable through the tubes, attach a new string to it.

(3) Leave enough slack for disassembly inside the exposure room.

(4) Cut the string at the joints in the room and tape the ends to the tubes or tie the ends together .

b. Ensure the tube plug is in place, and the control unit is secured.

3. Operations:

a. On the motor control, initially set controls as follows:

(1) Power switch: "OFF".

(2) Torque control: "OFF".

(3) In/out switch: "BRAKE".

(4) Speed control: "0%".

b. Plug motor control into AC outlet; switch the power switch to "ON".

c. Switch in/out switch to appropriate position.

d. Slowly increase speed to an appropriate level; as the carriage approaches its full in/out position, decrease the speed slowly to "0%".

e. Turn the in/out switch to "BRAKE".

f. During retrieval operations, the Safety and Health Department (SHD) monitor will be present.

This procedure has been approved by the Reactor Facility Director

29 May 91
Reactor Facility Director APPROVED Date

28 May 91
Chairman, Safety and Health Department Date

Reviewed by the Reactor Staff

| | |
|---|---|
| T. Wright | 5-29-91 |
| Wright | Date |
| R George | 5-29-91 |
| George | Date |
| | 29 May 91 |
| Forsbacka | Date |
| Spence | 29 May 91 |
| Spence | Date |
| M. Laughery Jr | 29 May 91 |
| Laughery | Date |
| John Nguyen | 29 May 91 |
| Nguyen | Date |
| Owens | 29 May 91 |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991
Date

# PNEUMATIC TRANSFER SYSTEM (PTS)

## GENERAL

1. This PTS procedure is inactive. If the PTS Facility is reactivated, then this procedure must be reviewed and approved by the RRFSC and the Reactor Facility Director.

2. ALARA principles will be practiced during PTS operations.

3. All PTS operations will be directly supervised by a reactor operator present in the Hot Lab.

## SPECIFIC

1. PTS Setup:

   a. Position core at 700 (inside region III).

   b. Ensure communications are established between the hot lab and the control room.

   c. Inspect rabbits to be used in the PTS for cracks or other damage.

   d. Aluminum rabbits must be diverted to the Hot Cell and therefore may only be used on the "A" system.

   e. If the anticipated radiation level of any returned rabbit is greater than 1.0 R/hr at 1 meter, take the following precautions:

      (1) Use the remote control unit, unless experiment requirements dictate otherwise.

      (2) Place a radiation survey meter next to the receiver/sender station so that it can be monitored from the remote control unit.

      (3) The rabbit will be irradiated in the "A" system and then diverted to the Hot Cell or returned to the irradiation location.

2. Manual Operations:

   a. Ensure all switches on both the local and remote control units are in the "OFF" position; place the local/remote switch in the desired position.

   b. Place blower switch in the "ON" position.

c. Insert key into local control unit; turn key to "ON" position.

d. Ensure tubes are empty.

e. Set mode switch (man/off/auto) to "MAN" position. Blower will start.

f. Set in/out switch to the "OUT" position and the tube on/off switches to "ON"; allow the system to run for a short time.

g. Set tube on/off switches to "OFF" and turn in/out switch to "IN".

h. Load samples into tubes.

i. Check communications with reactor operator at the reactor console.

j. When the reactor is at the designated power level, set the tube on/off - switches to "ON" one at a time, to send rabbits into the irradiation location.

k. Begin stopwatch or timer.

l. Turn tube on/off switches to "OFF" and turn in/out switch to "OUT".

m. Ensure a Safety & Health Department (SHD) monitor is present during retrievals.

n. Set on/off switch to "ON" one at a time; rabbits will return to sender/receiver station.

o. Set all switches to "OFF", and remove key from control unit.

3. Automatic Mode:

a. Complete steps 2a-d above.

b. Set mode switch to "AUTO" position. Blower will start.

c. Complete steps 2f-i above.

d. Set timer (0 to 5 minutes) by turning the red and black arrows to the desired irradiation time.

e. When the reactor is at the desired power level, briefly push the timer push button and release. The rabbits will leave the receiver/sender station and will automatically return at the end of the preset irradiation period. The timer will automatically reset.

f. Turn all switches to "OFF" and remove key from control unit.

4. Diverting Samples:

a. Diversion of samples to the Hot Cell may only be made using the "A" system.

b. After the rabbit has returned to the receiver/sender station, set the divert/send switch to "DIVERT" and hold it until the loading port handle trips to the rear position.

c. Send the divert/send switch to "SEND" and hold for a few seconds. The rabbit will leave the receiver/sender station and travel to the Hot Cell.

This procedure has been approved by the Reactor Facility Director

29 May 9.

Reactor Facility Director Date

APPROVED

29 May 91

Chairman, Safety and Health Department Date

Reviewed by the Reactor Staff

| | |
|---|---|
| 5-29-91 | |
| Wright | Date |
| 5-29-91 | |
| George | Date |
| 29 May 91 | |
| Forsbacka | Date |
| 29 May 91 | |
| Spence | Date |
| 29 May 91 | |
| Laughery | Date |
| 29 May 91 | |
| Nguyen | Date |
| 29 May 91 | |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991

Date

# IN-POOL/IN-CORE EXPERIMENTS

## GENERAL

ALARA principles will be followed during these experiments. These procedures apply to all in-pool or in-core experiments except CET operations (See Procedure 1 - Tab B).

## SPECIFIC

1. All operations will be supervised by a licensed operator

2. Actions will be taken to prevent damage to the reactor core or aluminum tank.

3. Ensure that a member of the reactor staff and a SHD representative are present in the reactor room during the removal of samples from in-pool or in-core locations.

4. The removal of experimental materials from the pool or core will be monitored with a radiation survey meter; additionally, a reactor operator will monitor the reactor console during insertion and removal of in-core/in-pool experiments.

This procedure has been approved by the Reactor Facility Director

APPROVED

29 May 91

Reactor Facility Director Date

Reviewed by the Reactor Staff

| | |
|---|---|
| T Wright 5-29-91 | |
| Wright | Date |
| 5-29-91 | |
| George | Date |
| 29 May 91 | |
| Forsbacka | Date |
| 29 May 91 | |
| Spence | Date |
| 29 May 91 | |
| Laughery | Date |
| 29 May 91 | |
| Nguyen | Date |
| 29 May 91 | |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991

Date

# REACTOR STAFF TRAINING

1. The reactor staff training is delineated in the current "AFRRI Reactor Operator Requalification Program".

2. The Reactor Facility Director (RFD) determines who is allowed into the training program. As part of the training/requalification program, the following will be performed:

   a. A training file will be maintained for each trainee/operator.

   b. When a section of training is completed, it will be annotated on the training checklist in each file.

This procedure has been approved by the Reactor Facility Director

APPROVED

M L Mae 24 May 91

Reactor Facility Director Date

Reviewed by the Reactor Staff

| | |
|---|---|
| T Wright | 5-29-91 |
| Wright | Date |
| R George | 5-29-91 |
| George | Date |
| | 29 May 91 |
| Forsbacka | Date |
| | 29 May 91 |
| Spence | Date |
| M Laughery | 29 May 91 |
| Laughery | Date |
| John Nguyen | 29 May 91 |
| Nguyen | Date |
| Chen Owens | 29 May 91 |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991

Date

# MAINTENANCE PROCEDURES

## GENERAL

Maintenance procedures are provided in other references.

## SPECIFIC

1. Preventive maintenance procedures for each item of the reactor systems are provided in the maintenance logbook and console systems manuals.

2. Annual shutdown procedures are given in the Annual Shutdown Checklist which is revised each year by the Reactor Operations Supervisor (ROS) and approved by the Reactor Facility Director (RFD).

3. Malfunctions are annotated in the Malfunction Logbook. Entries are normally made by the operator who discovered the deficiency. When corrective actions have been made and annotated in the malfunction logbook, the RFD or ROS shall review and initial the entry.

4. Procedures for maintenance of specific equipment are provided in the manufacturers' literature.

This procedure has been approved by the Reactor Facility Director

M. L. Moore 18 Nov 91

Reactor Facility Director APPROVED Date

11-20-91

Chairman, Safety and Health Department Date

Reviewed by the Reactor Staff

| | |
|---|---|
| 20 Nov 9 | |
| Holmes | Date |
| 18 Nov 91 | |
| George | Date |
| 18 Nov 91 | |
| Forsbacka | Date |
| Harry Spence 18 Nov 91 | |
| Spence | Date |
| Michael Laughery 18 Nov 91 | |
| Laughery | Date |
| John T. Nguyen 18 Nov 91 | |
| Nguyen | Date |
| 18 Nov 91 | |
| Owens | Date |
| 20 Nov 9 | |
| | Date |
| 11-20-91 | |
| RPO | Date |

Reviewed by RRFSC 17 DEC 1991

Date

# PERSONNEL RADIATION PROTECTION

## GENERAL

All activities performed in areas of potential personnel radiation exposure will be done in accordance with ALARA principles. These areas are the reactor room, upper equipment room (3152), lower equipment room (2158), warm storage, prep area, exposure room 1, exposure room 2, and the hot lab/cell. AFRRI Instruction 6055.8B, Occupational Radiation Protection Program, is the radiation protection program followed by RSDR.

## SPECIFIC

1. Reactor Room:

   a. CET Operations: See Procedure 1-Tab B.

   b. When working inside chained area around pool: The reactor operator on the console shall be responsible for controlling entry into the chained area during operations

2. Warm Storage: See HPP 3-1.

3. Prep Area: See Prep Area Briefing.

4. Exposure Rooms: See HPP 3-1 and Procedure 1-Tab A.

5. Hot Lab/Cell: See HPP 3-5 and Procedure 1-Tab D.

6. Upper and Lower Equipment Rooms:

   a. No written radiation protection procedures are required for entry into these rooms.

   b. Access to these areas is controlled by the AFRRI Reactor Physical Security Plan.

7. Personnel Dosimetry and Monitoring: See HPP 3-1, 3-2, and the Prep Area Briefing.

This procedure has been approved by the Reactor Facility Director

APPROVED

Reactor Facility Director 29 May 91
Date

Reviewed by the Reactor Staff

| | |
|---|---|
| 5-29-91 | |
| Wright | Date |
| 5-29-91 | |
| George | Date |
| 29 May 91 | |
| Forsbacka | Date |
| 29 May 91 | |
| Spence | Date |
| 29 May 91 | |
| Laughery | Date |
| 29 May 91 | |
| Nguyen | Date |
| 29 MAY 91 | |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991
Date

# PHYSICAL SECURITY

## GENERAL

Physical Security requirements are given in the AFRRI Reactor Physical Security Plan.

## SPECIFIC

1. The reactor control room and the reactor room will be secured if no reactor staff member is present for a prolonged period of time during duty hours.

2. Control of keys is delegated to the Reactor Operations Supervisor. Key inventories will be performed annually, not to exceed 15 months.

3. The Physical Security Plan will be reviewed annually.

This procedure has been approved by the Reactor Facility Director

APPROVED

29 May 91

Reactor Facility Director Date

Reviewed by the Reactor Staff

| | |
|---|---|
| 5-29-91 | |
| Wright | Date |
| 5-29-91 | |
| George | Date |
| 29 May 91 | |
| Forsbacka | Date |
| 29 May 91 | |
| Spence | Date |
| 29 May 91 | |
| Laughery | Date |
| 29 May 91 | |
| Nguyen | Date |
| 29 May 91 | |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991

Date

# EMERGENCY PROCEDURES

## GENERAL

The reactor emergency organization, emergency classes, and emergency action levels are set forth in the AFRRI Reactor Facility Emergency Plan and its Implementing Procedures.

## SPECIFIC

Perform the following, as appropriate (need not be done in order).

1. Reactor Emergency:

   a. SCRAM reactor.

   b. Check radiation monitors; use portable survey instruments to assess situation, if necessary.

   c. Notify ERT Commander of situation.

   d. Activate emergency organization.

2. AFRRI Complex Emergency Evacuation:

   a. SCRAM reactor.

   b. Secure any exposure facilities which are in use so that personnel access to that facility is not possible.

   c. Remove logbook, emergency guide, radios, teletector, tool kit, and keys; report to ERT.

   d. Ensure reactor area doors are secured upon departure.

3. Proper classification of emergency situation: All SROs must review the referenced Emergency Plan documents and be able to properly classify the events as they occur. Below is a tabulation of emergency classification to be used as guidance.

| EMERGENCY CLASS | Radiation Alarms (Unanticipated) | Activate AFRRI Complex Emergency Evacuation | Activate Emergency Response Team |
|---|---|---|---|
| **Class 0** | | | |
| | Fire Alarm (non-reactor) | Yes | Yes |
| **Class 1** | | | |
| | R1 > 1 min. | * | Yes |
| | R2> 1 min. | * | Yes |
| | R3 | No | No |
| | R5 > 1 min. | * | Yes |
| | R6 | No | No |
| | E3 > 1 min. | * | Yes |
| | E6 >1 min. | * | Yes |
| | SGM> 1 min. | * | Yes |
| | Reactor Stack Fan Monitor | No | No |
| | Fire Alarm (reactor) | Yes | Yes |
| **Class 2** | | | |
| | CAM> 1 min. concurrent with R1, R2, R5, and/or SGM | * | Yes |

NOTE: * A decision to evacuate the Institute will be made by the ECP Commander based on input from the ERT Commander.

This procedure has been approved by the Reactor Facility Director

APPROVED

M L Moore 29 May 91
Reactor Facility Director Date

Reviewed by the Reactor Staff

T Wright 5-29-91
Wright Date

R George 5-29-91
George Date

29 May 91
Forsbacka Date

29 May 91
Spence Date

M Laughery 29 May 91
Laughery Date

John Nguyen 29 May 91
Nguyen Date

29 MAY 91
Owens Date

Date

Date

Reviewed by RRFSC 24 SEP 1991
Date

# REACTOR CORE LOADING AND UNLOADING

## GENERAL

Loading and unloading of the reactor core shall be performed under the supervision of the Reactor Facility Director or the Reactor Operations Supervisor.

## SPECIFIC

1. Setup

   a. Ensure that at least one nuclear instrumentation channel is operational.

   b. Ensure that the source is in core.

   c. Ensure that an operator monitors the reactor console during all fuel movements.

   d. Check new FFCRs before insertion into the core; this includes cleaning, visual inspection, and length and bow measurements.

   e. Install all control rods.

   f. If irradiated fuel elements are to be removed unshielded from the pool, obtain a Special Work Permit (SWP) from the Safety and Health Department (SHD); do not remove fuel elements with a power history (greater than 1 KW) in the previous 2 weeks from the reactor pool.

2. Core loading

   a. After each step of fuel movement perform the following:

      (1) Record detector readings.

      (2) Withdraw control rods 50%; record readings.

      (3) Withdraw control rods 100%; record readings.

      (4) Calculate 1/M.

      (5) Plot 1/M versus number of elements (and total mass of U-235).

      (6) Predict critical loading.

      (7) Insert ALL control rods; continue to next step.

   b. Load fuel elements in the following order:

      (1) Load the B-ring and C-ring thermocouple elements.

(2) Connect thermocouple outputs to reactor control console display.

(3) Install any other thermocouple elements.

(4) Complete loading of B- and C-ring elements (total of 18 standard elements plus 3 FFCRs).

(5) Load D-ring (total of 33 standard elements plus 3 FFCRs)

(6) Load the following E-ring elements in order:

16, 17, 18, 20, 6, 8, 9, 10 (total of 41 standard elements plus 3 FFCRs).

(7) Complete the E-ring by loading the following elements in order:

15, 21, 11, 5, 14, 22, 4, 12, 13, 1 (total of 51 standard elements plus 3 FFCRs)

(8) Load the following F-ring elements in two elements per step until criticality is achieved using the following loading order:

22, 23, 24, 21, 20, 25, 26, 27, 28, 29, 30, 1, 2, 3, 4, 5, 19, 18, 17, 16, 15, 14, 13, 6, 12, 7, 11, 8, 10, 9.

Once criticality has been achieved, perform control rod worth measurements at core position 500 by rod drop technique. Calculate shutdown margin:

SDM =Total Control Rod Worth - K-excess - TRANS Rod Worth

(9) Load core to $2.00 excess reactivity by loading two elements per step using the loading order in instruction 8.

(10) Verify control rod worth using rod drop techniques, calculate SDM

(11) Load the core to achieve a K-excess that will allow calibration of the TRANS rod based on the last available worth curve of the TRANS rod (approximately $4.00). Calculate the reactivity value of each element as it is added.

(12) Calibrate all control rods.

(13) Calculate the shutdown margin.

(14) Estimate K-excess with a fully loaded core (must not exceed $5.00).

(15) Load core to fully operational load using loading order in instruction 8, and recalibrate all control rods. Calculate the shutdown margin.

(16) Adjust the core loading pattern to meet operational requirements if necessary. Recalibrate all control rods. Calculate the shutdown margin.

8. At the end of each day in which a Daily Operational Startup Checklist or Daily Safety Checklist has been completed, perform a Daily Operational Shutdown Checklist (Tab I).

9. Complete the monthly summary .

10. Respirator equipment will not be used on a routine basis. Respirator equipment is provided for use during emergency conditions only.

This procedure has been approved by the Reactor Facility Director

APPROVED

Mark [signature] 9 Nov 91

Reactor Facility Director Date

Reviewed by the Reactor Staff

[signature] 25 Nov 91
George Date

[signature] 25 Nov 91
Forsbacka Date

[signature] 2 Dec 91
Spence Date

[signature] 3 Dec 91
Laughery Date

John E. Nguyen 3 Dec 91
Nguyen Date

[signature] Owen 25 Nov 91
Owens Date

[signature] 13 Dec 91
Holmes Date

Date

Date

Reviewed by RRFSC 17 DEC 1991
Date

# REACTOR OPERATIONS

## GENERAL

Logbook entries will be made in accordance with the Logbook Entry Checklist (Tab A).

## SPECIFIC

1. The names of the individuals who supervised and performed the daily and weekly checklists will be shown at the top of the checklist. Checkmarks or numbers, as appropriate, will then be entered on each checklist line as that item is performed.

2. Perform reactor Daily Operational Startup Checklist (Tab B), utilizing appropriate nuclear instrumentation set points (Tab C). In the case of no planned operations, a Daily Safety Checklist (Tab B1) may be performed.

3. Record at the beginning of each day in the reactor operations logbook the SRO on-call for that date. ~~At the end of the working day, record the name of the SRO on-call for the upcoming night, weekend, or holiday.~~ EFFECTIVE 13JAN92

4. At the begining of each working day, also record ~~the~~ name of the physicist in charge (PIC) present at the reactor facility. If ~~the PIC~~ either the PIC or SRO on-call changes during the day, a revised entry will be made in the logbook.

5. Perform K-excess measurement (Tab D).

6. Perform operations in accordance with the following:

   a. Steady state operation (Tab E).

   b. Square wave operation (Tab F).

   c. Pulse operation (Tab G).

   d. CET operations (Procedure 1, Tab B).

   e. Pneumatic Transfer System (Procedure 1, Tab D).

   f. In-pool/in-core experiment (Procedure 1, Tab E)

7. Perform Weekly Operational Instrument Checklist once during each calendar week (Tab H).

This procedure has been approved by the Reactor Facility Director

APPROVED

Mark Moore 16 July 91
Reactor Facility Director Date

Reviewed by the Reactor Staff

| | |
|---|---|
| 16 Jul 91 | |
| George | Date |
| 16 Jul 91 | |
| Forsbacka | Date |
| 16 July 91 | |
| Spence | Date |
| Michael E Laughery 16 July 91 | |
| Laughery | Date |
| John Nguyen 16 July 91 | |
| Nguyen | Date |
| 16 July 91 | |
| Owens | Date |
| | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991
Date

# LOGBOOK ENTRY CHECKLIST

1. The reactor operations logbook is a before-the-fact record, that is, entries will be logged before the operator actually performs the planned function. Any late entries will be so noted.

2. The operations logbook will have a hardbound cover and will be sequentially numbered by volume. The pages will be dated at the top of each page and each page will be sequentially numbered.

3. The Reactor Facility Director (RFD) will review each logbook upon its completion; he will make an appropriate entry in the back of the logbook and sign the entry. The operator who makes the final entry at the end of a logbook is responsible for ensuring that the ROS is notified that the logbook is ready for RFD review.

4. All items in GREEN (see below) that are not closed out during the working day will be carried in GREEN at the end of the day and again at the beginning of the next operational day.

5. The entries will be made in ink and in accordance with the following designated color code:

a. BLACK and BLUE-BLACK:

(1) Console locked and unlocked. The individual at the console will enter his/her name and the supervisory licensed operator's name, if necessary.

(2) Checklist number and completion time.

(3) Power level at criticality and subsequent power level changes.

(4) Reactor SCRAM.

(5) Mode of operations. Use appropriate stamp or entry to designate the operation:

(a) Steady State

(b) Square Wave

(c) Pulse

(6) Operation of reactor associated facilities such as lead shield doors, pneumatic tube systems, etc., unless such operations cause a change of reactivity (see 5.b.(2) below).

(7) Change of personnel at the console. Name of personnel will be entered along with the licensed operator present in the control room, if the person at the console is not a licensed operator.

(8) The operator in charge will be designated in the logbook whenever multiple operators are signed on the console.

(9) Completion of the daily startup and shutdown checklists and weekly checklist.

(10) Signature of reactor operator to close out the log for the day.

(11) Designation of the SRO on-call and physicist in charge (PIC).

(12) Reactor calibrations and data.

(13) All line outs, entry errors, changes in mode of operation stamp lines, and end of page line outs will be initialed or signed by the operator.

b. RED:

(1) K-excess measurements, to include experiment worth determinations.

(2) Actions which affect reactivity:

(a) Core movement.

(b) Fuel movement.

(c) Control rod physical removal for maintenance.

(d) Experiment loading and removal from the CET, PTS, pool, or core.

c. GREEN:

(1) Any reactor malfunctions noted upon discovery/occurrence with a second entry noting corrective action has been completed.

(2) Additional items entered at the discretion of the operator such as addition of makeup water to the reactor pool, etc.

(3) Any Technical Specification required equipment taken out of service for any reason. A second entry is made when the unit is returned to service.

6. When an operation requiring entry into the logbook falls under more than one color code, the color to be used will be determined via the following order of precedence: RED - GREEN - BLACK/BLUE-BLACK.

This procedure has been approved by the Reactor Facility Director

APPROVED

Reactor Facility Director — 29 May 91 Date

Reviewed by the Reactor Staff

| | |
|---|---|
| T. Wright | 5-29-91 |
| Wright | Date |
| | 5-29-91 |
| George | Date |
| | 29 May 91 |
| Forsbacka | Date |
| | 29 May 91 |
| Spence | Date |
| M. Laughery | 29 May 91 |
| Laughery | Date |
| John Nguyen | 29 May 91 |
| Nguyen | Date |
| | 29 May 91 |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991

Date

# DAILY OPERATIONAL STARTUP CHECKLIST

Checklist number ____________________ Date ____________________

SRO On Call ____________________ Supervised by ____________________

Assisted by ____________________

____________________

Operators ____________________ Time completed ____________________

## I. EQUIPMENT ROOM (Room 3152)

1. Air compressor pressure (psi) .................... ________
2. Air compressor water trap drained .................... ________
3. Air dryer Operating .................... ________
4. Doors 231,231A, 3152, and roof hatch SECURED.................... ________

## II. LOBBY AREA

Lobby audio alarm turned off .................... ________

## III. EQUIPMENT ROOM (Room 2158)

1. Prefilter differential pressure .................... ________
2. Primary discharge pressure (psi) .................... ________
3. Demineralizer flow rates set to 6 gpm .................... ________
4. Stack roughing filter (inches of water) .................... ________
5. Stack absolute filter (in es of water) .................... ________
6. Visual inspection of area .................... ________
7. Door 2158 SECURED .................... ________

## IV. PREPARATION AREA

Visual inspection of area .................... ________

## V. REACTOR ROOM (Room 3161)

1. Transient rod air pressure (psi) .................... ________
2. Shielding doors bearing air pressure (psi) .................... ________
3. Visual inspection of core and tank .................... ________
4. Number of fuel elements and control rods in tank storage
   fuel elements .................... ________
   control rods .................... ________
5. Air particulate monitor (CAM)
   (a) Operating and Tracing .................... ________
   (b) Alarm test completed, damper closure verified.................... ________
6. Door 3162 SECURED .................... ________
7. Stack gas monitor quality assurance checked .................... ________

## VI. REACTOR CONTROL ROOM (Room 3160)

1. Emergency air dampers reset .................. ____________
2. Console recorders dated .................. ____________
3. Stack flow and fuel temperature recorders dated .................. ____________
4. Logbook dated and reviewed .................. ____________
5. Water monitor box (resistivity must be > 0.5 Mohm-cm)
   (a) Background activity(cpm) .................. ____________
   (b) Water monitor box resistivity [Mohm-cm] .................. ____________
   (c) DM1 resistivity [Mohm-cm] .................. ____________
   (d) DM2 resistivity [Mohm-cm] .................. ____________
6. Stack gas flow rate [Kcfm] .................. ____________
7. Stack linear flow rate (Kft/min) .................. ____________
8. Gas stack monitor
   (a) Background (cpm) .................. ____________
   (b) Alarm check .................. ____________
   (c) High alarm set to 800 MPC Ar-41 .................. ____________
9. Radiation monitors

| Monitor | Alarm Point Functional | Reading (mR/hr) | Alarm Setting (mR/hr) |
|---|---|---|---|
| (a) R-1 | ____________ | ____________ | ____________ 500 |
| (b) R-2 | ____________ | ____________ | 10 |
| (c) R-3 | ____________ | ____________ | 10 |
| (d) R-5 | ____________ | ____________ | ____________ 50 |
| (e) E-3 | ____________ | ____________ | 10 |
| (f) E-6 | ____________ | ____________ | 10 |

10. TV monitors on .................. ____________
11. CAM high level audible alarm check .................. ____________
12. Water temperature (inlet) .................. ____________
13. Water level log completed .................. ____________
14. Time delay operative .................. ____________
15. Source level power greater/equal to 0.5 cps. .................. ____________
16. Prestart operability checks performed .................. ____________
17. Interlock Tests

| | | | |
|---|---|---|---|
| (a) Rod raising, SS mode | ____________ | (e) 1 kW/Pulse mode | ____________ |
| (b) Rod raising, Pulse mode | ____________ | (f) NM-1000 HV | ____________ |
| (c) Source RWP | ____________ | (g) Inlet Temp | ____________ |
| (d) Period RWP | ____________ | | |

18. SCRAM checks (at least one per rod)

| | | | |
|---|---|---|---|
| (a) % Power 1 | ______ ______ | (h) Reactor key | ____________ |
| (b) % Power 2 | ____________ | (i) Manual | ____________ |
| (c) Fuel temp 1 | ____________ | (j) Emergency Stop | ____________ |
| (d) Fuel temp 2 | ____________ | (k) Timer | ____________ |
| (e) HV loss 1 | ____________ | (l) CSC Watchdog | ____________ |
| (f) HV loss 2 | ____________ | (m) DAC Watchdog | ____________ |
| (g) Pool level | ____________ | | |

19. Zero power pulse .................. ____________

This procedure has been approved by the Reactor Facility Director

M. H. Moore 29 May 91
Reactor Facility Director Date

Reviewed by the Reactor Staff

| | |
|---|---|
| T Wright | 5-29-91 |
| Wright | Date |
| R George | 5-29-91 |
| George | Date |
| [illegible] | 29 May 91 |
| Forsbacka | Date |
| B Spence | 29 May 91 |
| Spence | Date |
| M. Laughery | 29 May 91 |
| Laughery | Date |
| John Nguyen | 29 May 91 |
| Nguyen | Date |
| Chris Owens | 29 MAY 91 |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991
Date

# DAILY SAFETY CHECKLIST

Checklist number ____________ Date ____________
SRO On Call ____________ Supervised by ____________
Assisted by ____________
____________
Operators ____________ Time completed ____________

## I. EQUIPMENT ROOM (Room 3152)

1. Air compressor pressure (psi) .................... ________
2. Air compressor water trap drained .................... ________
3. Air dryer Operating .................... ________
4. Doors 231,231A, 3152, and roof hatch SECURED.................... ________

## II. EQUIPMENT ROOM (Room 2158)

1. Prefilter differential pressure .................... ________
2. Primary discharge pressure (psi) .................... ________
3. Demineralizer flow rates set to 6 gpm .................... ________
4. Stack roughing filter (inches of water) .................... ________
5. Stack absolute filter (inches of water) .................... ________
6. Visual inspection of area .................... ________
7. Door 2158 SECURED .................... ________

## III. PREPARATION AREA

Visual inspection of area .................... ________

## IV. REACTOR ROOM (Room 3161)

1. Transient rod air pressure (psi) .................... ________
2. Shielding doors bearing air pressure (psi) .................... ________
3. Visual inspection of core and tank .................... ________
4. Number of fuel elements and fuel elements .................... ________
   control rods in tank storage control rods .................... ________
5. Air particulate monitor (CAM)
   (a) Operating and Tracing .................... ________
   (b) Alarm test completed, damper closure verified.................... ________
6. Door 3162 SECURED .................... ________
7. Stack gas monitor quality assurance checked .................... ________

## V. LOBBY AREA

| | |
|---|---|
| Lobby audio alarm turned off .............................................. | ________ |

## VI. REACTOR CONTROL ROOM (Room 3160)

1. Emergency air dampers reset .............................................. ________
2. Console recorders dated .............................................. ________
3. Stack flow and fuel temperature recorders dated .............................................. ________
4. Logbook dated and reviewed .............................................. ________
5. Water monitor box (resistivity must be > 0.5 Mohm-cm)
   - (a) Background activity(cpm) .............................................. ________
   - (b) Water monitor box resistivity [Mohm-cm] .............................................. ________
   - (c) DM1 resistivity [Mohm-cm] .............................................. ________
   - (d) DM2 resistivity [Mohm-cm] .............................................. ________
6. Stack gas flow rate [Kcfm] .............................................. ________
7. Stack linear flow rate (Kft/min) .............................................. ________
8. Gas stack monitor .............................................. ________
   - (a) Background (cpm) .............................................. ________
   - (b) Alarm check .............................................. ________
   - (c) High alarm set to 800 MPC Ar-41 .............................................. ________
9. Radiation monitors

| Monitor | Alarm Point Functional | Reading (mR/hr) | | Alarm Setting (mR/hr) |
|---|---|---|---|---|
| (a) R-1 | ________ | ________ | ________ | 500 |
| (b) R-2 | ________ | ________ | | 10 |
| (c) R-3 | ________ | ________ | | 10 |
| (d) R-5 | ________ | ________ | ________ | 50 |
| (e) E-3 | ________ | ________ | | 10 |
| (f) E-6 | ________ | ________ | | 10 |

10. TV monitors on .............................................. ________
11. CAM high level audible alarm check .............................................. ________
12. Water temperature (inlet) .............................................. ________
13. Water level log completed .............................................. ________
14. Source level power greater/equal to 0.5 cps. .............................................. ________

This procedure has been approved by the Reactor Facility Director

Reactor Facility Director 29 May 91
Date

Reviewed by the Reactor Staff

| | |
|---|---|
| T. Wright | 5-29-91 |
| Wright | Date |
| R George | 5-29-91 |
| George | Date |
| | 29 May 91 |
| Forsbacka | Date |
| | 29 May 91 |
| Spence | Date |
| M. Laughery | 29 May 91 |
| Laughery | Date |
| John Nguyen | 29 May 91 |
| Nguyen | Date |
| Chris Owens | 29 MAY 91 |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991
Date

# NUCLEAR INSTRUMENTATION SET POINTS

## GENERAL

These set points may be adjusted for a specific operation by the RFD or ROS but in no case may they be set at a point non-conservative to the technical specifications.

## SPECIFIC

The following are channel or monitor set points (alarm, scram, rod withdrawal prevent).

1. Scrams:
   a. Fuel Temperature 1 & 2: 575 degrees C
   b. High Flux 1 & 2: 110% (1.1 MW)
   c. Safe Chambers 1 & 2 HV Loss: Loss of 20%
   d. Pulse Timer: Less than 15 seconds
   e. Steady State Timer: as necessary
2. Rod Withdrawal Prevents:
   a. Period: 3 seconds
   b. 1 KW (Pulse Mode): 1 KW
   c. Source: 0.5 CPS
   d. Water Inlet Temperature: 60 degrees C
   e. Fission Chamber HV Loss: 20%
3. Alarms:
   a. RAMS: As directed in procedures
   b. CAMS: 10,000 CPM
   c. Stack Gas: 800 MPC Ar-41
   d. Water Monitor Box Gamma: 7000 CPM
   e. Criticality Monitor (R5): 50 mR/hr day - 20 mR/hr night

This procedure has been approved by the Reactor Facility Director

APPROVED

29 May 91

Reactor Facility Director Date

Reviewed by the Reactor Staff

| | |
|---|---|
| T Wright | 5-29-91 |
| Wright | Date |
| R George | 5-29-91 |
| George | Date |
| | 29 May 91 |
| Forsbacka | Date |
| | 29 May 91 |
| Spence | Date |
| M Laughery | 29 May 91 |
| Laughery | Date |
| John Nguyen | 29 May 91 |
| Nguyen | Date |
| | 29 MAY 91 |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991

Date

# K-EXCESS

1. Withdraw SAF and SHIM rods 100% ~~and withdraw the TRANS rod 25%.~~ EFFECTIVE 16 JAN 92

2. Use the REG rod to bring the reactor to cold critical at 5 watts. If criticality can not be reached with the REG rod full out, use the TRANS rod to bring to critical.

3. When power is stabilized at 5 watts, record rod positions in reactor operations logbook, entering all information in red ink.

4. Using rod worth curves, compute K-excess for the core position* used and record in the reactor operations logbook .

* Note: Use the curves for infinite water when doing K-excess between positions 300 and 700.

This procedure has been approved by the Reactor Facility Director

APPROVED 5/29/91

Reactor Facility Director Date

Reviewed by the Reactor Staff

| | |
|---|---|
| T. Wright | 5.29.91 |
| Wright | Date |
| R George | 5-29-91 |
| George | Date |
| | 24 May 91 |
| Forsbacka | Date |
| | 29 May 91 |
| Spence | Date |
| M. Laughery | 29 May 91 |
| Laughery | Date |
| John Nguyen | 29 May 91 |
| Nguyen | Date |
| Chris Owens | 29 MAY 91 |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991

Date

# STEADY STATE OPERATION

## GENERAL

The reactor shall not be operated at a power greater than 1.0 MW.

## SPECIFIC

1. Set the mode switch to manual mode and clear all warning messages and scrams.

2. For runs greater than 200 KW, adjust alarm points on R-1 and R-5 to full scale.

3. Raise control rods with the appropriate banking, taking into consideration the location in the pool, power level, and experimental requirements.

4. If final approach to critical is to be made in Auto mode, perform the following:

    a. Set the the thumb wheel dials to the desired power.

    b. Raise the rods to the appropriate banking.

    c. Select the rods that are to servoed.

    d. Make sure that all rods that will be servoed have been raised at least 5%.

    e. Enter Auto mode.

5. Scram the reactor at the end of the run using the manual or timer scram.

6. Ensure the appropriate entries have been made in the operations logbook.

7. If no further steady state runs, square waves or pulses are anticipated, adjust R-1 and R-5 alarm points to their normal settings.

This procedure has been approved by the Reactor Facility Director

______________________ 5/29/91
Reactor Facility Director Date

Reviewed by the Reactor Staff

| | |
|---|---|
| T Wright | 5-29-91 |
| Wright | Date |
| R George | 5-29-91 |
| George | Date |
| | 29 May 91 |
| Forsbacka | Date |
| | 29 May 91 |
| Spence | Date |
| M Laughery | 29 May 91 |
| Laughery | Date |
| John Nguyen | 29 May 91 |
| Nguyen | Date |
| Chris Owens | 29 MAY 91 |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991
Date

# SQUARE WAVE OPERATION (Subcritical)

## GENERAL

The square wave mode will not be used above a demand power of 250KW.

## SPECIFIC

1. Set R1 and R5 to full scale

2. Determine the transient rod critical position using the core position, the final transient rod position, the rod curves and the equation below. Note that a square wave insertion can not exceed 75 cents.

CRITICAL POSITION($) = FINAL POSITION($) - INSERTION ($)

3. Apply air to the transient rod and raise the anvil to the citical position that was calculated above.

4. Bring the reactor cold critical using the three standard control rods; use a rod configuration commensurate with the core position and experimental requirements. If Auto Mode is used, select the rods to be used. Ensure that these rods have been raised at least 5% before entering Auto Mode. Set the cold critical power level on the Power Demand thumb wheels and enter Auto Mode.

5. Stabilize the reactor in Manual Mode.

6. Set power demand thumb wheels to desired power level.

7. Select the standard control rods to be servoed. Make sure that all control rods to be servoed have been raised at least 5%.

8. Scram the transient rod.

9. Raise the anvil to the desired final position.

10. Allow the power level to fall below 10 watts.

11. Switch into Square Wave mode.

12. Depress Fire button.

13. As the power level approaches the power demand level, the console will switch into Auto Mode. If power can not reach the demand power, it will automatically change to manual mode. At this time, either switch to Auto Mode or bring the reactor to the desired power level manually.

14. Scram the reactor at the end of the run using the manual or timer scram.

15. Ensure all pertinent information has been entered in the reactor operations log-book.

16. If no further steady state runs, square waves or pulses are anticipated, adjust R-1 and R-5 alarm points to their normal settings.

This procedure has been approved by the Reactor Facility Director

APPROVED 5/29/91

Reactor Facility Director Date

Reviewed by the Reactor Staff

| | |
|---|---|
| 5-29-91 | |
| Wright | Date |
| 5-29-91 | |
| George | Date |
| 29 May 91 | |
| Forsbacka | Date |
| 29 May 91 | |
| Spence | Date |
| M. Laughery 29 May 91 | |
| Laughery | Date |
| John Nguyen 29 May 91 | |
| Nguyen | Date |
| 29 May 91 | |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991

Date

# SQUARE WAVE OPERATION (Critical)

## GENERAL

The square wave mode will not be used above a demand power of 250KW.

## SPECIFIC

1. Set R-1 and R-5 to full scale.

2. Bring the reactor cold critical using the three standard control rods; use a rod configuration commensurate with the core position and experimental requirements. If Auto Mode is used, select the rods to be used, ensure that these rods have been raised at least 5% before entering Auto Mode, set the cold critical power level on the Power Demand thumb wheels, and enter Auto Mode.

3. Determine TRANS rod anvil setting for desired insertion. Insertion cannot exceed 75 cents. Raise the anvil to that setting.

4. Stabilize the reactor in Manual Mode.

5. Set power demand thumb wheels to desired power level.

6. Select the standard control rods to be servoed. Make sure that all control rods to be servoed have been raised at least 5%.

7. Switch into Square Wave mode.

8. Depress Fire button.

9. As the power level approaches the power demand level, the console will switch into Auto Mode. If power can not reach the demand power, it will automatically change to Manual Mode. At this time, either switch to Auto Mode or bring the reactor to the desired power level manually.

10. Scram the reactor at the end of the run using the manual or timer scram.

11. Ensure all pertinent information has been entered in the reactor operations logbook.

12. If no further steady state runs, square waves, or pulses are anticipated, adjust R-1 and R-5 alarm points to their normal settings.

This procedure has been approved by the Reactor Facility Director

APPROVED

M h Ng 9/29/91

Reactor Facility Director Date

Reviewed by the Reactor Staff

| | |
|---|---|
| TWright | 5-29-91 |
| Wright | Date |
| R George | 5-29-91 |
| George | Date |
| | 29 May 91 |
| Forsbacka | Date |
| D Spence | 29 May 91 |
| Spence | Date |
| M Laughery | 29 May 91 |
| Laughery | Date |
| John Nguyen | 29 May 91 |
| Nguyen | Date |
| Chris Owens | 29 May 91 |
| Owens | Date |
| | |
| | Date |
| | |
| | Date |

Reviewed by RRFSC 24 SEP 1991

Date

# PULSE OPERATION (CRITICAL)

## GENERAL

Pulses above $2.00 must be approved by the RFD (prior to pulse initiation). Specification on the RUR may be used to meet this requirement.

## SPECIFIC

1. Set the alarm points on R-1 and R-5 (criticality monitor) to full scale.

2. Bring the reactor cold critical using the three standard control rods; use a rod configuration commensurate with core position and experimental requirements. Note: A series of repetitive pulses may be fired using the same rod positions on the same day as long as the reactor power is not increasing and is less than 1 kW.

3. Stabilize in the manual mode.

4. Raise the transient rod anvil to the desired pulse position. (This position is obtained from the control rod worth curves for the appropriate core operating position)

5. Select the proper pulse detector according to the table below. **If the Cerenkov detector is selected, turn off the reactor room and tank lights.**

| |
|---|
| Detector 1 = Pulse Ion (Maximum insertion = $2.00) |
| Detector 2 = Cerenkov (Maximum insertion = $4.00) |

6. Enter Pulse Mode and enter an identifying string at the prompt. The power level must be below 1 kW to enter Pulse Mode.

7. Fire the pulse by depressing the " Fire" button on the reactor console.

8. Record the appropriate data in the reactor operations logbook from the pulse display.

9. Reset R-1 and R-5 to their normal alarm points when pulsing, square wave, or steady state operations are complete.

This procedure has been approved by the Reactor Facility Director

APPROVAL

M L Meyer 5/29/91

Reactor Facility Director Date

Reviewed by the Reactor Staff

T Wright 5-29-91

Wright Date

L George 5-29-91

George Date

29 May 91

Forsbacka Date

29 May 91

Spence Date

Laughery 29 May 91

Laughery Date

John Nguyen 29 May 91

Nguyen Date

Chris Owen 29 May 91

Owens Date

Date

Date

Reviewed by RRFSC 24 SEP 1991

Date

# PULSE OPERATION (SUBCRITICAL)

## GENERAL

Pulses above $2.00 must be approved by the RFD (prior to pulse initiation). Specification on the RUR may be used to meet this requirement.

## SPECIFIC

1. Set the alarm points on R-1 and R-5 (criticality monitor) to full scale.

2. Given a core position, set the transient rod at a position corresponding to the dollar value determined by the following equation:

   $ Value = Total worth ($) Transient rod - Desired pulse ($) Value

3. Bring the reactor cold critical using the three standard control rods, use a rod configuration commensurate with core position and experimental requirements. Note: A series of repetitive pulses may be fired using the same rod positions on the same day as long as the reactor power is not increasing and is less than 1 kW.

4. Stabilize in the manual mode.

5. Select the proper pulse detector according to the table below. **If the Cerenkov detector is selected, turn off the reactor room and tank lights.**

| | |
|---|---|
| Detector 1 = Pulse Ion | (Maximum insertion = $2.00) |
| Detector 2 = Cerenkov | (Maximum insertion = $4.00) |

6. Scram the Transient rod.

7. Raise the Transient rod anvil to 100%.

8. Let the power decay to approximately 1 watt or less.

9. Enter Pulse Mode and enter an identifying string at the prompt.

10. Fire the pulse by depressing the " Fire" button on the reactor console.

11. Record the appropriate data in the reactor operations logbook from the pulse display.

12. Reset R-1 and R-5 to their normal alarm points when pulsing, square wave, or steady state operations are complete.

This procedure has been approved by the Reactor Facility Director

M. L. Moriz 5/29/91

Reactor Facility Director Date

Reviewed by the Reactor Staff

| | |
|---|---|
| T Wright | 5-29-91 |
| Wright | Date |
| L George | 5-29-91 |
| George | Date |
| Forsbacka | 29 May 91 |
| Forsbacka | Date |
| Spence | 29 May 91 |
| Spence | Date |
| M. Laughery | 29 May 91 |
| Laughery | Date |
| John Nguyen | 29 May 91 |
| Nguyen | Date |
| Chas Owen | 29 MAY 91 |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991

Date

# WEEKLY OPERATIONAL INSTRUMENT CHECKLIST

CHECKLIST # ____________ DATE ____________
SUPERVISED BY ____________
ASSISTED BY ____________ REVIEWED BY ____________

## I. WATER LEVEL INDICATOR

A. In pool, east side, depress float on water level indicator.................................. ______
B. Observe scram on console .................................................................................. ______

## II. WATER RESISTIVITY

List resistivity readings for previous week from daily startup checklists. Determine the average at each point is > 0.5 Mohm-cm.

| | MON | TUE | WED | THU | FRI | AVG |
|---|---|---|---|---|---|---|
| Monitor Box | ____ | ____ | ____ | ____ | ____ | ____ |
| DM1 | ____ | ____ | ____ | ____ | ____ | ____ |
| DM2 | ____ | ____ | ____ | ____ | ____ | ____ |

## III. RADIATION ALARMS

A. Test alarm functions for high level and failure

| Monitor | Failure alarm functional | HIGH Level alarm functional |
|---|---|---|
| R-1 | ______ | ______ |
| R-2 | ______ | ______ |
| R-5 (criticality) | ______ | ______ |
| E-3 | ______ | ______ |
| E-6 | ______ | ______ |
| Reactor Room CAM | ______ | ______ |
| Gas Stack Monitor | ______ | ______ |

B. Reset alarms.................................................................. ______

## IV. OTHER

A. Top lock key seals at Security Desk and at LOG verified intact.............. ______
B. Change Filter in the Stack Gas Monitor ............................................... ______

This procedure has been approved by the Reactor Facility Director

ML Moo Jr 5/29/91

Reactor Facility Director Date

Reviewed by the Reactor Staff

| | |
|---|---|
| TWright | 5-29-91 |
| Wright | Date |
| | 5-29-91 |
| George | Date |
| | 29 May 91 |
| Forsbacka | Date |
| | 29 May 91 |
| Spence | Date |
| M Laughery | 29 May 91 |
| Laughery | Date |
| | 29 May 91 |
| Nguyen | Date |
| | 29 MAY 91 |
| Owens | Date |
| | |
| | Date |
| | |
| | Date |

Reviewed by RRFSC 24 SEP 1991

Date

# DAILY OPERATIONAL SHUTDOWN CHECKLIST

Checklist No. ______________ Date ______________________
Time Completed ___________ Supervised by ______________
Assisted by ________________
__________ ________________

## I. REACTOR ROOM (Room 3161)

1. All rod drives DOWN ........................................ ________
2. Carriage lights OFF ........................................ ________
3. Door 3162 SECURED ........................................ ________
4. Door 3161 locked with key ........................................ ________

## II. EQUIPMENT ROOM (Room 3152)

1. Distillation unit discharge valve CLOSED ........................................ ________
2. Air dryer OPERATIONAL ........................................ ________
3. Doors 231, 231A, 3152 and Roof hatch SECURED ........................................ ________

## III. EQUIPMENT ROOM (Room 2158)

1. Primary discharge pressure (PSI) ........................................ ________
2. Demineralizer flow rates set to 6 GPM ........................................ ________
3. Visual inspection for leaks ........................................ ________
4. Door 2158 SECURED ........................................ ________

## IV. PREPARATION AREA

1. ER 2 plug door CONTROL LOCKED;
   Door closed; and handwheel PADLOCKED ........................................ ________
2. ER 2 lights ON and rheostat at 10% ........................................ ________
3. ER 1 plug door CONTROL LOCKED;
   Door closed; and handwheel PADLOCKED ........................................ ________
4. ER 1 lights ON and rheostat at 10% ........................................ ________
5. Visual inspection of area ........................................ ________

## V. LOBBY ALARM

| |
|---|
| Lobby alarm audio ON .............................................................. ________ |

## VI. REACTOR CONTROL ROOM (Room 3160)

1. Reactor tank lights OFF .............................................................. ________
2. Console chart recorder pens raised .............................................. ________
3. TV monitors OFF .......................................................................... ________
4. Console LOCKED, and all required keys returned to lock box ........................................................................................ ________
5. Diffuser and secondary pumps OFF ................................................ ________
6. Purification and primary pumps ON ............................................... ________
7. Reactor monthly usage summary completed ................................... ________
8. Exposure room camera power supply turned OFF ........................... ________
9. Radiation monitors ........................................................................ ________

| MONITOR | READING | HIGH LEVEL ALARM SETTING (mr/hr) |
|---|---|---|
| a. R-1 | ________ | 20 ________ |
| b. R-2 | ________ | N/A |
| c. R-3 | ________ | N/A |
| d. R-5 | ________ | 20 ________ |
| e. E-3 | ________ | N/A |
| f. E-6 | ________ | N/A |
| g. R-6 | ________ | N/A |

This procedure has been approved by the Reactor Facility Director

5/29/91

Reactor Facility Director Date

Reviewed by the Reactor Staff

T Wright 5.29.91
Wright Date

5-29-91
George Date

29 May 91
Forsbacka Date

29 May 91
Spence Date

M Laughery 29 May 91
Laughery Date

John Nguyen 29 May 91
Nguyen Date

29 MAY 91
Owens Date

Date

Date

Reviewed by RRFSC 24 SEP 1991
Date

# REACTOR ROOM SAFETY

## GENERAL

The following safety procedures will be observed while in the reactor room.

## SPECIFIC

1. Hoist Operations: Perform the following before/during any hoist operations:

   a. Inspect any lifting equipment (ropes, cables, etc.) for wear or damage prior to use.

   b. Ensure that the hoist has a current load testing (within last 12 months) .

   c. Ensure areas beneath the hoist are clear of personnel when operations are underway. This is particularly important when using the hatches between several floors.

   d. Each time a load approaching 10,000 pounds is handled, test the brakes by raising the load a few inches, applying the brakes, and checking for slippage.

   e. Ensure a load is not lowered below the point where two full wraps of cable remain on the drum.

   f. Ensure no tools or poles longer than 10 feet are raised vertically in the reactor room while the power rails are energized.

2. Mercury or mercury compounds in any form are not allowed in the reactor room at any time.

This procedure has been approved by the Reactor Facility Director

M L Wright APPROVED 5/29/91
Reactor Facility Director Date

Stephen G. Roth 28 May 91
Chairman, Safety and Health Department Date

Reviewed by the Reactor Staff

| | |
|---|---|
| T. Wright | 5-29-91 |
| Wright | Date |
| R George | 5-29-91 |
| George | Date |
| M Forsbacka | 29 May 91 |
| Forsbacka | Date |
| Spence | 29 May 91 |
| Spence | Date |
| M Laughery | 29 May 91 |
| Laughery | Date |
| John Nguyen | 29 May 91 |
| Nguyen | Date |
| Owens | 29 May 91 |
| Owens | Date |
| | |
| | Date |
| | |
| | Date |

Reviewed by RRFSC 24 SEP 1991
Date

# STACK GAS MONITOR PROCEDURE

## GENERAL

This procedure specifies all the requirements for operation of the Stack Gas Monitor (SGM) in the reactor room. This instrument is used to sample and measure the gaseous effluent in the building exhaust system.

## SPECIFIC

A quality assurance check (QA) is performed daily, prior to reactor operations, as part of the reactor start-up. This check is performed in the following manner:

1. The particulate filter is changed if necessary.

2. The front cover of the detector shield is removed and the check source is inserted all the way in to the face of the detector. The blue alert light should come on as the count rate rises above the alert setpoint. The red high level alert light and bell should come on as the count rate rises above the high level set point. The audible alarm can be silenced by pushing the rec button on the front of the SGM cabinet.

3. The detector voltage system set point is checked by pressing menu function 5 and then menu function 1. The high voltage set point is displayed under item 7 on the display screen. Press 0 twice to return to the main menu.

4. The air sampling flow rate (should be greater than 3.5 cubic feet per minute).

5. The counts per minute reading shown on the display should be checked against the plot of counts per minute versus Julian date to determine if it falls within the plus or minus 5% deviation lines for the detector and check source. If it does, the check source should then be removed and the detector cover replaced.

If the counts per minute consistently fall outside the +/-5% window, it is considered an abnormality and should be reported immediately to the Reactor Facility Director and to the Safety and Health Department.

6. The SGM alarms will be acknowledged by pushing the "ACK" button on the SGM keyboard.

This procedure has been approved by the Reactor Facility Director

APPROVED 5/29/91
Reactor Facility Director Date

28 May 91
Chairman, Safety and Health Department Date

Reviewed by the Reactor Staff

| | |
|---|---|
| T Wright | 5.29.91 |
| Wright | Date |
| R George | 5-29-91 |
| George | Date |
| | 29 May 91 |
| Forsbacka | Date |
| | 29 May 91 |
| Spence | Date |
| M Laughery | 29 May 91 |
| Laughery | Date |
| | 29 May 91 |
| Nguyen | Date |
| | 29 May 91 |
| Owens | Date |
| | Date |
| | Date |

Reviewed by RRFSC 24 SEP 1991
Date

# AIR PARTICULATE MONITOR PROCEDURE

## GENERAL

This procedure specifies how to test the CAM to ensure proper operation of this monitoring device.

## SPECIFIC

This procedure uses a radioactive source to test the alarm set points of the CAM.

### 1. OPERATING and TRACING

Observe to see that the CAM is operating and tracing.

### 2. ALARM TEST WITH SOURCE

Open the detector chamber door and slowly bring a radioactive source near the detector. Observe the meter on the front of the CAM. The yellow light will be come on at approximately 4,000 counts per minute. The red light will come on at approximately 10,000 counts per minute, the alarm will sound and the dampers will close. Reset the alarm, close the chamber door and replace the source in the drawer.

# ATTACHMENT C

Amendment No. 21 to Facility Operating License

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE

DOCKET NO. 50-170

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 21
License No. R-84

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

   A. The application for amendment to Facility Operating License No. R-84 filed by the Armed Forces Radiobiology Research Institute (the licensee), of April 30, 1990, as supplemented on December 17, 1990, March 5, 1991, May 17, 1991, August 16, 1991, and September 10, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations, and all applicable requirements have been satisfied; and

   F. Prior notice of this amendment was not required by 10 CFR 2.105(a)(4), and publication of notice for this amendment is not required by 10 CFR 2.106(a)(2).

2. Accordingly, paragraph 2.C.(1) of Facility Operating License No. R-84 is hereby amended to read as follows:

   (1) Maximum Power Level

   AFRRI may operate the reactor at steady state power levels up to a maximum of 1100 kilowatts (thermal), and at pulse power levels not to exceed a pulse reactivity insertion of 4.00 dollars.

3. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment, and paragraph 2.C.(2) of License No. R-84 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 21, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Frank J. Miraglia

for Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A Technical
Specifications Changes

Date of Issuance: October 8, 1991

ENCLOSURE TO LICENSE AMENDMENT NO. 21

FACILITY OPERATING LICENSE NO. R-84

DOCKET NO. 50-170

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

| Remove | Insert |
|---|---|
| i | i |
| 2 | 2 |
| 7 | 7 |
| 8 | 8 |
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| 32 | 32 |
| 34 | 34 |
| 35 | 35 |

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.

## 3.0 LIMITING CONDITIONS FOR OPERATIONS

### 3.1 REACTOR CORE PARAMETERS

#### 3.1.1 STEADY STATE OPERATION

Applicability

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

Objective

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

Specifications

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

Basis

Thermal and hydraulic calculations and operational experience indicate that TRIGA fuel may be safely operated up to power levels of at least 1.5 megawatts with natural convective cooling.

#### 3.1.2 PULSE MODE OPERATION

Applicability

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

Objective

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

Specification

The maximum step insertion of reactivity shall be 2.8% $\Delta k/k$ ($4.00) in the pulse mode.

Basis

Based upon the Fuchs-Nordheim mathematical model (cited by C.E. Clifford et al. in the April 1961 GA Report #2119, "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 settings that protect the safety limit. The 50°C margin to the Limiting Safety

System Setting and the 450°C margin to the safety limit amply allow for uncertainties due to extrapolation of measured data, accuracy of measured data, and location of instrumented fuel elements in the core.

3.1.3 REACTIVITY LIMITATIONS

Applicability

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

Objective

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

Specifications

a. The reactor shall not be operated with the maximum available excess reactivity above cold critical with or without all experiments in place greater than \$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 \$0.50 (0.35% $\Delta k/k$) for any condition of operation.

Basis

a. The limit on available excess reactivity establishes the maximum power if all control elements are removed.

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

3.1.4 SCRAM TIME

Applicability

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

Objective

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

Specification

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

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 the pool water and the water-conditioning system.

Objective

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

Specifications

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

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

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 isolation.

Objective

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

Specification

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

Basis

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

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 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 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 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. Building construction and gaskets around doorways help restrict leakage of air into or out of the reactor oom. 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

The 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 analysed in the Safety Analysis Report.

Specifications

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

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

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

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

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

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 an individual fuel element reduces the safety margin by 10%, at most. The hydrogen-to-zirconium ratio of 1.7 will produce a maximum pressure within the cladding well below the rupture strength of the cladding.

The local powe. 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.

## 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.

## 5.8 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.

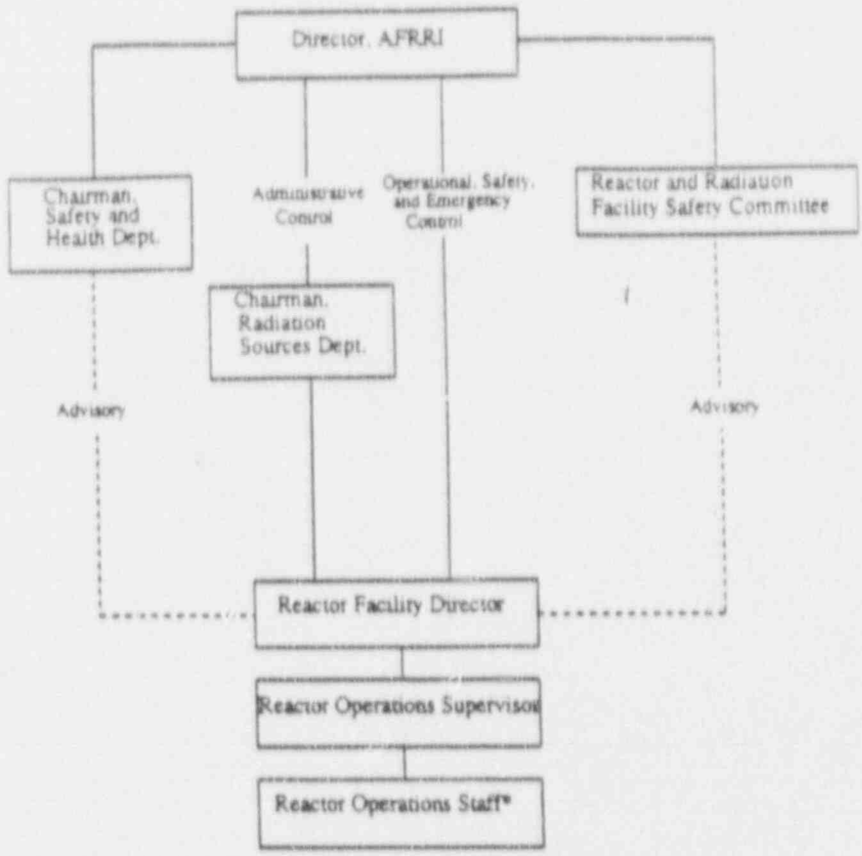



Figure 1. Organization of Personnel for Management and Operation of the AFRRI Reactor Facility.

*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 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 [illegible] 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

b. Maintenance activities that could affect the reactivity of the reactor shall be accomplished under the supervision of an SRO.

c. A list of the names and telephone numbers of the following personnel shall be readily available to the operator on duty:

1. Management personnel (Reactor Facility Director, AFRRI Director)

2. Radiation safety personnel (Head, Safety and Health Department)

3. Other operations personnel (Reactor Staff, ROS)

6.1.4 TRAINING OF PERSONNEL

A training and retraining program will be maintained, to insure adequate levels of proficiency in persons involved in the reactor and reactor operations.

## 6.2 REVIEW AND AUDIT - THE REACTOR AND RADIATION FACILITY SAFETY COMMITTEE (RRFSC)

### 6.2.1 COMPOSITION AND QUALIFICATIONS

6.2.1.1 Composition

a. Regular RRFSC Members (Permanent Members)

(1) The following shall be members of the RRFSC.

(a) Chairman, Safety and Health Department, AFRRI

(b) Reactor Facility Director, AFRRI

(2) The following shall be appointed to the RRFSC by the Director, AFRRI:

(a) Chairman as appointed by the AFRRI Directorate.

(b) One to three non-AFRRI members who are knowledgeable in fields related to reactor safety. At least one shall be a Reactor Operations Specialist, or a Health Physics Specialist.

b. Special RRFSC Members (Temporary Members)

(1) Other knowledgeable persons to serve as alternates in item a(2)(b) above as appointed by the AFRRI Director.

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

c. Nonvoting members as invited by the Chairman, RRFSC.

6.2.1.2 Qualifications

The minimum qualifications for a person on the RRFSC shall be 6 years of professional experience in the discipline or specific field represented. A baccalaureate degree may fulfill 4 years of experience.

6.2.2 FUNCTION AND AUTHORITY

6.2.2.1 Function

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

6.2.2.2 Authority

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

6.2.3 CHARTER AND RULES

6.2.3.1 Alternates

Alternate members may be appointed in writing by the 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.

6.2.3.2 Meeting Frequency

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

6.2.3.3 Quorum

A quorum of the RRFSC for review shall consist of the Chairman (or designated alternate) and two other members (or alternate members), one of which must be a non-AFRRI member. A majority of those present shall be regular members.

6.2.3.4 Voting Rules

Each regular RRFSC member shall have one vote. Each special appointed member shall have one vote. The majority is 51% or more of the regular and special members present and voting.

6.2.3.5 Minutes

Minutes of the previous meeting shall be available to regular members at least 1 week before a regular scheduled meeting.

f. Any other area of Facility operations considered appropriate by the RRFSC or the Director/AFRRI.

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

## 6.3 PROCEDURES

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

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

b. Reactor staff-training program.

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

d. Personnel radiation protection consistent with 10 CFR 20.

e. Implementation of required plans such as the Security Plan and Emergency Plan.

f. Reactor core loading and unloading.

g. Checkout startup, standard operations, and securing facility.

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

## 6.4 REVIEW AND APPROVAL OF EXPERIMENTS

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

a. Reactor Facility Director

b. Safety and Health Department

c. Reactor and Radiation Facility Safety Committee (RRFSC)

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

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

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

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

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

## 6.5 REQUIRED ACTIONS

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

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

b. The safety limit violation shall be reported to the Director of NRC Region I, Office of Inspection and Enforcement (or designate); the Director, AFRRI; and 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 on facility components, structures, or systems, and (3) corrective action taken to prevent or reduce the probability of recurrence.

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

### 6.5.2 REPORTABLE OCCURRENCES

Reportable occurrences as defined in 1.21 (including causes, actual or 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.

a. Prompt Notification With Written Followup. The types of events listed below shall be reported as soon as possible by telephone and confirmed by

# ATTACHMENT D

Routine Reactor Authorizations

Routine Reactor Authorization #1

September 1991

Introduction: The Reactor and Radiation Facility Safety Committee has reviewed and approved the operations described below. These operations have been performed many times in the past and are now considered part of the routine operations for use of the AFRRI reactor.

Authorization: As permitted by NRC license R-84 , the Reactor Facility Director, Reactor Operations Supervisor, and NRC licensed operators may perform measurements, conduct operations, effect maintenance, perform experiments, conduct tours, and conduct operator training within the scope of AFRRI procedures, Technical Specifications, and NRC regulations.

ALARA principles will be followed at all times during the conduct of operations.

**Approved:** ______________________
Reactor Facility Director

for Chairman, SHD

N.W. Manderfield COL, USAF, MSC
Chairman, Reactor and Radiation
Facility Safety Committee

Reactor and Radiation Facility Safety Committee

| | |
|---|---|
| M. L. Moore | CDR J.E. DeCicco, MSC, USN |
| D. Ashby | CAPT C.B. Galley, MSC, USN |
| R. Luersen | |
| M. Voth | |

Routine Reactor Authorization #2

September 1991

Introduction: The Reactor and Radiation Facility Safety Committee has reviewed and approved the dosimetry operations described below. These operations have been safely performed over many years and are classified as routine.

Authorizations: As permitted by NRC license R-84, dosimetry instrumentation and other measuring instrumentation may be used alone, attached to, or included with experiments in the reactor radiation facilities within the site boundary, subject to the limitations imposed by Technical Specifications. All dosimetry devices, equipment or experiments will be used or performed under the supervision of the RFD or his designee. In particular, the dosimetry devices and the experiment shall:

- not be able to cause the release of radioactive gases and aerosols such that the annual isotope concentration limits of Table II, Appendix B, of 10 CFR 20 are exceeded,

- not create inventories of I-131 through I-135 greater than 1.3 curies and Strontium-90 inventories greater than 5 millicuries,

- limit known explosive materials to less than 25 milligrams and its explosive potential shall be determined to be within the design limits of its container,

- be doubly encapsulated if the release of the contained material can cause corrosion to the radiation facility,

- have an absolute worth less than $3.00, and

- have been inspected and approved by a reactor operator prior to performance of the experiment.

The term "dosimetry instrumentation" shall include, but not be limited to: fission chambers, ionization chambers including those with flowing gas, thermoluminesent devices, foils, tablets, and phantoms meeting the limitations listed above.

Such dosimetry instrumentation may also be irradiated separately for the purpose of dosimetric calibration or device evaluation and testing.

ALARA principles will be followed at all times during the design and conduct of experiments.

Approved: M. L. Moore
Reactor Facility Director

Chairman, SHD

N. W. Manderfield
N. W. Manderfield COL, USAF, MSC
Chairman, Reactor and Radiation
Facility Safety Committee

Reactor and Radiation Facility Safety Committee

M. L. Moore

D. Ashby

R. Luersen

M. Voth

CDR J.E. DeCicco, MSC, USN

CAPT C.B. Galley, MSC, USN

Routine Reactor Authorization #3

September 1991

Introduction: The Reactor and Radiation Facility Safety Committee has reviewed and approved the reactor experiment described below. This experiment has been performed on numerous occasions in past years and is considered routine.

Authorizations: As permitted by NRC license R-84, The Reactor Facility Director may permit a principal investigator to irradiate animals, animal tissue and other biological materials in the reactor irradiation facilities subject to the limitations imposed by Technical Specifications. This authorization includes shielding and support materials, sensors, control devices and the use of phantoms and other dosimetry instrumentation authorized under Routine Reactor Authorization #2. In particular, the experiments shall:

- o not be able to cause the release of radioactive gases and aerosols such that the annual isotope concentration limits of Table II, Appendix B, of 10 CFR 20 are exceeded,
- o not create inventories of I-131 through I-135 greater than 1.3 curies and Strontium-90 inventories greater than 5 millicuries,
- o limit known explosive materials to less than 25 milligrams and its explosive potential shall be determined to be within the design limits of its container,
- o be doubly encapsulated if the release of the contained material can cause corrosion to the radiation facility,
- o have an absolute worth less than $3.00, and
- o either have movement precluded or be monitored by a Senior Reactor Operator.

ALARA principles will be followed at all times during the design and conduct of experiments.

Approved: M L Moore
Reactor Facility Director

Chairman, SHD

N.M. Manderfield COL, USAF, MSC
Chairman, Reactor and Radiation
Facility Safety Committee

Reactor and Radiation Facility Safety Committee

M L Moore

M. L. Moore

D. Ashby

R. Luersen

M. Voth

CDR J.E. DeCicco, MSC, USN

C.B. Galley

CAPT C.B. Galley, MSC, USN

Routine Reactor Authorization #4

September 1991

Introduction: The Reactor and Radiation Facility Safety Committee has reviewed and approved the operations described below. These experiments have been performed on numerous occasions in the past and are considered routine.

Authorizations: As permitted by NRC license R-84, the Reactor Facility Director may irradiate biological or non-biological materials with an atomic number less than 93 and any experiment structural support and experiment containers in the reactor irradiation facilities subject to the limitations imposed by Technical Specifications and applicable procedures. In particular, these experiments shall:

- not be able to cause the release of radioactive gases and aerosols such that the annual isotope concentration limits of Table II, Appendix B, of 10 CFR 20 are exceeded,
- not create inventories of I-131 through I-135 greater than 1.3 curies and Sr-90 inventories greater than 5 millicuries,
- limit known explosive materials to less than 25 milligrams and its explosive potential shall be determined to be within the design limits of its container,
- be doubly encapsulated if the release of the contained material can cause corrosion to the radiation facility,
- have an absolute worth less than $3.00, and
- either have movement precluded or be monitored by a Senior Reactor Operator.

ALARA principles will be followed at all times during the design and conduct of experiments.

**Approved**: M L Moore

Reactor Facility Director

[signature]

for Chairman, SHD

H W Manderfield

N.W. Manderfield COL, USAF, MSC
Chairman, Reactor and Radiation
Facility Safety Committee

Reactor and Radiation Facility Safety Committee

M L Moore

M. L. Moore

. Ashby

R Luersen

. Luerson

M W Voth

M. Voth

CDR J.E. DeCicco, MSC, USN

C B Galley

CAPT C.B. Galley, MSC, USN

RSDR

Forsbacka/mjf/51221
Date: 19 SEP 91

MEMORANDUM FOR RRFSC

SUBJECT: Request Concurrence to Classify Krypton Gas Irradiation Experiment Protocol Under Routine Reactor Authorization #4

1. Purpose: The Naval Medical Research Institute (NMRI) has a requirement for a radioactive inert gas for use in the study of tissue inert gas exchange kinetics and bubble dynamics. Krypton when irradiated produces a isotope, namely Kr-85m, which is useful in this research. Inert gasses under pressures up to 2 atm have been irradiated safely under Routine Reactor Authorization # 102 dated September 4, 1990.

2. Experiment Description: See attached letter from LCDR Novotny dated 16 SEP 91.

3. Safety Analysis: A worst case scenario accident analysis involves the complete release of the irradiated Krypton gas after a 1 MW-hr irradiation. The gas irradiation vessel has a volume of 446 ml. At a gas pressure of 40 psig, the vessel will deliver 1214 ml of gas at STP (0.0542 moles). The Neutron Activation Tables (Ed. Gerhard Erdtmann, 1976) allow us to determine the quantities of radioactive Krypton produced after an irradiation at 1 MW for 1 hour. The composition of the gas is as follows:

| Isotope | Abundance | Quantity in Vessel | Product | Half Life |
|---|---|---|---|---|
| $^{78}$Kr | 0.35% | 1.498E-2 g | $^{79}$Kr | 34.9 hr |
| $^{80}$Kr | 2.25% | 9.872E-2 g | $^{81}$Kr | 13.3 sec |
| $^{82}$Kr | 11.6% | 0.5218 g | $^{83}$Kr | 1.86 hr |
| $^{84}$Kr | 57.0% | 2.656 g | $^{85}$Kr | 4.5 hr |
| $^{86}$Kr | 17.3% | 0.8157 g | $^{87}$Kr | 76 min |

The Neutron Activation Tables tell us the decays/second per microgram of target element given a 1 hour irradiation in a thermal neutron flux of $10^{13}$ n/cm$^2$sec, so the total amount of radioactivity produced is as follows:

| Radioisotope | Total Quantity Produced | Concentration in Reactor Room of Released |
|---|---|---|
| $^{79}$Kr | 9.6 microCi | 1.05E-8 microCi/ml |
| $^{81}$Kr | 1.98E4 microCi | 2.16E-5 microCi/ml |
| $^{83}$Kr | 7.39E5 microCi | 8.05E-4 microCi/ml |
| $^{85}$Kr | 5.48E4 microCi | 5.97E-5 microCi/ml |
| $^{87}$Kr | 7.01E3 microCi | 7.64E-6 microCi/ml |

The free volume of the reactor room is 32,400 ft$^3$, so the concentration of radioisotope in the reactor room is found by dividing the total quantity produced by the free volume. Adding the above quantities of radioactive materials yields a total of 0.82 Ci of radioactive krypton.

10 CFR 20, Appendix B specifies the allowed concentrations of radioactive materials in air in restricted and unrestricted areas based on inhalation for 40 hours per week for a 13 week period. These limitations are as follows:

| Radioisotope | Allowed Concentration in Restricted Area | Allowed Concentration in Unrestricted Area |
| --- | --- | --- |
| $^{79}$Kr | 3E-9 microCi/ml | 1E-10 microCi/ml |
| $^{81}$Kr | 1E-6 microCi/ml | 3E-8 microCi/ml |
| $^{83}$Kr | 1E-6 microCi/ml | 3E-8 microCi/ml |
| $^{85}$Kr | 6E-6 microCi/ml | 1E-7 microCi/ml |
| $^{87}$Kr | 1E-6 microCi/ml | 2E-8 microCi/ml |

Given the condition of the total release of the irradiated krypton gas, it would take 15 half lives (27.9 hours) for $^{83}$Kr and 10 half lives (45 hours) for $^{85}$Kr to decay to below the specified allowed concentration in an unrestricted area. After 48 hours, $^{79}$Kr concentrations will still exceed the 10 CFR 20, Appendix B limit to an unrestricted area by a factor of 50. To resume normal operations, it will be necessary to slug discharge the remaining gas through the reactor room dampers and dilute it with the air passing through the reactor gas stack. AFRRI TR83-1, "Safety Analysis of Modifications to Upgrade the Reactor Ventilation System at Armed Forces Radiobiology Research Institute" specifies that 3430 cfm of air is exhausted from the reactor room into the reactor gas stack which has a total flow rate of 35,000 cfm. Simply opening the dampers dilutes the gas by a factor of 10. Opening the dampers for one minute in ten minute intervals lets out approximately one tenth of the gas and further dilutes the remaining gas in the room. Since the air system is designed to change the air in the reactor room 4.4 times per hour, the dampers will need to be opened for 14 one minute periods to completely clear the room of krypton gas.

A straight forward approach to ensure the safe irradiation of krypton gas would be to perform the irradiation with the reactor room air dampers closed to isolate the air in the reactor room. In the event of a gas vessel failure, the room wili remained isolated for 48 hours to allow most of the radioactive gas to decay to acceptable levels for release to an unrestricted area. Since the design basis accident radiation release referred to in the facility safety analysis report is approximately 7 Ci for a ruptured fuel element (Safety Analysis Report for AFRRI TRIGA Mark-F Reactor, page 6-19), the reactor confinement should easily contain the 0.82 Ci that could be released in a gas vessel rupture. After a 48 hour decay period, the dampers will be opened for one minute at ten minute intervals for 2.5 hours to sufficiently dilute the remaining radioactive gas as it is released to an unrestricted area.

These precautions are extremely conservative as 10 CFR 20.103 specifies that the concentration levels are based on a 40 hour per week exposure for 13 weeks. Because the gas will be effectively completely decayed in 14 days, there is no chance of exceeding the quarterly limit. The hazard is further mitigated due to the fact that radioactive krypton is noble gas and only presents a submersion dose hazard unlike fission products which are far more dangerous.

4. Experiment Procedure: The following steps shall be taken during the irradiation of krypton gas in the gas irradiation vessel described earlier:

1) Manually close reactor room dampers, ensure good seal of dampers with a vane anemometer. (DELETED 24 JAN 92)
2) Load krypton gas into irradiation vessel.
3) Irradiate krypton gas at 1 Mw for 1 hour.
4) Monitor gas pressure in vessel to ensure gas has not leaked.

A) If gas has pressure has decreased, stop irradiation immediately. Notify ROS/RFD. Continue to monitor gas pressure. Assume worst case scenario and isolate the reactor room for 48 hours. Following the 48 hour waiting period, open the dampers for one minute at ten minute intervals for 2.5 hours. Leave the dampers open after the 2.5 hours of opening and closing the dampers. At least one reactor operator will be present in the reactor facility until the gas has been released from the reactor room. (SEE ATTACHED CHANGE)

B) If gas has been irradiated without incident, the gas will be cryogenically transferred into transfer vessel. Once the gas is outside of the reactor facility boundaries, the dampers shall be reopened. (CHANGED 24 JAN 92)

**Submitted by:**

Matt Forsbacka
Capt, USAF
Reactor Operations Supervisor

**Approved by Reactor Facility Director:**

Mark Moore
Reactor Facility Director

**Approved by Chairman, Safety and Health Department:**

for Doug Ashby
Chairman, Safety and Health Department

Approved by Reactor and Radiation Facility Safety Committee:

N.W. Manderfield
Colonel, USAF, MSC
Chairman, RRFSC

Doug Ashby
Chairman, Safety and Health Department

Mark Moore
Reactor Facility Director

Mark Voth

Ron Luersen

CDR Joseph DeCicco, MSC, USN

CAPT C.B. Galley, MSC, USN

Routine Reactor Authorization #5

September 1991

Introduction: The Reactor and Radiation Facility Safety Committee has reviewed and approved the operations described below. These experiments have been performed on numerous occasions in the past and are considered routine.

Authorizations: As permitted by NRC license R-84, the Reactor Facility Director may irradiate up to 1000 gm of active or passive electronic components in the reactor irradiation facilities subject to the limitations imposed by Technical Specifications and applicable procedures. In particular, these experiments shall:

- o not be able to cause the release of radioactive gases and aerosols such that the annual isotope concentration limits of Table II, Appendix B, of 10 CFR 20 are exceeded,
- o not create inventories of I-131 through I-135 greater than 1.3 curies and Sr-90 inventories greater than 5 millicuries,
- o limit known explosive materials to less than 25 milligrams and its explosive potential shall be determined to be within the design limits of its container,
- o be doubly encapsulated if the release of the contained material can cause corrosion to the radiation facility,
- o have an absolute worth less than $3.00, and
- o either have movement precluded or be monitored by a Senior Reactor Operator.

ALARA principles will be followed at all times during the design and conduct of experiments.

Approved: M L Moore
Reactor Facility Director

[signature]
For Chairman, SHD

[signature]
N.W. Manderfield COL, USAF, MSC
Chairman, Reactor and Radiation
Facility Safety Committee

Reactor and Radiation Facility Safety Committee

M.L. Moore

M. L. Moore

D. Ashby

R. Luerson

R. Luerson

M. Voth

CDR J.E. DeCicco, MSC, USN

C.B. Galley

CAPT C.B. Galley, MSC, USN

# ATTACHMENT E

10 CFR 50.59 Safety Evaluations of Modifications, Changes, and Enhancements to Procedures or Facilities (Other than Fuel-Follower Control Rods)

ATTACHMENT E-1

Facility Modification Worksheet 1

10 CFR 50.59 Analysis

Proposed Change: Modify Transient Rod Drive Support Structure

Submitted by: Capt Forsbacka  Date: 23 OCT 91

1. Description of change:

Raise transient rod drive to position approximately 60 inches above its current position. This will require a connecting rod which is approximately 60 inches longer than the current connecting rod.

2. Reason for change:

The purpose of this modification is to facilitate the maintenance of the transient rod drive mechanism and to unclutter the space between the pool water and bottom of the carriage. This modification will allow for the complete change out of the transient rod drive mechanism without disrupting the other control rods.

3. Verify that the proposed change does not involve a change to the Technical Specifications or produce an unresolved safety issue as specified in 10 CFR 50.59(a)(2). Attach an analysis to show this.

This change does not involve a Technical Specifications change.

Analysis attached? Yes X

4. The proposed modification constitutes a changes in the facility or an operational procedure as described in the SAR. Describe which (check all that apply).

Procedure ____ Facility X Experiment ____



ATTACHMENT E-1

5. Specify what sections of the SAR are applicable. In general terms describe the necessary updates to the SAR. Note that this description need not contain the final SAR wording.

Section 4.10, Reactor Control Components will need to be modified to describe the new physical position of the transient control rod drive mechanism.

6. For facility modifications, specify what testing is to be performed to assure that the systems involved operate in accordance with their design intent.

No modification to the actual rod drive mechanism will be made. This modification is simply to change th( )hysical location of the control rod drive mechnaism. The transient rod drive mechansim will operate in accordance with its design intent.

Facility Modification Worksheet 1

7. Specify associated information.

New drawings are: Attached _X_ (Photo is attached, drawing to be produced at a later date)

Not required ____

Does a drawing need to be sent to Logistics? Yes _X_ No ____

Are training materials effected? Yes ____ No _X_

Will any Logs have to be changed? Yes ____ No _X_

Are other procedures effected? Yes ____ No _x_

List of items affected:

None.

Drawings in Reactor file to be encluded or up dated.

8. Create an Action Sheet containing a list of associated work specified in item # 7, attach a copy, and submit another to the RFD.

Action Sheet: Submitted _X_ Not Required ____

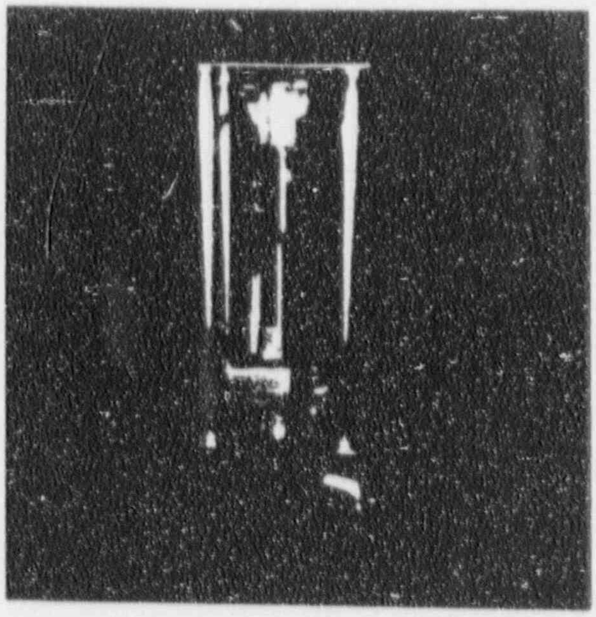
Reviewed and approved by RFD M L Moody Date 24 Oct 91

RRFSC Concurrence __________ Date 17 DEC 1991

**Safety Analysis of Transient Rod Drive Support Modification**

The transient rod drive will be raised approximately 60 inches above its current location. The purpose of this modification is to facilitate maintenance of the transient rod drive mechnanism. No modification of the function of the transient control rod drive will be made. All normal testing will be performed to insure that the systems will operate in accordance with its design intent.

The photographs below illustrate the modification made:



# Action Sheet

| Control Number | |
|---|---|
| Office Symbol: RSDR | Suspense |
| Date: 23 OCT 91 | |

**Subject**

Modification of Transient Control Rod Drive Support

**Action Required**
Concurrence and Approval

**Memorandum for Record.** (Describe briefly the requirements, background and action taken or recommended. Must be sufficiently detailed to identify the action without recourse to other sources.)

1. The transient control rod drive mechanism needs to be raised by about 60 inches above its current location to allow for ease of maintenance. LOGD needs to design and appropriate support structure to keep the rod drive mechanism securely in place above its current location.

2. Once built, we'll need to make a drawing of the finalized design for our as-built drawings. Autocad would probably be best.

3. No training materials should be effected by this modification, but we should brief the staff on why the change was made.

4. Capt Forsbacka to procure/make/develop Drawings to be filed in reactor system. Use autocad/ as available.

(Continue on plain bond)

| Office | Name | Phone | | Initials | Date |
|---|---|---|---|---|---|
| | **Coordinations** | | **Approvals** | | |
| RSDR | ~~Mark Moore~~ | ~~5-1290~~ | | [illegible] | 24 Oct 91 |
| RSDR | CAPT Forsbacka | 5-1250 | | | |
| | | | | | |
| | | | | | |
| | | | Dispatched (Dtg) | | |

Show Additional Coordination on Reverse Side or Continuation Sheet

Action Officer (Name, grade, phone and signature)

Matt Forsbacka, Capt USAF, 5-1221

DNA Form 547
4 Jan 83

Regraded When Separated from Classified Document

ATTACHMENT E-2

Attachment 1

REFERENCE: ADMINISTRATIVE PROCEDURE I, FACILITY MODIFICATIONS

ANALYSIS OF PROPOSED MODIFICATION

Modification Nomenclature: **REPLACEMENT OF WATER GAMMA ACTIVITY MONITORING SYSTEM**

Analysis: **SFC LAUGHERY** Date: **12 February 1991**

SECTION A

1. Document analysis to determine if a change to the Technical Specification is required. Include 10 CFR and/or Technical Specification references as applicable.

   **NO CHANGE TO THE TECHNICAL SPECIFICATION IS REQUIRED**

2. If your analysis determines that a Technical Specification change is required, go to SECTION B.

3. If a Technical Specification change is not required, document an analysis to determine if the proposed modification would constitute a change to the facility as described in the SAR. Include 10 CFR and/or SAR references as applicable.

   **NO CHANGE TO THE FACILITY AS DESCRIBED IN THE SAR IS REQUIRED. The proposed modification is the replacement of the vacuum tube operated monitoring system with a solid state system. This modification is necessary because replacement vacuum tubes are no longer available.**

4. Document an analysis to determine if the proposed modification would constitute a change in a procedure as described in the SAR. Include license and/or SAR references as applicable.

   **PROPOSED MODIFICATION DOES NOT CONSTITUTE A CHANGE IN A PROCEDURE AS DESCRIBED IN THE SAR**

5. Document an analysis to determine if the proposed modification would constitute a change in the tests or experiments as described in the SAR. Include license and/or SAR references as applicable.

   **PROPOSED MODIFICATION DOES NOT CONSTITUTE A CHANGE IN THE TESTS OR EXPERIMENTS AS DESCRIBED IN THE SAR**

6. If the proposed modification does not constitute a change to the facility, procedures, tests or experiments as

ATTACHMENT E-2

SECTION C

1. No 10 CFR 50.59 is required. The analysis in SECTIONS A.3, A.4, A.5 provide the bases for this determination that:

   a. no change in the Technical Specification is required.

   b. no change in the facility as described in the current SAR is proposed.

   c. no change in the procedures as described in the current SAR is proposed, and

   d. the proposed test or experiment:

      (1) coincides with those described in the current SAR;

      (2) is permitted by the Technical Specifications, and

      (3) has been previously reviewed and conducted.

2. This modification has been reviewed in accordance with ALARA principles. Comments (if necessary) are as follows

3. Reviewed and approved by Acting [illegible] Thomas Wright Date 2/12/91

4. Rev[illegible] approved by RFD M L Morey Date 2/14/91

5. RRFSC Concurrence [illegible signature] 20 Feb 91

ATTACHMENT E-3

Attachment 1

REFERENCE: ADMINISTRATIVE PROCEDURE 1. FACILITY MODIFICATIONS

ANALYSIS OF PROPOSED MODIFICATION

Modification Nomenclature: **REPLACEMENT OF WATER CONDUCTIVITY MONITORING SYSTEM**

Analysis: **SFC LAUGHERY** Date: **12 February 1991**

SECTION A

1. Document analysis to determine if a change to the Technical Specification is required. Include 10 CFR and/or Technical Specification references as applicable.

   **NO CHANGE TO THE :HNICAL SPECIFICATION IS REQUIRED**

2. If your analysis ·· ·ermines that a Technical Specification change is required, go SECTION B.

3. If a Technical Specification change is not required, document an analysis to determine if the proposed modification would constitute a change to the facility as described in the SAR. Include 10 CFR and/or SAR references as applicable.

   **NO CHANGE TO THE FACILITY AS DESCRIBED IN THE SAR IS REQUIRED. The proposed modification is the replacement of the present monitoring system with a new system by the same manufacture. This modification is necessary because replacement parts for the old system are no longer available.**

4. Document an analysis to determine if the proposed modification would constitute a change in a procedure as described in the SAR. Include license and/or SAR references as applicable.

   **PROPOSED MODIFICATION DOES NOT CONSTITUTE A CHANGE IN A PROCEDURE AS DESCRIBED IN THE SAR**

5. Document an analysis to determine if the proposed modification would constitute a change in the tests or experiments as described in the SAR. Include license and/or SAR references as applicable.

   **PROPOSED MODIFICATION DOES NOT CONSTITUTE A CHANGE IN THE TESTS OR EXPERIMENTS AS DESCRIBED IN THE SAR**

6. If the proposed modification does not constitute a change to the facility, procedures, tests or experiments as described in the SAR (Answer to SECTIONS A.3, A.4 and A.5 is "NO" in all cases). Go to SECTION C.

TTACHMENT E-3

SECTION C

1. No 10 CFR 50.59 is required. The analysis in SECTIONS A.3, A.4, A.5 provide the bases for this determination that:

   a. no change in the Technical Specification is required.

   b. no change in the facility as described in the current SAR is proposed.

   c. no change in the procedures as described in the current SAR is proposed, and

   d. the proposed test or experiment:

      (1) coincides with those described in the current SAR;

      (2) is permitted by the Technical Specifications, and

      (3) has been previously reviewed and conducted.

2. This modification has been reviewed in accordance with ALARA principles. Comments (if necessary) are as follows:

3. Reviewed and approved by Acting ROS ______________ Date 2/12/91

4. Reviewed and approved by RFD ______________ Date 2/14/91

5. RRFSC Concurrence ______________ Date 20Feb9

ATTACHMENT E-4

Facility Modification Worksheet 2

No 10 CFR 50.59 Analysis Required

Proposed Change CAM Damper Closure System Override box

Modification to: Procedure ____ Facility XXXX Experiment ____

Submitted by: George Date 6 Dec 91

1.Description of change:

Installation of small box in 3152 above air compressor regulator pressure guage that will allow operators to override the cam damper closure system to allow the dampers to open when the cams are alarming. System will only be operated upon the orders of the RFD.
Operation of the damper closure system is tested every morning so any problems will be detected before any daily operations.

THIS SYSTEM IS TEMPORARY AND WILL BE REMOVED AFTER SUCESSFUL PULSE TESTING OF THE FFCRS. USE OF THIS SYSTEM WILL BE ALLOWED ONLY BY DIRECT AUTHROIZATION BY THE RFD.

2. Verify that the proposed change does not involve a change to the Technical Specifications, the facility as described in the SAR, or procedures as described in the SAR, and does not produce an unresolved safety question as defined in 10 CFR 50.59(a)(2).

This system is temporary, the CAMs will still trip the dampers as intended.

3. If change involves a facility modification, attach a drawing if appropriate. If structural facility drawings need updating, forward a copy of changes necessary to Logistics. N/A

4. Determine what other procedures, logs, or training material may be affected and record below.

Staff briefing will be required.

5. List of associated drawings, procedures, logs, or other materials to be changed:

See attached drawing of system for reactor files.

6. Create an Action Sheet containing the list of associated work specified above, attach a copy, and submit it to the RFD.

Action Sheet: Submitted X Not Required _

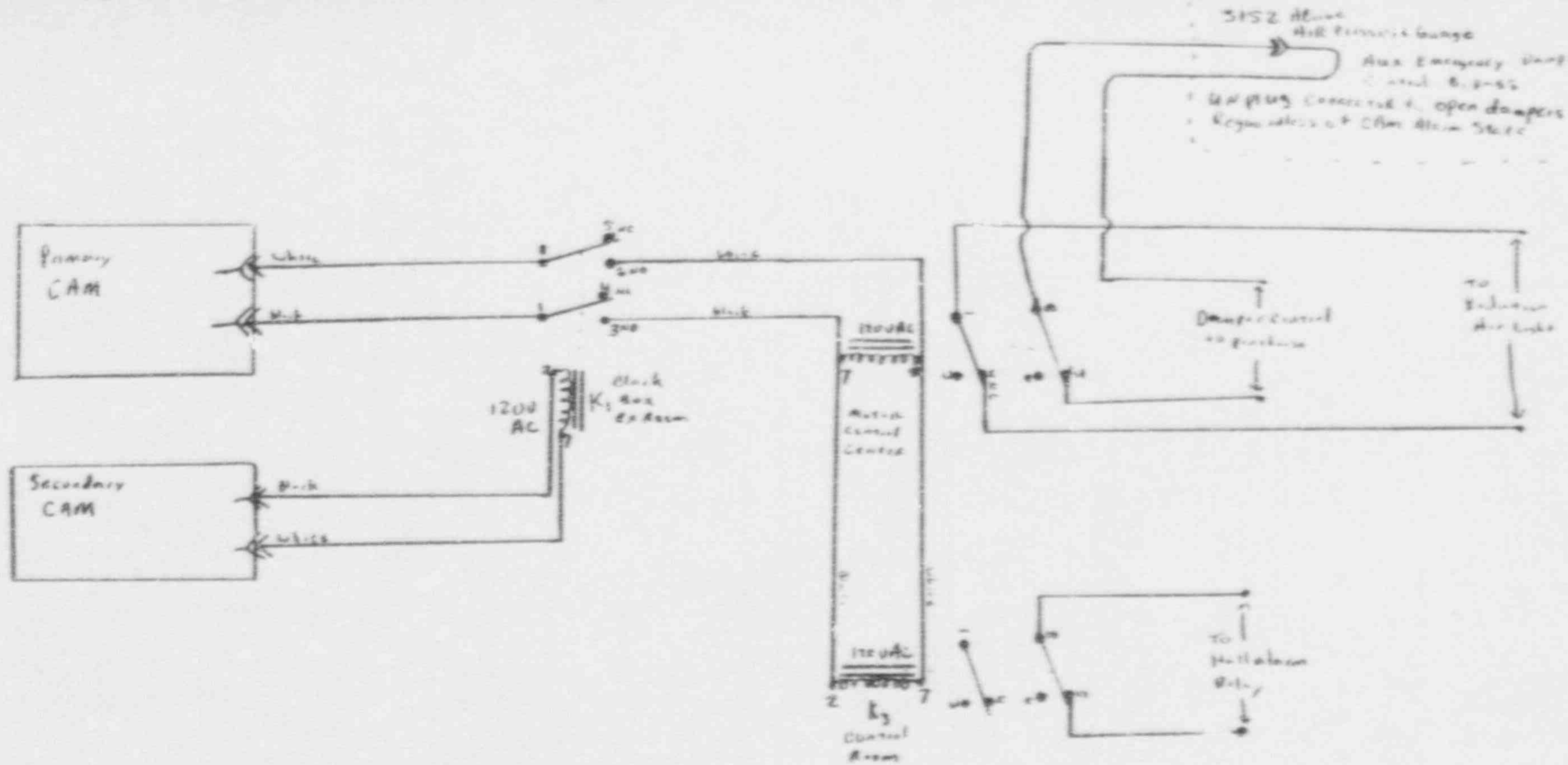
Reviewed and approved by RFD MLNaf Date 6 Dec 91

RRFSC Notified ____________ Date 17 DEC 1991



ATTACHMENT E-4

CAM ACTIVATED DAMPER CLOSURE SYSTEM




KS af 6 DEC 91

# TRAINING FOR DAMPER OVERRIDE BOX

GENERAL
The damper closure system located in the penthouse requires a closed loop to cause the dampers to close. When the loop is open the dampers can be opened and will remain open. A two conductor wire is run from the penthouse to a relay in the motor control center. This relay is operated by AC voltage from the primary CAM. If the primary CAM alarms, AC power will be turned off to the relay in the motor control center. If the backup CAM alarms it will break the AC power with a relay in the black box in the reactor room under R2.

Under normal operation power from the primary cam energizes the relay in the motor control center. This opens the loop to the penthouse damper closure system which will allow the dampers to be open.

DAMPER OVERRIDE:
The Cverride system consists of a two conductor wire with a plug on the end of each wire so that they can be plugged together. The wire breaks into the two conductor wire which comes from the penthouse and simply acts as an extension of that wire. One end of the new wire connects into the penthouse wire at the relay in the motor control center. The other end of the wire, the end with the connector, is located in 3152 on the wall in a labeled box above the air compressor.

Under normal operations the connector will always be connected. This will act like a normal wire and the system will operate properly. In the event of an emergency and if the RFD orders it, the connector can be disconnected. Opening this connector will open the circuit to the penthouse thus allowing the dampers to be opened.
**The connector should never be disconnected without direct orders from the RFD.**

# Action Sheet

| Control Number | |
|---|---|
| Office Symbol: RSDR | Suspense: N/A |
| Date: 6 DEC 91 | |

**Subject**

Temporary Installation of CAM Damper Override System

**Action Required**

Training of Staff in use of override system, removal of system when no longer requir

**Memorandum for Record.** (Describe briefly the requirements, background and action taken or recommended. Must be sufficiently detailed to identify the action without recourse to other sources.)

1. A CAM damper override system is proposed for the purpose of allowing the air dampers to be opened if necessary during the pulse testing of the FFCRs. THIS SYSTEM IS TEMPORARY, AND USE OF THIS SYSTEM MAY ONLY BE AUTHORIZED BY THE RFD.

2. Once the FFCRs are fully tested, this override system will be removed.

(Continue on plain bond)

**Coordinations** / **Approvals**

| Office | Name | Phone | | Initials | Date |
|---|---|---|---|---|---|
| RSDR | Mark Moore, RFD | 5-1290 | | | |
| | | | | | |
| | | | | | |
| | | | | | |
| | | | | | |
| | | | Dispatched (Dtg) | | |

Show Additional Coordination on Reverse Side or Continuation Sheet

**Action Officer** (Name, grade, phone and signature)

Matt Forsbacka, Capt USAF, 5-1290

DNA Form 547
4 Jan 83

Regraded When Separated from Classified Document

2.5.9

ATTACHMENT E-5

[illegible]

REFERENCE: ADMINISTRATIVE PROCEDURE 1, FACILITY MODIFICATIONS

ANALYSIS OF PROPOSED MODIFICATION

Modification Nomenclature <u>Auxiliary Timer on New Console</u>

Analysis <u>SFC Laughery</u> Date <u>20 December 1990</u>

SECTION A

1. Document analysis to determine if a change to the Technical Specification is required. Include 10 CFR and/or Technical Specification references as applicable.

   **NO CHANGE TO THE TECHNICAL SPECIFICATION IS REQUIRED**

2. If your analysis determines that a Technical Specification change is required, go to SECTION B.

3. If a Technical Specification change is not required, document an analysis to determine if the proposed modification would constitute a change to the facility as described in the SAR. Include 10 CFR and/or SAR references as applicable.

   **THERE IS NO CHANGE TO THE FACILITY AS DESCRIBED IN THE SAR.**

4. Document an analysis to determine if the proposed modification would constitute a change in a procedure as described in the SAR. Include license and/or SAR references as applicable.

   **THERE IS NO CHANGE IN ANY PROCEDURES AS DESCRIBED IN THE SAR.**

5. Document an analysis to determine if the proposed modification would constitute a change in the tests or experiments as described in the SAR. Include license and/or SAR references as applicable.

   **THERE ARE NO CHANGES IN THE TESTS OR EXPERIMENTS AS DESCRIBED IN THE SAR.**

6. If the proposed modification does not constitute a change to the facility, procedures, tests or experiments as described in the SAR (Answer to SECTIONS A.3, A.4 and A.5 is "NO" in all cases), Go to SECTION C.

TTACHMENT E-5

SECTION C

1. No 10 CFR 50.59 is required. The analysis in SECTIONS A.3, A.4, A.5 provide the bases for this determination that:

   a. no change in the Technical Specification is required.

   b. no change in the facility as described in the current SAR is proposed.

   c. no change in the procedures as described in the current SAR is proposed, and

   d. the proposed test or experiment

      (1) coincides with those described in the current SAR;

      (2) is permitted by the Technical Specifications.

      (3) has been previously reviewed and conducted.

2. This modification has been reviewed in accordance with ALARA principles. Comments (if necessary) are as follows:

   The new reactor console has a COUNT DOWN timer.
   The auxiliary timer is a COUNT UP timer which facilitates record keeping during experiments by displaying the actual time the reactor was at power.

3. Reviewed and approved by Acting ROS Thomas Wright Date 14 Feb 91

4. Reviewed and approved by RFD [signature] Date 14 Feb 91

5. RRFSC Concurrence [signature] Date 20 Feb 91

ATTACHMENT E-6

Facility Modification Worksheet 1

10 CFR 50.59 Analysis

Proposed Change: Upgrade pulse display capabilities and install variable pulse timer as specified in original contract with General Atomic.

Submitted by: Capt Forsbacka Date 13 NOV 91

1. Description of change:

Modification of both NPP-1000s for small pulse operation. This modification allows a gain change for high sensitivity pulse data aquisition. Addition of an Eagle dig timer and relay logic to scram the reactor after initiating a pulse.

2. Reason for change:

To fulfill original GA contract obligations.

3. Verify that the proposed change does not involve a change to the Technical Specifications or produce an unresolved safety issue as specified in 10 CFR 50.59(a)(2). Attach an analysis to show this.

Analysis attached? Yes Not Required

4. The proposed modification constitutes a changes in the facility or an operational procedure as described in the SAR. Describe which (check all that apply).

Procedure X Facility X Experiment ____



ATTACHMENT E-6

5. Specify what sections of the SAR are applicable. In general terms describe the necessary updates to the SAR. Note that this description need not contain the final SAR wording.

Section 4.14, Scram Logic Circuitry: Add description of pulse mode scram timer.

6. For facility modifications, specify what testing is to be performed to assure that the systems involved operate in accordance with their design intent.

Both modifications are for the purpose of ensuring that the console operates in accordance with its design intent.

Facility Modification Worksheet 1

7. Specify associated information.

New drawings are: Attached __X__

Not required ____

| | | |
|---|---|---|
| Does a drawing need to be sent to Logistics? | Yes ____ | No __X__ |
| Are training materials effected? | Yes __X__ | No ____ |
| Will any Logs have to be changed? | Yes ____ | No __X__ |
| Are other procedures effected? | Yes __x__ | No ____ |

List of items affected:

Training materials will have to reflect the ability to view pulses in two modes.

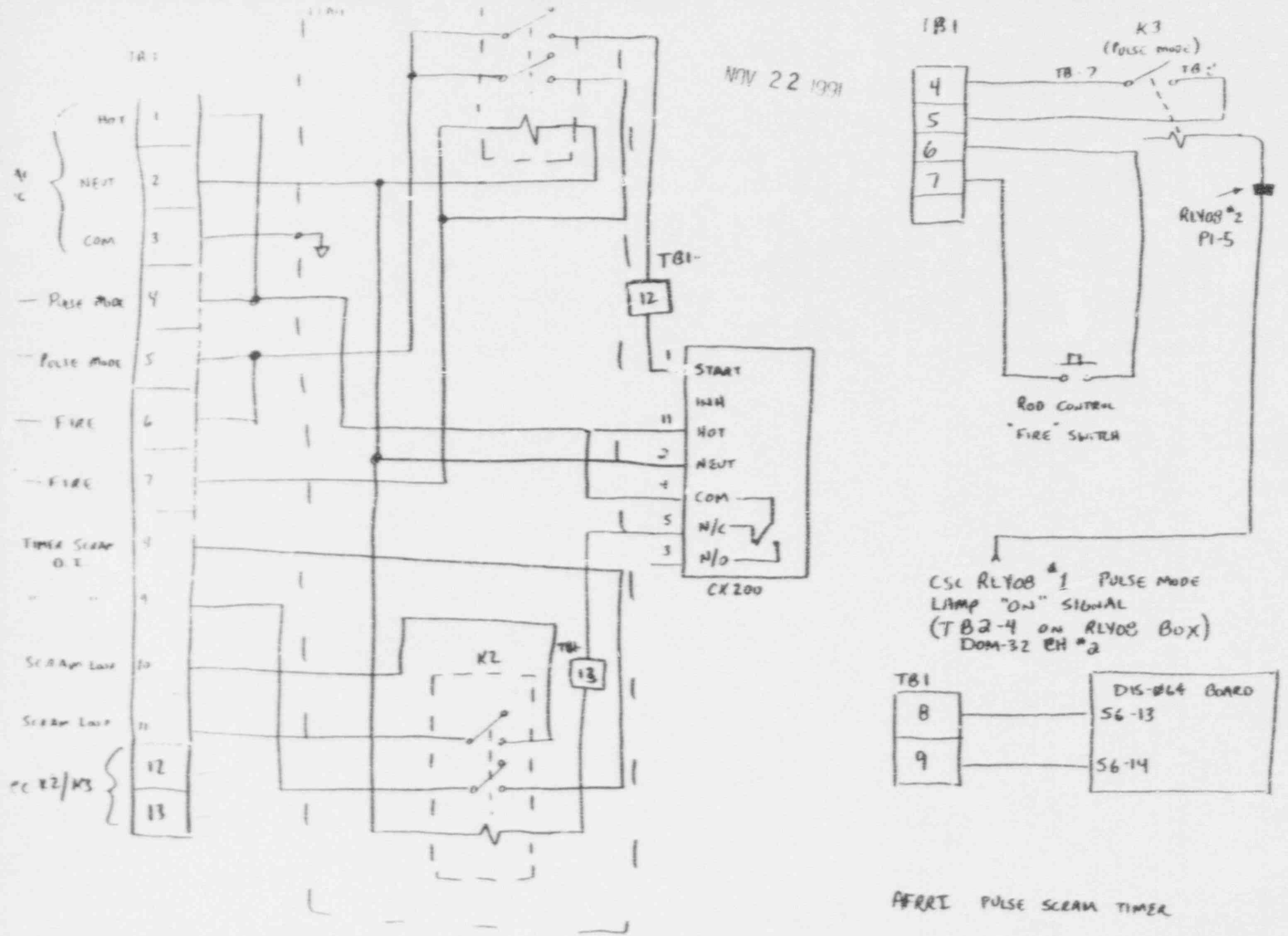
Procedure 8, TAB G1 and G2, will have to be modified to instruct the operator which mode of pulse display should be used.

8. Create an Action Sheet containing a list of associated work specified in item # 7, attach a copy, and submit another to the RFD.

Action Sheet: Submitted __X__ Not Required ____

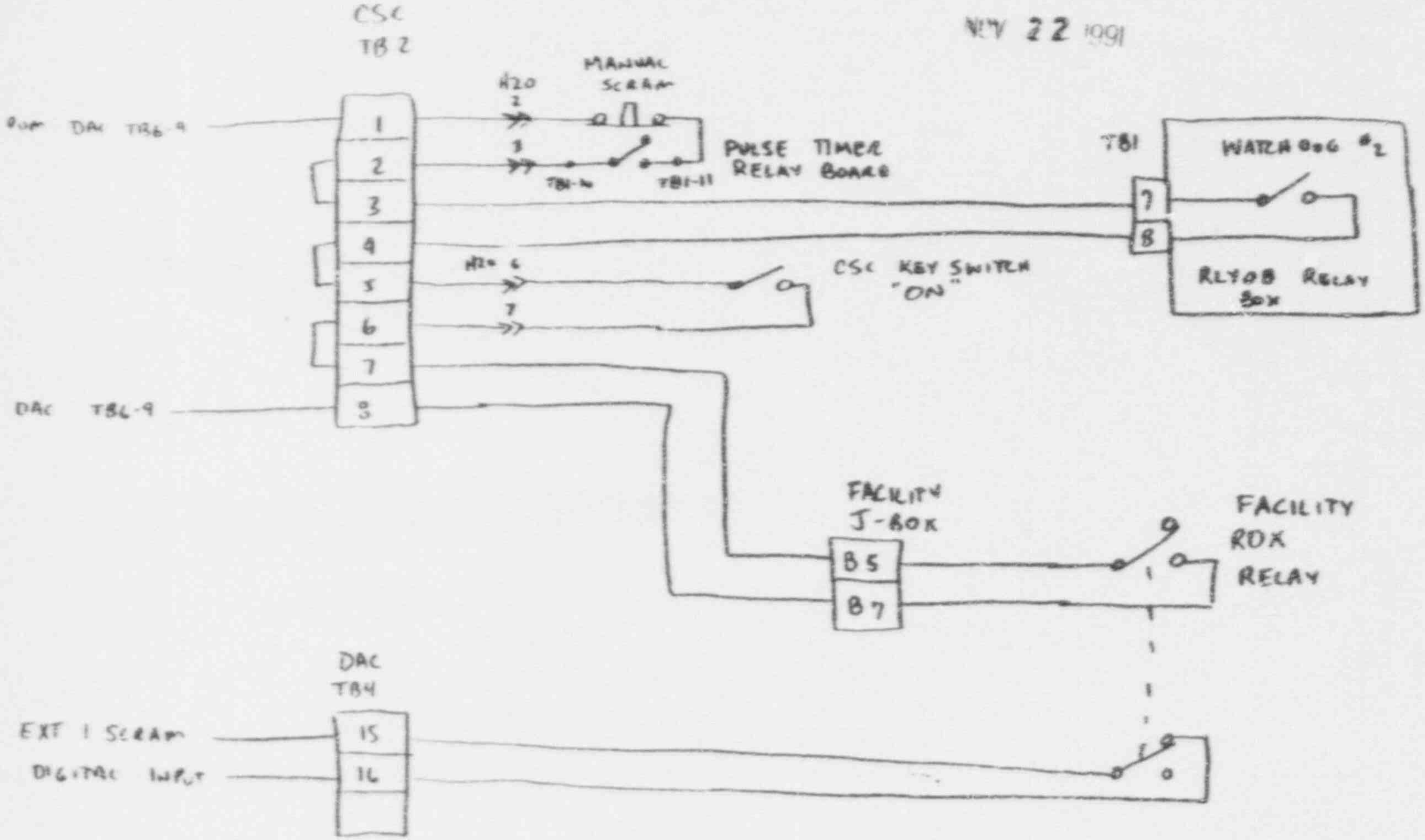
Reviewed and approved by RFD ________ Date 13/Dec/91

RRFSC Concurrence ________ Date 17 DEC 1991



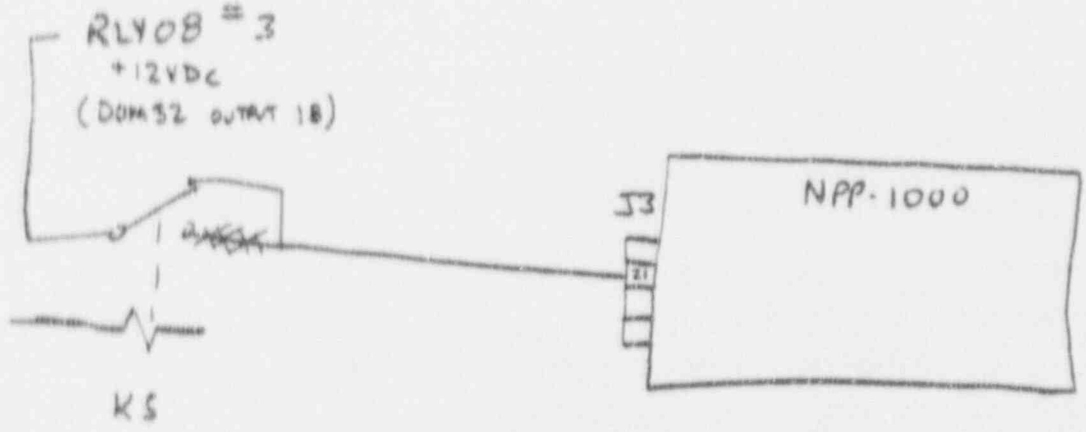

AFRRI PULSE SCRAM TIMER

NOV 22 1991




AFRRI SCRAM LOOP - HIGH SIDE

NPP Adjustment




NPP-1000 SWITCHES

| | |
|---|---|
| S5-1, 5 | CLOSED |
| S5-8 | CLOSED |
| S5-6, 7 | OPEN |
| S4-4, 3, 2 | OPEN |
| S4-6, 7, 8 | CLOSED |
| S4-1 | OPEN |
| S6-3 | CLOSED |
| S6-1, 2, 4 | OPEN |

TO ADJUST: - Low Sen. Gain (Normal S.S., Normal Pulse)
R-23 (0.50 mA F.S.)
- High Sen. Pulse Gain, R-27 (0.10 mA F.S.)

PRETEST CAL (110%) R-196

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

October 8, 1991

Docket No. 50-170

Colonel George W. Irving, III, BSC, USAF
Director
Armed Forces Radiobiology Research Institute
Bethesda, Maryland 20814-5415

Dear Colonel Irving:

SUBJECT: ISSUANCE OF AMENDMENT NO. 21 TO FACILITY OPERATING LICENSE NO. R-84 - ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE (AFRRI)

The Commission has issued the enclosed Amendment No. 21 to Facility Operating License No. R-84 for the AFRRI TRIGA Research Reactor. The amendment consists of changes to the Technical Specifications in response to your submittal of April 30, 1990, as supplemented on December 17, 1990, March 5, 1991, May 17, 1991, August 16, 1991, and September 10, 1991.

The amendment (1) corrects errors in typography and grammar, (2) increases the maximum licensed steady state reactor power to 1100 kilowatts, (3) authorizes installation of fuel follower control rods, (4) clarifies the transfer of Reactor Facility Director (RFD) responsibilities in the absence of the RFD, and (5) allows operational flexibility in performing surveillance testing of the ventilation system for the reactor facility.

Enclosure 2 is a copy of the related Safety Evaluation supporting Amendment No. 21.

Sincerely,

Richard F. Dudley, Jr. for

Alexander Adams, Jr., Project Manager
Non-Power Reactors, Decommissioning and
Environmental Project Directorate
Division of Advanced Reactors
and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 21
2. Safety Evaluation

cc w/enclosures:
See next page

# Action Sheet

| Control Number | |
|---|---|
| Office Symbol: RSDR | Suspense: N/A |
| Date: 13 NOV 91 | |

**Subject**

Installation of upgraded pulse display and variable pulse timer.

**Action Required**

Modify Procedure 8, TAB G1 and TAB G2.

**Memorandum for Record.** (Describe briefly the requirements, background and action taken or recommended. Must be sufficiently detailed to identify the action without recourse to other sources.)

1. The new pulse display will allow us to select high or low resolution, depending on the size of the pulse. The low resolution will be used for large pulses and the high resolution will be used for small pulses. We need to modify procedure 8, TAB G1 and TAB G2 to reflect this option.

(Continue on plain bond)

| Coordinations | | | Approvals | | |
|---|---|---|---|---|---|
| Office | Name | Phone | | Initials | Date |
| RSDR | Mark Moore, RFD | 5-1290 | | [illegible] | 13 Nov 91 |
| | | | | | |
| | | | | | |
| | | | | | |
| | | | Dispatched (Dtg) | | |

Show Additional Coordination on Reverse Side or Continuation Sheet

Action Officer (Name, grade, phone and signature)

Matt Forsbacka, Capt USAF, 5-1290

DNA Form 547
4 Jan 83

Regraded When Separated from Classified Document

# FAX TRANSMISSION

## GENERAL ATOMICS TRIGA REACTORS

**FROM:** Bill Hood **PAGE 1 OF 2**

**TO:** Harry Spence, AFRRI

**SUBJECT:** Work Performed Week of 11-18-91 **DATE:** 12-06-91

**OLD BUSINESS:**

a) Installed Pulse Mode Scram Timer. Added a Eagle digital timer and relay logic to scram the reactor a preset time after initiating a pulse. The timer is mounted behind the chart recorders in the CSC, and is interfaced to the CSC computer and scram loop through a pair of relays. The timer is only active in PULSE mode, and is started by firing the pulse rod.

There is no provision to disable the timer in pulse mode. The console software provides for a maximum scram delay of 15 seconds in pulse. If the operator wishes to scram the reactor prior to the 15 second software scram timeout, the desired time must be entered to the eagle timer prior to initiation of the pulse. Note that due to the software configuration the CSC will not post the Pulse Mode Scram Timer scram message until all pulse data acquisition is completed.

Rob and Mike were given a copy of the Pulse Mode Scram Timer and the updated CSC scram loop schematics.

b) Replaced the serial data cable between the CSC computer and the Tektronix high resolution monitor. The new cable is a two piece cable, which allows the disconnection of the high resolution monitor at the rear of the computer chassis. The AFRRI wire list was updated to reflect the wiring change (see Rob or Mike for a copy of the list).

**CURRENT ORDERS:**

a) Fuel Follower Control Rods. You should have all of the information on this subject already.

b) Small Pulse Data Acquisition.

HARDWARE: Modified both NPP-1000's for the small pulse option. This modification allows a gain change for high sensitivity pulse data acquisition. Also modified DOM-32 relay output #18 (in the DAC) to enable the gain change in pulse mode.

Rob and Mike have copies of the changes.

SOFTWARE: Installed new software to allow the acquisition of both normal and small pulses. The high sensitivity pulse data acquisition will acquire up to 1000 MW pulses (full scale).

# ATTACHMENT F

May 1991 Summary of Changes to
Administrative and Operational Procedures

## Summary of Changes to Administrative and Operational Procedures

### Introduction

In May, 1991 RSDR staff completely updated the Administrative and Operational Procedures to implement grammatical changes, a new format for the procedures, and corrections throughout including updating the core position references.

### Administrative Procedures

A1. Add reporting requirements to (2).

A2. Delete reference to "escorted access roster".

A3. Complete revision/simplification of 10 CFR 50.59 worksheets.

A4. Completely new procedure developed in conjunction with NRC recommendations.

### Operational Procedures

O. Change "initial block" to "signature block".

1. Add "acting ROS" to 2.f.

1A. Delete escorted access roster (2.b.).
Change "open" to "enter" (4.a.).
Delete material that duplicated HPP 3-1 (7.c.).
Update badge types for entry (2.b.).

1B. Change "element F28" to "desired element" (1.e. and 4.g.).
Delete reference to "upper end indicator" painted on cable (3.b.).

1C. Delete requirement place lead bricks on tube supports (1.a.(1)).
Delete references to former "limit switch" (1.a.(5)).
Delete reference to having prep area "sealed off" during operations (3.f.).

1D. No changes.

1E. Change "SRO" to "licensed operator" (1.).
Add "in the reactor room" to (3).

2. Delete statement "a record of operations will be kept for each trainee/operator" (2.c.).

3. Add "and console system manuals" to (1).

4. add "during operations" to (1.b.).

5. Add (3).

6. No changes.

7. Revise references to Shutdown Margin throughout.

8. Add (5.f.).

8A. Delete reference to "Mode 1, 1A, etc" (5.a.(5)).

8B. Delete measurement of pool level in 3161 (V.3.).
Change "conductivity" to "resistivity" (VI.5.).
Change "pool" to "inlet" (VI.17.g.).

8B1. Same changes as in 8B.

8C. No changes.

8D. Change 15 Watts to 5 Watts.
Delete reference to monthly summary sheet (4).

8E. Update title.
Change "servo mode" to "auto mode" throughout.
Change 800 Kw to 200 kW in (2).
Add (7).

8F1. Change "servo" to "auto" throughout.
Add general statement.
Add specific (1).
Add second sentence in (13).
Add (16).

8F2. Change "servo" to "auto" throughout.
Add general statement.
Add specific (1) and (12).
Add second sentence in (9).

8G1. Add maximum pulse sizes to (5).
Add different types of operations to (9).

8G2. Add pulse sizes to (5).
Add different types of operations to (12).

8H. Add (IV.b.). Remove refere 1.B to automatic scram reset, change "conductivity to "resistivity".

8I. Update (VI.2.) and (VI.4.).

9. Add "while the power rails are energized" to (1.f.).
Add "or mercury compounds in any form" to (2).

10. Moved from Procedure 8, Tab K to new Procedure 10. Add additional instructions to step 3 to better illustrate how to check the high voltage setting of the SGM

11. Formerly was attachment to Procedure VIII Tabs B and B1.