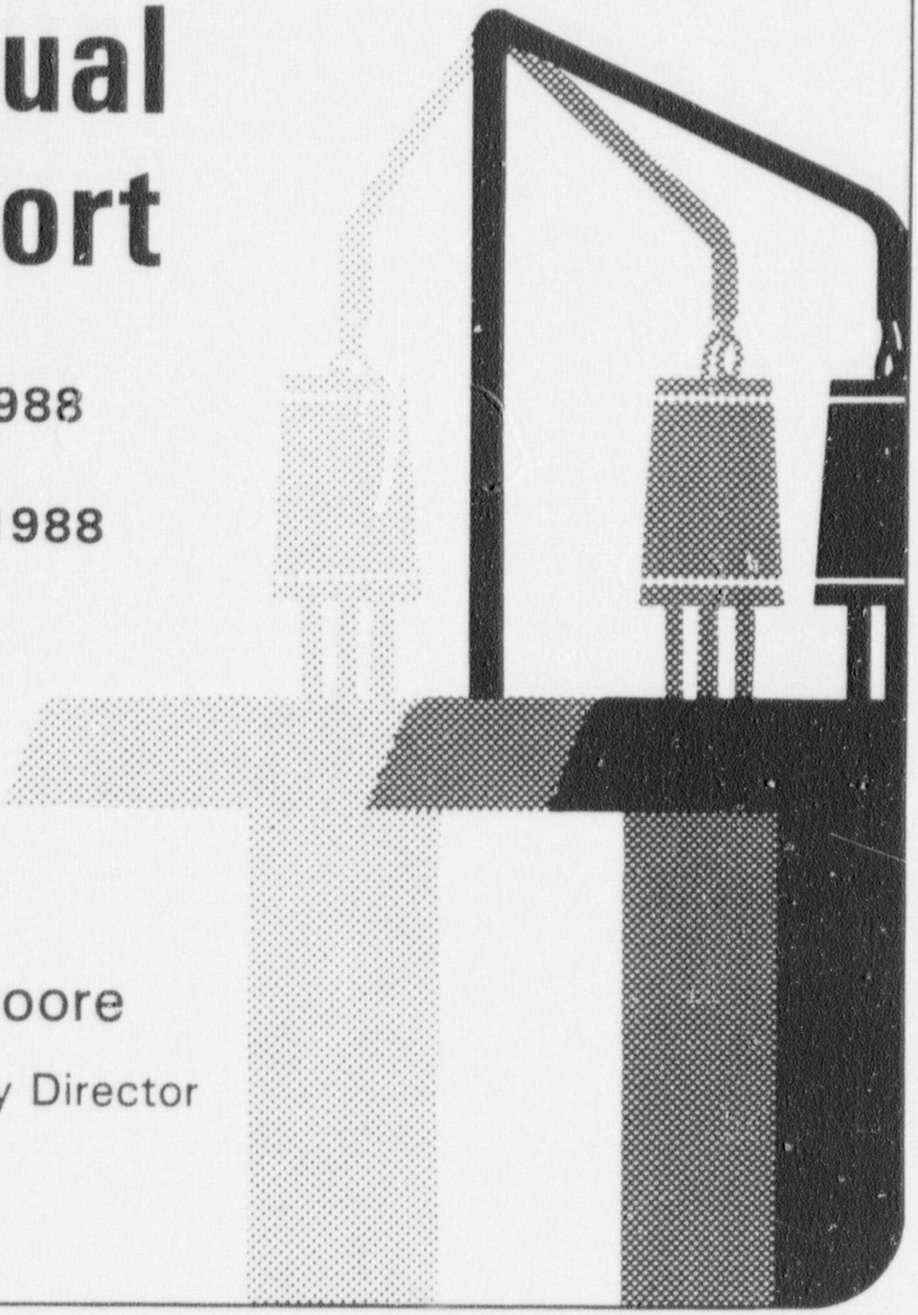
Reactor Facility

Annual Report

1 Jan 1988
to
31 Dec 1988



M. L. Moore

Reactor Facility Director

DEFENSE NUCLEAR AGENCY

ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE

BETHESDA, MARYLAND 20814-5145

ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE

TRIGA MARK-F REACTOR FACILITY

ANNUAL REPORT
1 JANUARY 1988 to 31 DECEMBER 1988

Prepared by:

Wendy Ting
MAJ James R. Felty
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Mark Moore
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Reactor Facility Director

Approved for Release:

George W. Irving, III
Colonel, USAF, BSC
Director, AFRRI

TABLE OF CONTENTS

Introduction

1988 began with expectations that the new facility instrumentation would be placed in an operational status. The new microprocessor based instrumentation and control system, developed by General Atomics, arrived at AFRRI and began undergoing pre-installation check out and testing. The year included some notes of frustration, however. Continuing software problems prevented the installation of the new console.

In October 1988 a concerned employee raised allegations of violations of procedures, technical specifications, parts of the Code of Federal Regulations, and other items not related to reactor operations. An internal investigation by the Defense Nuclear Agency Inspector General (DNAIG) found that the majority of the allegations were unsubstantiated and were based on differences of interpretation and understanding of rules and regulations between the concerned employee and management. However, based on a recommendation made by the DNAIG team, a thorough, exhaustive review of all reactor operational procedures was conducted, resulting in many changes that greatly benefitted the facility. Because of safety concerns, the employee was temporarily removed from the reactor area. Following a complaint to the Department of Labor (DoL) by this employee, the DoL found that management had discriminated against this employee. In a response to the DoL, management disagreed with the DoL determination, but complied with the DoL recommendation and the employee was returned to the reactor staff in December 1988. Management still had serious safety concerns and engaged an industrial psychologist to work with the reactor staff members to ease tensions and improve communication.

The Reactor facility was inspected by the U. S. Nuclear Regulatory Commission (USNRC) during 26-28 Oct 1988 and 7 Nov 1988 (USNRC Report 50-170/88-04). The results of this inspection coupled with the findings by the Reactor Facility Director (RFD) during a subsequent internal audit resulted in the RFD placing the reactor in a non-operational status on 16 Dec 1988. This non-operational status allowed time for an intense review and modification of existing procedures as well as time for a thorough procedural training exercise for all SROs.

AFRRI hosted a very successful National meeting of the TRIGA Owners and Users Association in early spring.

Changes made to the reactor facility necessitated the development of documents which described each modification and its applicability under 10 CFR part 50.59. These included the new microprocessor based instrumentation and control system and several other facility modifications. Each of these changes was supported by the required safety review process. These changes are elaborated upon in Section V of this report.

One Senior Reactor Operator (SRO) was added to the staff during CY 1988 along with two trainees. One SRO departed during the year. Requests from non-AFRRI investigators to use the reactor facility continued to supplement the substantial in-house experimental workload. These experimenters included representatives from the National Institutes of Health (NIH), Federal Bureau of Investigation (FBI), and the National Institute of Standards and Technology (NIST). Again this year the reactor staff was tasked to provide personnel to assist in conducting Department of the Army Inspector General (DAIG) inspections of the Fast Burst Reactor at White Sands Missile Range, New Mexico and the Fast Burst Reactor at Aberdeen Proving Grounds, Maryland.

A new charter for the Reactor and Radiation Facility Safety Committee (RRFSC) was approved. This charter, as shown in Attachment A to this report, details the functions and responsibilities of the RRFSC.

Two License Event Reports (LER) were submitted to the USNRC during the calender year. Positive steps were implemented to prevent further occurrences of the events detailed in these LERs in Attachment D to this report.

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 a 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

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

1. Current key AFRRI personnel (as of 31 Dec 1988) are as follows:

   Director - Col George W. Irving, III, BSC, USAF
   Scientific Director - Capt Richard Walker, MSC, USN
   (position vacant since 8 Aug 1988)
   Chairman, Radiation Sources Department - Mr. Mark Moore (SRO)
   Manager, Radiation Sources Program - MAJ Leonard A. Alt (SRO)
   Chairman, Safety and Health Department - Mr. Douglas Ashby
   (effective 4 Dec 1988)

2. Current key Reactor Operations Personnel:

   Reactor Facility Director - Mr. Mark Moore (SRO)
   Chief, Reactor Division - MAJ James R. Felty (SRO)
   Reactor Operations Supervisor - Capt Kenneth Hodgdon (SRO)
   (appointment terminated 24 Aug 1988)
   Reactor Operations Supervisor - MAJ James R. Felty (SRO)
   (appointment effective 24 Aug 1988)
   Asst Reactor Division Chief - Ms Wendy Ting (SRO, effective 23 Aug 1988)
   Training Coordinator - SFC Gary F. Talkington (SRO)
   Procurement Coordinator - SFC Philip Cartwright (SRO)
   Maintenance Coordinator - SFC Wayne Reed (SRO)
   Nuclear Engineer - Ms Angela Munno (SRO)
   Research Physicist - Dr. Jen Shu Hsieh

3. Other personnel:

   Senior Reactor Operator - MAJ Leonard A. Alt
   Senior Reactor Operator - SFC Stephen Holmes
   Senior Reactor Operator Candidates:
   CPT Philip Mattson
   Mr. Robert George (effective 26 Sep 1988)
   Mr. Boris Stallings (effective 22 Dec 1988)

4. Departures during CY 1988:

   Capt Kenneth Hodgdon (SRO License terminated 1 Sep 88)

5. There were changes to the RRFSC during the 1988 calendar year. Dr. Naresh Chawla was replaced (26 Apr 1988) by Mr. Thomas O'Brien, as Acting Chairman of the Safety and Health Department (SHD), who later was replaced (5 Dec 1988) by Mr. Douglas Ashby when he assumed the chairmanship of the Safety and Health Department (SHD) during the calendar year. Dr. Frank Munno resigned (5 Oct 1988) from the committee and was replaced by Dr. Marcus Voth, Pennsylvania State University. Mr. John Misner, ORI Incorporated, was appointed (18 Dec 1987) as a Special Member to study the software for the new digital reactor console.

The new charter for the RRFSC is enclosed as Attachment A to this report.

The 1988 RRFSC consisted of the following membership to satisfy the Reactor Technical Specifications:

Chairman - CAPT Richard I. Walker (former Deputy Director, AFRRI)

Regular Members

Mr. Douglas Ashby (Chairman, Safety and Health Dept., AFRRI)

Mr. Mark Moore (Chairman, Radiation Sources Dept. and Reactor Facility Director, AFRRI)

Dr. Marcus Voth (Director, Pennsylvania State University Breazeale Reactor and Professor of Nuclear Engineering, Pennsylvania State University) (appointment effective 5 Oct 1988)

Mr. Jathan W. Stone (Head, Safety Directorate, Naval Research Labs)

MAJ David P. Alberth (Radiation Safety Officer, Uniformed Services University of the Health Sciences)

Special Members

CDR Gary H. Zeman (Chairman, MRA, AFRRI)

Mr. John Misner (ORI Incorporated)

Observer

Mr. John Menke (EPA, Montgomery County, MD)

Meetings of the RRFSC were held on:

26 Apr 1988
26 May 1988
19 Oct 1988
15 Dec 1988

Section I

Changes to the facility design, performance characteristics, operating procedures, and results of surveillance testing.

A. DESIGN CHANGES

A number of facility design changes were completed during CY 1988. These changes are discussed in more detail in Section V and Attachment C to this report.

B. PERFORMANCE CHARACTERISTICS

There were no changes in the performance characteristics during the calender year.

C. OPERATIONAL PROCEDURES

During the year, numerous changes were made to the Operational Procedures. A complete set of the revised operational procedures as of 15 Dec 1988 is at Attachment B to this report. Through the brief summary of changes listed below, the 1988 operational procedural evolutions can be traced.

<u>Change 1</u>

Procedure VIII - Tab B: Daily Operational Start-up Checklist
Procedure VIII - Tab B1: Daily Safety Checklist
(RRFSC Meeting - 26 April 1988)

Change:

1. Remove Item I.7 - "Gas Stack Monitor(SGM) & Cooling Air Blower Off". The old SGM was replaced by the new SGM and this air blower was no longer being used.
2. Reword Item VI.7.c - ""Gas Stack Monitor High Alarm set to .."
   Changed from "1.5 E3" to "800 MPC Ar-41"
   Based on calibration factors for the SGM system, the instrument setpoint changed for each calibration. However, the 800 MPC Ar-41 requirement does not change. This change provides procedural consistency.

<u>Change 2</u>

Procedure VIII - Tab I: Daily Operational Shutdown Checklist
(RRFSC Meeting - 26 April 1988)

Change:

Remove Item VI.10 - "Coffee Pot Off"
The coffee pot is no longer in the reactor area.

Change 3
Procedure VI - Emergency Procedures
(RRFSC Meeting - 26 April 1988)

Change:
Reword Item 1.c
"EAS Commander" to ERT Commander"
Changed for consistency with the new Emergency Plan.

Change 4
Procedure VIII - Tab F: Square Wave Operation (Mode II)
(RRFSC Meeting - 26 April 1988)

Change:
Section 2.b has been elaborated upon to elucidate the procedure required to perform a Square Wave Operation when the TRANS rod is needed to achieve initial criticality.

Change 5
Procedure I - Tab A: Reactor Exposure Room Entry Procedure
(RRFSC Meeting - 26 April 1988)

Changes:
1. Item 1.c.(2) changed to add the requirement for wrist dosimetry to enter the exposure rooms.
2. Change Item 5.d to delete the phrase "and re-secure yellow area, if necessary". A yellow area painted on the floor was no longer being used as a restricted area because the entire Prep Area was now secured.

Change 6
Procedure VIII - Tab G: Pulse Operation (Mode III)
(RRFSC Meeting - 19 Oct 1988)

Change:
Item 12 Removed: "Select proper pulse detector according to table below". This was a redundant line.

Change 7
Procedure VI - Emergency Procedures
(RRFSC Meeting - 19 Oct 1988)

Change:
Item 2.b reworded to state "Secure any exposure facility in use so that personnel access to that facility is not possible". This allowed the reactor staff to adequately secure the facility and evacuate the building more rapidly.

Change 8
Procedure 0 - Procedure Changes
(RRFSC Meeting - 15 Dec 1988)

Change:
Paragraph 5 added "If the entire book of procedures is reviewed, a single signature block on a title page will substitute for individual review".

Change 9
Procedure I - Conduct of Experiments
(RRFSC Meeting - 15 Dec 1988)

Changes:
1. Added to Item 1.c the following phrase "and the core position of the experiment facility to be utilized". This codified in the procedure a requirement that was already being done.
2. Changed Item 1.d from "or his designee" to "acting RFD or ROS". This change was made for clarification.
3. Changed department title in Item 1.e to "Military Requirements & Applications, Operational Dosimetry Division (MRAD)" This was done to correct a misnomer.

Change 10

Procedure I - Tab A - Reactor Exposure Room Entry Procedure
(RRFSC Meeting - 15 Dec 1988)

Changes:

1. Deleted from first line of Item 2.b the phrase "and Head, Safety and Health Department or his representative". The RFD alone is responsible for granting access to these areas.
2. Changed department title in Item 2.b from "Head SHD" to "Chairman, Safety and Health Department (SHD)". This was done to correct a misnomer.
3. The remaining correction to Item 2.b clarified on Prep Area entry status, required personnel listings, and special entry conditions.
4. Item 2.c.(2) was clarified to mean "AFRRI TLD Whole Body badge".
5. Item 2.c.(5) was clarified to include "data obtained" as part of the information available before ER entry.
6. Item 3.b was modified to address exposure room openings in a more general manner, but without decreasing safety concerns.
7. Additional entry survey areas were added to Item 3.c.
8. Item 3.c was clarified to address accessible areas and extended stay time for personnel entering the exposure rooms.
9. Item 5.b was modified to address the case of a non-monitored opening.
10. Item 6.e was reworded to make a technically correct statement.
11. Item 7.c was entirely rewritten to more adequately detail the procedure.
12. Item 8.a was expanded to include conditions when the warning horn in the exposure rooms is disconnected.

Change 11
Procedure I - Tab B: Core Experiment Tube (CET)
(RRFSC Meeting - 15 Dec 1988)

Changes:

1. Item 1.l and Item 4.i were modified to ensure that the reactor core pegboard is kept current.
2. Item 2.e and Item 4.c were modified to ensure that the reactor operator is aware of reactivity changes.
3. Item 3.a clarifies the personnel dosimetry requirements during retrieval of samples from the core.
4. The procedure for sample withdrawal from the CET with respect to radiation levels was clarified in Item 3.i.

Change 12
Procedure I - Tab E: In-Pool/In-Core Experiments
(RRFSC Meeting - 15 Dec 1988)

Change:
The following line was added "Ensure that a member of the reactor staff and a SHD representative are present during the removal of samples from in-pool and in-core locations". This insured Health Physics monitoring of sample removal.

Change 13
Procedure V - Physical Security
(RRFSC Meeting - 15 Dec 1988)

Change:
The requirement to lock the reactor room during prolonged absences was added.

Change 14
Procedure VI - Emergency Procedures
(RRFSC Meeting - 15 Dec 1988)

Change:
Item 2.b reworded to state "Secure any exposure facility which are in use so that personnel access to that facility is not possible". This allowed the reactor staff to adequately secure the facility and evacuate the building more rapidly.

Change 15
Procedure VIII - Reactor Operations
(RRFSC Meeting - 15 Dec 1988)

Changes:

1. Item 2 now provides the rationale for performing a Daily Safety Checklist.
2. Item 3 requires that the SRO-on-call be annotated at the top each page in the operations logbook.
3. Item 7 requires that Daily Operational Shutdown Checklist be performed at the end of each day in which a Daily Operational Startup Checklist or a Daily Safety Checklist has been performed.

Change 16
Procedure VIII - Tab A: Logbook Entry Checklist

Change:
The following line was added to Item 4.a: "The operator in charge will be designated in the logbook whenever multiple operators are signed on the console".

Change 17
Procedure VIII - Tab I: Daily Operational Shutdown Checklist

Change:
Items VI.7 and VI.9 were reworded to be grammically similar to other items in the same section. No performance standards were changed.

D. SURVEILLANCE TESTING
All surveillance items were accomplished on time. Malfunctions discovered during operations are discussed in Section IV.

Section II

Energy generated by current reactor core

MONTH	kWHr
JAN	2756.4
FEB	6610.1
MAR	2294.6
APR	1665.7
MAY	2362.7
JUN	5620.2
JUL	3030.2
AUG	2093.9
SEP	2112.4
OCT	1433.5
NOV	1736.9
DEC	255.1
Total	31971.7

Total energy generated this year	31971.7 kWHr
Total energy on this core	629716.2 kWHr
Total pulses this year > $2.00	211
Total pulses on this core > $2.00	4093

Section III
Unscheduled shutdowns

There were no unscheduled shutdowns during this reporting period.

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 unit to its proper operational status.

7 Jan 88

Problem: While securing the Core Experiment Tube into storage position S-6, the bulk water temperature connecting wire was knocked loose.

Solution: The wire was repaired. The system was tested and found to be operational.

13-14 Jan 88

Problem: The meter of the NV/NVT circuit was indicting a false reading when the left-hand console drawer was moved (opened/closed).

Solution: A connector to the NV/NVT circuit board was replaced. The NV/NVT system was calibrated/tested and found to be operational.

1-2 Feb 88

Problem: The Square Wave mode of operation was found to be non-operational during testing.

Solution: Investigation showed that a blown fuse was the problem. The fuse was replaced. The system was tested and found to be operational.

4-5 Feb 88

Problem: Air was slowly leaking from the primary compressor that supplied compressed air to the Transient rod air system.

Solution: A loose bearing was found to be the cause of the leaking air. While this bearing was replaced, the air supply was temporarily switched to the back-up compressor system. After the bearing was replaced and tested, the air supply was returned to the primary compressor. The air system was tested for leakage and was found to be functioning properly.

6 May 88

Problem: During a routine pulsing operation, the console indicated that the Transient rod did not SCRAM after the pulse. The console also indicated that the remaining rods had SCRAMmed, leaving the reactor in a subcritical condition.

Solution: A preliminary investigation showed that the Transient rod did SCRAM following the pulse operation. The SCRAM function was fully operational. However, further investigation showed that the wiper arm of the rod down microswitch was broken. The microswitch was replaced. The Transient rod system was tested and found to be operational.

2 Aug 88

Problem: The low level alarm light on the Stack Gas Monitor became illuminated. Although no operations were currently being conducted, further operations were administratively prohibited.

Solution: The circuit boards were removed and re-seated. The system was reinitialized, tested, and found to be operational.

10 Aug 88

Problem: During the Daily Operational Start-up, Remote Area Monitor E-3 was found to be non-operable. Proper exposure levels were indicated but the monitor failed in the alarm test position.

Solution: RAM E-3 was replaced by another calibrated RAM. Prior to any operations or openings of Exposure Room 1, the system was tested and found to be operational.

29-30 Nov 88	Problem: During monthly maintenance checks, the air dampers in the ventilation system failed to function normally, i.e., the dampers did not open. Solution: Upon investigation, a solenoid that supplied air to open the dampers was not operational. Air pressure to the solenoid was checked. The air lines to the dampers were bled and the damper mechanisms were lubricated. Upon the reapplication of air, the system was tested and found to be operational.
1 Dec 88 and 5 Dec 88	Problem: The Stack Gas Monitor (SGM) was turned off for replacement of the CPU board with a newer version supplied by the manufacturer. The CPU board was replaced. A QA test was performed and the system was found to be operational. However, the printer printed constantly without stopping. Solution: Notified the manufacturing representative in order to fix the printing problem. They stated that this was a minor software problem which in no way affected the monitoring capability or operability of the SGM. The programmable ROM chips were replaced. The printer was tested and functioned properly. The SGM was reinitialized, tested and found to be operational.

Section V

Facility change: ..anges to procedures, and new experiments

A. The 10 CFR 50.59 safety reviews of; the new reactor instrumentation and control system, the warning lights in the control room from the Primary Continuous Air Monitor and the Stack Gas Monitor system and the Cerenkov detector, and the digital voltmeter are included as in Attachment C to this report.

B. A safety review was also performed concerning the relocation of the Equipment Room 3152 roof hatch, as shown in Attachment C to this report. Movement of the roof hatch served to improve the drainage on the reactor roof. In addition, the main door entrance to the reactor facility is being moved outward so that the door to Equipment Room 3152 falls inside the main door entrance to the reactor facility. Also, the ceiling above the relocated doorway is lined with wire mesh to further enhance the reactor physical security. The work on the relocation of the main door entrance to the reactor facility is still in progress and is expected to be completed by the end of 1989. The safety reviews performed on these two facility modifications showed that no unreviewed safety question existed.

C. A complete set of revised Operating Procedures is included as Attachment B to this report.

D. The new experiment performed during CY 1988 is covered in Attachment E.

Section VI

Summary of safety evaluation changes not submitted to NRC pursuant to 10 CFR 50.59

Attachment C satisfies the requirements of this section. Each modification is described and the basis for the conclusion that each change involves no unreviewed safety question, and that there are no changes to the Technical Specifications, has been provided.

The additional modification concerning the movement of the Equipment Room 3152 roof hatch, addressed in Section V, is not included in the SAR nor the Technical Specifications. Neither is it required for the safe operation of the reactor. However an analysis was performed and shows that no unreviewed safety questions exists.

License Event Reports are included as Attachment D to this report.

Section VII
Summary of radioactive effluent released

A. Liquid Waste - The reactor has produced no liquid waste during CY 1988.

B. Gaseous Waste - There were no particulate discharges in CY 1988. The total Argon-41 discharges in CY 1988 was 9.145 Ci.

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

Section VIII
Environmental radiological surveys

A. The environmental sampling of soil, water, and plant growth reported radionuclide levels that were not demonstrable 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 1988:

1. The average background of about 20 Thermoluminescent dosimeters (TLD) located within a 15 mile radius from the AFRRI site was determined to be 99.49 +/- 3.80 milliREM.

2. The average reading of approximately 30 environmental stations located on the AFRRI site from background was determined to be (-5.42 +/- 2.05) milliREM.

3. The single highest environmental station reading was (15.72 +/- 19.31) milliREM above background.

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 Section VII) in all areas outside the normal restricted-access areas.

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

Section IX

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

CHARTER FOR THE
ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE
REACTOR AND RADIATION SAFETY COMMITTEE

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

AFRRI/DIR 3 October 1988

SUBJECT: RRFSC Charter

TO: RRFSC Committee

1. Reference: Ammendent Charter

2. This Charter is now in affect until further notification.

GEORGE W. IRVING, III
Colonel, USAF BSC
Director

APPROVED by Col Irvi & passed out at 190 88 RRFSC Meet JRF

CHARTER FOR THE ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE REACTOR AND RADIATION FACILITY SAFETY COMMITTEE

I. INTRODUCTION:

This Charter governs the Armed Forces Radiobiology Research Institute's (AFRRI's) Reactor and Radiation Facility Safety Committee (RRFSC). The charter codifies the requirements of the AFRRI TRIGA Reactor Technical Specifications, complies with descriptions in the Safety Analysis Report for AFRRI TRIGA Mark-F Reactor, and adheres to the guidelines contained in American National Standard Institute (ANSI)/ American Nuclear Society (ANS) Standard 15.1-1982. In the execution of the duties and functions prescribed by Department of Defense Directive number 5105.33 (Subject: Armed Forces Radiobiology Research Institute, dated 25 NOV 87), the AFRRI Director is the proponent for this charter and is the sole authority for changes to or deviations from this document.

II. PURPOSE and AUTHORITY:

The RRFSC is directly responsible to the AFRRI Director. The committee reviews items that could affect the radiological health and safety aspects of facility operations and makes recommendations to the AFRRI Director concerning the following sources at the Institute:

TRIGA Reactor
LINEAR Accelerator
Cobalt Facility
Theratron Facility
X-Ray Facility
Other radiation sources as designated by the AFRRI Director.

The committee's review oversight includes: The physical facilities, the planned operations, and the qualifications of supervisory and operating personnel which relate to the safety of the Institute, its staff, the public, and the environment. The RRFSC discharges its principal responsibility of broad oversight of the Institute's radiation sources health and safety issues through the review of studies and reports prepared by the staff of the Radiation Sources Department and Safety and Health Department, and by commissioning periodic external audits of key operations.

Additionally, the RRFSC advises the Reactor Facility Director, the Radiation Sources Department Chairman, and the Safety and Health Department Chairman in those functional areas specified in section V of this document.

III. COMPOSITION OF COMMITTEE, QUALIFICATIONS AND TERMS OF SERVICE:

A. Committee Chairman, as appointed from the AFRRI Directorate by the AFRRI Director. Term as appointed by AFRRI Director.

B. Reactor Facility Director. Permanent with position.

C. Radiation Sources Department Chairman (if different from Reactor Facility Director). Permanent with position.

D. Safety and Health Department Chairman. Permanent with position.

E. One to three non-AFRRI members, appointed by the AFRRI Director, who are knowledgeable in fields related to reactor safety. At least one shall be a Reactor Operations Specialist, or a Health Physics Specialist. Annual term, renewable by AFRRI Director's appointment.

F. Special RRFSC Members (Temporary Members):

1. Other knowledgeable persons to serve as alternates for those in paragraph III.D., as appointed by the AFRRI Director. Temporary, for duration specified by AFRRI Director.

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

3. Non-voting members as invited by the RRFSC Chairman. Temporary, as indicated by RRFSC Chairman.

G. 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.

IV. MEETINGS and RULES:

A. Special Members (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.

B. Meeting Frequency

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

C. 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.

D. 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.

E. Minutes

Minutes of the previous meeting shall be available to regular members at least 1 week before a regular scheduled meeting.

F. Subcommittee

An RRFSC Subcommittee will consist of the regular RRFSC Chairman (or his designated substitute), one other permanent member, and selected special members as designated by the regular RRFSC Chairman. No more than fifty percent of the voting attendees may be special (temporary) members or alternate members.

V. FUNCTIONS:

A. REVIEW OF REACTOR OPERATIONS

The RRFSC shall review the items specified at sub-paragraphs 1 through 10 below, and will concur or non-concur with the findings of the Reactor Facility Director that the items present no hazard to the public health and safety. The review will normally occur at the first meeting following the implementation of actions for which the Reactor Facility Director has determined that no unreviewed safety questions exist, and prior to implementation of actions for which the Reactor Facility Director has determined that an unreviewed safety question exists.

A written report or minutes of the findings and recommendations will be submitted to the AFRRI Director in a timely manner after the review has been completed.

1. Safety evaluations for (1) changes to procedures, equipment, or systems and (2) tests or experiments conducted within NRC approval under provisions of Section 50.59 of 10 CFR Part 50, to verify that such actions did not constitute an unreviewed safety question.

2. Changes to procedures, equipment, or systems that change the original intent or use, and are non-conservative, or those that involve an unreviewed safety question as defined in Section 50.59 of 10 CFR Part 50.

3. Additions and modification to, and testing procedures for, SAR stated systems including the ventilation system, the core and its associated support structure, the pool, coolant system, the rod drive mechanism, or the reactor safety system; unless the additions and modifications are made and tested to the specifications to which the systems were originally designed and fabricated.

4. New experiments of a type for which previous authorization has not been granted, for radiological safety, prior to issuance of a reactor authorization.

5. Proposed tests or experiments that are significantly different from previously approved tests or experiments, or those that might involve

an unreviewed safety question as defined in Section 50.59 of 10 CFR Part 50.

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

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

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

9. Events that have been reported to the NRC.

10. Audit reports of the reactor facility operations.

B. AUDIT OF REACTOR OPERATIONS

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

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

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

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

4. Facility emergency plan and implementing procedures.

5. Facility security plan and implementing procedures.

6. Any other area of Facility operations considered appropriate by the RRFSC or the AFRRI Director.

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

C. REVIEW OF OTHER RADIATION FACILITIES OPERATIONS:

The RRFSC shall review the items specified at sub-paragraphs 1 through 7 below and will concur or non-concur with the findings of the Radiation Sources Department Chairman that the items present no hazard to the public health and safety. The review will normally occur at the first meeting following the implementation of actions for which the Radiation Sources Department Chairman has determined that no unreviewed safety questions exist, and prior to implementation of actions for which the Radiation Sources Department Chairman has determined that an unreviewed safety question exists.

A written report or minutes of the findings and recommendations will be submitted to the AFRRI Director in a timely manner after the review has been completed.

1. Changes to routine authorized irradiations or procedures with respect to personnel safety for the linear accelerator, cobalt-60 facility, theratron, and other non-reactor major radiation sources.

2. Qualifications of new source operators, for compliance with appropriate regulations and guidelines governing training for operations of a particular source.

3. Proposed amendments to any USNRC Licenses governing the use of the other sources.

4. Violations of application statutes, codes, regulations, orders, license requirements, or of internal procedures or instructions having nuclear safety significance.

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

6. Events that have been reported to the NRC.

7. Audit reports of facility operations.

VI. REFERENCES:

a. Technical Specifications for the AFRRI Reactor Facility, Docket 50-170, License R-84, June 1984.

b. Safety Analysis Report for AFRRI TRIGA Mark-F Reactor, June 1987.

c. American National Standards Institute/American Nuclear Society Standard 15.1-1982, The Development of Technical Specifications for Research Reactors, September 1982.

ATTACHMENT B

CURRENT
REACTOR
OPERATING PROCEDURES

REACTOR OPERATING PROCEDURES

AFRRI TRIGA MARK F REACTOR

APPROVED BY THE
REACTOR FACILITY DIRECTOR:

ORIGINAL SIGNED

______________________ __________
MARK MOORE Date

Revised: 15 Dec 1988

REACTOR OPERATING PROCEDURES

INDEX

Revised: 15 Dec 1988

PROCEDURE 0 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 and subsequently 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 initial block for all operators and reactor staff members. When the initial block is completed, the procedure will be placed in the Reactor Operation Binder and kept available for operator review.

5. If the entire book of procedures is reviewed, a single signature block on a title page will substitute for individual review.

Revised: 15 Dec 1988

PROCEDURE I 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 (CCTV's) 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. The Reactor Staff will conduct a thorough inspection of all experiments to determine that no unauthorized materials, items or substances, or equipment are irradiated.
4. ALARA will be practiced during all experiments.

Specific:

1. Experiment Review (Processing of Reactor Use Request (RUR)):
   a. Check RUR for completeness (Section I should be filled out).
   b. Check experiment protocol against reactor authorizations. Assign reactor authorization number.
   c. Fill-in Section II of RUR with special instructions, as appropriate. Assign 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).
   d. Have the Reactor Facility Director (RFD), acting RFD, or Reactor Operations Supervisor (ROS) review and sign the form.
   e. Forward the RUR to the Military Requirements & Applications Department, Operational Dosimetry Division (MRAD) and the Safety & Health Department (SHD) for coordination.
   f. Ensure the RUR form is returned prior to irradiation.

2. 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.

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

4. Attach form to clipboard in the control room.

Revised: 15 Dec 1988

TAB A: REACTOR EXPOSURE ROOM ENTRY PROCEDURE

1. REFERENCES
   a. 10 CFR 20, "Standards for Protection Against Radiation"
   b. USNRC licenses: R-84, 19-08330-02
   c. AFRRI Radiological Safety Instructions

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 green and orange badged 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 themselves. All personnel who are granted either escorted or 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 or warm storage. The RFD is responsible for maintaining two seperate rosters in the prep area: one roster for personnel who have been granted unescorted access, and one roster for personnel who have been granted escorted 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 specical 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, wrist dosimeter, and pocket dosimeter.

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

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

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

(6) Ensure that all materials removed from the exposure room are properly labeled and entered on the exposure room entry log AFRRI FORM 130 (enclosure 2 of this procedure), and the activated materials control log.

(7) 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 lapsed 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 its 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. When working in a specific area for any extended time is expected the dose rate in that area will also be measured and recorded.

(1) If any accessible area inside the exposure room reads over 100 mR/hr (closed window), where extended stay time is possible, the SHD monitor will remain in the prep area until the room is closed. 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 (RUR's). If not, non-compliance will be reported to 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 (see enclosure 2). 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. The exposure rooms may be opened 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) The last survey indicated that there were no radiation levels in excess of 100 mR/hr in any area of the ER where extended stay time is possible.

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

(4) 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, 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 read and log dosimeters when leaving the exposure room using the dosimeter log in the prep area. 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 entering the prep area 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; insure all obstacles are clear of the door (including ropes).

c. Secure both entrances to the prep area.

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

e. 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.

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

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

h. 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 rad waste unless prior arrangements have been made with the reactor/SHD staffs for storage. Any material 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 and enclosure 1 of this procedure.

(2) All special equipment that has been activated such as chambers, rotaters, 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 the disposition of activated materials, section 7.c. of this procedure.

c. DISPOSITION OF ACTIVATED MATERIALS

(1) All materials coming out of the exposure rooms will fall into one of two categories. Category one consusts of materials that are to be removed from the prep area and category two are those materials designated to remain in the prep area. Prep area materials should be tagged with a yellow radiation material label, or tag filled in "Prep Area Materials" or painted yellow. Materials labeled or painted yellow are not to be removed from the prep area without SHD approval.

(2) If tagged material must be returned to the exposure room before it has been cleared from the activated materials log, return the material with the label to the prep area before the exposure room opening. At the time of the exposure room opening, give the tag to the SHD representative who will then clear the materials from the log. When the materials come out of the exposure room a new log entry will be made and a new number assigned the materials.

(3) When materials to be remove from the prep area come out of the exposure rooms, the materials must be tagged appropriately and an entry must be made in the activiated materials control log. The tagging procedure and information that must be entered in this log is as follows:

(a) ITEM NUMBER: will be assigned by the SHD monitor in sequence and be prefixed by an "R" for reactor, "L" for Dr. Ledney, "Z" for CDR Zeman, etc. The next character in the item is the calender year. The last character is the sequential item number.

EXAMPLE: L87-005

"L" indicates Dr. Ledney is the Priciple Investigator
"87" is the calender year
"005" indicates it is the fifth activated item removed from the prep arfea by Dr. Ledney in 1987

Each item that has been activated must be assigned an activated item number, i.e., if a dog was activiated in a plactic cage, both the dog and the cage must be numbered and tagged; or if an activated camera is tagged, both the lens and the camera must be numbered if the lens is removed.

NOTE: All labels must be kept with the materials until the materials are disposed of as regular waste or as radioactive waste, or cleared by SHD. In any case the tags must be returned to SHD to facilitate removing activated items from the log.

(b) ITEM DESCRIPTION, AFRRI NUMBER, SERIAL NUMBER, enter a brief description of the item removed. The AFRRI number and serial number shall be entered if applicable.

(c) BACKGROUND LEVELS, enter the background radiation levels of the area where the survey is conducted.

(d) LEVELS ON CONTACT CLOSED WINDOW, LEVELS ON CONTACT OPEN WINDOW, enter the radiation levels detected on the surface of the item being surveyed using both open and closed windows.

(e) SMEAR RESULTS, enter the results of the smear test if it was taken, see section 7.b.1. of this procedure.

(f) LOCATION MATERIAL REMOVED TO: enter the lab or area to which the materials are being taken. Ensure that the lab is qualified to hold the radioactive materials in accordance with enclosure one to this procedure, all appropriate radiological Safety Instructions and Health Physics Procedures.

(g) PERSON REMOVING MATERIALS, enter the name of the investigator or technician that is taking the materials, ensuring that the person is listed under the principal investigators authorization for handling radioactive materials.

(h) SIGNATURE OF PERSON REMOVING MATERIAL, have person receiving custody of the materials sign the log with the understanding that the materials being received have been activated.

(i) INITIAL OF SHD PERSONNEL, the person that released the activated materials will initial here.

(j) REMOVE FROM LOG, this space is to be checked off when the radioactive material has decayed below activation action levels indicated in enclosure 1 to this procedure.

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 insure that no personnel remain in the room. To ensure compliance the 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.

Revised: 15 Dec 1988

TAB B: CORE EXPERIMENT TUBE (CET)

General: ALARA principles will be practiced during CET operations.

Specifics:

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 element F28. 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, ensuring that the CET remains as low in the water as possible.
   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. Bring the reactor up to the appropriate power.
   e. After notifying the reactor operator on console, drop or lower the rabbit into the core WITH THE CAP UP. Ensure that this individual spends 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 and that they are wearing all required (whole body TLD, pocket chamber and wrist) dosimetry. 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 upper end indicator is visible on the cable; 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 very low (<1 mR/hr), store rabbit or irradiated material in a lead pig or storage cask.

i. 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 element into fuel position F28.

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.

Revised: Dec 1987

TAB C EXTRACTOR SYSTEM

GENERAL: The extractc- 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 and place lead bricks on them.
      (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) Connect the electrical cable to the limit switch.
      (6) 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) Tie the string from the end of the small tube to the end of the wire cable.
      (4) Pull the string in the large tube slowly while having someone inside the room guide the string.
      (5) 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).
      (6) While someone else is pushing the table toward the wall, insert two screws into the holes on the securing bracket (beneath the table).
      (7) Position and tighten clamp over the joint; position carrier in tube and connect cable to each end; removethe 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.
   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 power operations, ensure that the following requirements are met:

(1) The prep area is sealed off.

(2) A Safety & Health Department (SHD) monitor is present.

Revised: Jan 1984

TAB D 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 833 (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 botton 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.

Revised: 15 Dec 88

TAB E 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 I - Tab B).

Specific:

1. All operations will be supervised by an SRO.
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 during the removal of samples from in-pool or in-core locations.
4. The removal of experiment 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 experiments.

Revised: Jan 1985

PROCEDURE II 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.
   c. A record of operations will be kept for each trainee/operator.

Revised: Jan 1985

PROCEDURE III MAINTENANCE PROCEDURES

General: Maintenance procedures are provided in other references.

Specific:

1. Preventative Maintenance procedures for each item of the reactor systems are provided in the maintenance logbook.

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.

3. Malfunctions are annotated in the Malfunction Logbook. Each entry is 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.

Revised: Jul 1982

PROCEDURE IV 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.

Specific:

1. Reactor Room:
   a. CET Operations: See Procedure I-Tab B.
   b. Working inside chained in area around pool: The reactor operator on the console shall be responsible for controlling entry into the chained area around the pool.

2. Warm Storage: See HPP 3-3.

3. Prep Area: See Prep Area Briefing.

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

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

6. Upper and Lower Equipment Rooms:
   a. No written radiation protection procedures are required for entry into these rooms.
   b. However, 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.

Revised: 15 Dec 88

PROCEDURE V 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.

Revised: 15 Dec 1988

PROCEDURE VI
EMERGENCY PROCEDURES

General: The reactor emergency organization, emergency classes, and emergency action levels are set forth in the current copy of the AFRRI Reactor Emergency Plan.

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 EAS.
   d. Do NOT lock reactor area doors.

Dec '87

PROCEDURE VII REACTOR CORE LOADING AND UNLOADING

General: Loading and unloading of the reactor core shall be under the supervision of the Reactor Facility Director or the Reactor Operations Supervisor. These procedures are superseded in the following situations: during CBT Operations (see Procedure I-Tab B) and during annual shutdown maintenance (see the current Annual Shutdown Checklist).

Specific:

1. Setup

   a. Ensure at least one nuclear instrumentation channel is operational.
   b. Ensure an operator monitors the reactor console during all fuel movements.
   c. Check new fuel elements prior to insertion into the core; this includes cleaning, visual inspection, and length and bow measurements.
   d. If irradiated fuel elements are to be removed unshielded from the pool, a Special Work Permit (SWP) will be obtained from the Safety & Health Department (SHD); fuel elements with a power history (greater than 1 KW) in the previous two weeks shall not be removed 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 rods; continue to next step.

   b. Load elements in the following order:

      (1) Load the "B" ring thermocouple element.
      (2) Load the "C" ring thermocouple element.
      (3) Install temperature measurement system (to measure fuel temperature.

(4) Install any other thermocouple elements.
(5) Complete loading of "B" and "C" ring elements (total of 18 elements).
(6) Load "D" ring (total of 33 elements)
(7) Load the following "E" ring elements: 1,2,4,6,8,9,10,12,14,16,17,18,20,22,24 (total of 48 elements).
(8) Complete the "E" ring (total of 57 elements).
(9) Load the following "F" ring elements: 1,5,9,13,17,21,22,23,27 (total of 66 elements).
(10) Load two elements per step until critical loading is achieved.
(11) Load core to $2.00 excess reactivity.
(12) Estimate control rod worth using rod drop techniques.
(13) Estimate the control rod worth of the remaining unloaded elements.
(14) 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.
(15) Calibrate the TRANS rod.
(16) Estimate the shutdown margin.
(17) Estimate K-excess with a fully loaded core (must not exceed $5.00).
(18) Load core to fully operational load and recalibrate all control rods.

3. Core Unloading:

a. The reactor core will be unloaded starting with "F" ring and ending with the "B" ring.

b. The fuel elements will be individually removed from the reactor core, identified by serial number, and placed in either the fuel storage racks or a shipping cask.

c. If elements are to be loaded into a shipping cask, perform a complete cleaning of the cask and check for radiological contamination prior to 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, insure criticality monitoring in accordance with 10 CFR 70 is in place.

Revised: 15 Dec 1988

PROCEDURE VIII REACTOR OPERATIONS

General:

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

Specific:

1. Each line on the daily and weekly checklists shall be initialed by the Reactor Operator or Trainee who performs that item.
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 may be performed (Tab B1).
3. Record at the top of each page the SRO On-Call for that date.
4. Perform K-excess measurement (Tab D).
5. 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 I, Tab B).
   e. Pneumatic Transfer System (Procedure I, Tab D).
6. Perform Weekly Operational Instrument Checklist once during calendar week (TAB H).
7. At the end of each day in which a Daily Operational Startup Checklist or Daily Safety Checklist has been completed, perform Daily Operational Shutdown Checklist (Tab I).
8. Complete the monthly summary (Tab J).

Revised: 15 Dec 1988

TAB A 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 no'

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 numbere'.

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.

4. 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) Mode I or IA Steady State

(b) Mode II Square Wave

(c) Mode III 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 4.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 checklists.

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

(11) Reactor calibrations and data.

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) Reactor malfunction,to include the reactor systems and support equipment taken out of service for maintenance and returned to service.

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

5. 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.

TAB B: DAILY OPERATIONAL START UP CHECKLIST

Checklist # ______________________ Date ______________________

SENIOR SRO PRESENT/ON CALL __________ Performed 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 δp (inches of water) .... __________
5. Stack absolute filter δp (inches 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. Tank water level below full mark (inches) .... __________
4. Visual inspection of core and tank .... __________
5. Number of fuel elements and control ...fuel elements __________
   rods in tank storage control rods __________
6. Air particulate monitor (CAM)
   (a) Operating and tracing .... __________
   (b) Alarm test complete .... __________
7. Door 3162 SECURED .... __________
8. Stack gas monitor quality assurance checked .... __________

VI. REACTOR CONTROL ROOM

1. Emergency air system RESET ....
2. Console recorder dated ....
3. Stack gas and fuel temp. recorder dated ....
4. Logbook dated and reviewed ....
5. Water monitor box (all conductivities must be > 0.5 M -cm)
   (a) Background activity (mA)...........................
   (b) Alarm test completed and alarm reset to 0.5 mA
   (c) Water monitor box conductivity [MΩ-cm] ....
   (d) DM1 conductivity [MΩ-cm] ....
   (e) DM2 conductivity [MΩ-cm] ....
6. Stack gas flow rate [Kcfm] ....
7. Stack gas monitor
   (a) Background (cpm) ....
   (b) Alarm check ....
   (c) High alarm set to 800 MPC Ar-41 ....
8. Stack particulate monitor
   (a) Background (cpm) ....
   (b) Alarm check ....
   (c) High alarm set to 2.0 E3 cpm ....
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. Timer on ....
11. TV monitors and beeper system on ....
12. Source level on log channel > 0.5 Cps ....
13. Check 0.5 Cps source RWP ....
14. Time delay operative ....
15. Operational channels

Cal. log switch Pos. #	Range Switch	%Linear	%Log
1	0.3 watts		
2	30 watts		
3	300 watts		
4	1 kw		
5	10 kw		
6	3 Mw		

16. Rod raising interlock for mode I ....
17. Rod raising interlock for mode III ....
18. Zero power pulse (obtain signal w/trip test) .....
19. SCRAM checks (at least one per rod)
    (a) Safety flux 1 ...
    (b) Fuel temp 1 .....
    (c) HV loss safety 1
    (d) Manual ....
    (e) Reactor key .....
    (f) Timer ....
    (g) Emergency Stop ..
    (h) Pool H_2O level ..
    (i) Fuel temp 2 .....
    (j) Safety flux 2 ...
    (k) HV loss safety 2
20. Water temperature (inlet) ....
21. Period trip test for 1 kw interlock ....
22. CAM high level audible alarm check ....

Checklist # ____________ DATE ____________

SENIOR SRO PRESENT/ON CALL ____________ PERFORMED BY ____________

OPERATORS ____________________ TIME COMPLETED ____________

I. EQUIPMENT ROOM (RM - 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 (RM-2158)

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

IV. PREPARATION AREA

Visual Inspection of Area .... ________

Apr '88

TAB C NUCLEAR INSTRUMENTATION SET POINTS

General: These set points may be adjusted for a specific operation by of 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 C
b. High Flux 1 & 2:	110% (1.1 MW)
c. Safe Chambers 1 & 2 HV Loss:	Loss of 20%
d. Pulse Timer:	0.555 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 Bulk Temperature:	50 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. Stack Particulate:	2.0E+3 CPM
e. Water Monitor Box Gamma:	0.5 mA
f. Criticality Monitor (R5):	50 mR/hr day - 20 mR/hr night or as directed

Jul '82

TAB D K-EXCESS

1. Withdraw SAF and SHIM rods 100% and withdraw the TRANS rod 25%.

2. Use the REG rod to bring the reactor to cold critical at 15 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 15 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 and on the Monthly Summary Sheet.

*Note: Use the curves for position 567 when doing K-excess at 290.

Jan '85

TAB E: STEADY STATE OPERATION (MODE I/IA)

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

Specific:

1. Set the linear channel range select switch to the appropriate scale for the desired power (at this power the linear pen should be as close to 50% as possible).

2. Set the mode switch to manual mode.

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

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

   a. Set the proper percentage on the flux control to obtain the desired power.

   b. Raise the TRANS, SAFE, and SHIM rods to the appropriate banking.

   c. Raise the REG rod approximately 5%.

   d. When the servo has locked in, switch to automatic mode.

   e. When the reactor reaches critical, fine tune the flux control for the desired power.

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.

Note: For runs greater than 800 KW, adjust alarm points on R-1 and R-5 to full scale.

Apr '88

TAB F: SQUARE WAVE OPERATION (MODE II)

General: The square wave mode can not be used above 900KW.

Specific:

1. If appropriate, set timer for run duration and flip timer SCRAM switch to "ON".

2. Set the TRANS drive anvil as follows:

   a. If the TRANS rod is not to be used for criticality, use the applicable rod worth curve to determine the anvil position for a 75 cent insertion and raise the anvil to that position. Adjust the REG rod to 90 percent; achieve criticality using the SHIM and SAFE rods.

   b. If the TRANS rod is to be used for criticality, withdraw the REG rod 90 percent, SAFE and SHIM rods 100 percent, and adjust to critical configuration using the TRANS rod. When a critical configuration has been reached, drop air from the TRANS rod and adjust the anvil position for a 75 cent insertion above the critical position.

3. Adjust power range select switch to the desired range.

4. Set square wave percent demand dial to 80 percent of final desired power.

5. Set flux controller dial to final desired power level.

6. Switch into square wave mode, making sure the TRANS rod ready light is "ON".

7. Depress ready/fire button.

8. After the servo has locked in, raise the square wave percent demand dial setting to the final desired power.

9. Switch to manual mode and lower REG rod to 80 percent while raising the TRANS rod; maintain desired power level and then switch to automatic mode.

10. Scram the reactor with the timer or manually, as appropriate; move the core if applicable.

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

May 88 (Pending RRFSC Review)

TAB G: PULSE OPERATION (MODE III)

General: Pulses above $3.50 must be approved by the RFD or ROS. Specification on the RUR may be used to meet this requirement.

Specific: For pulses fired from COLD CRITICAL, omit steps 2, 5, 6, and 8. For pulses fired from SUBCRITICAL, omit step 7.

1. Set the alarm points on R-1 and R-5 (the 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 ($) TRANS rod - desired pulse ($) value

3. Bring the reactor to cold critical (15 watts) with the three standard rods, using a rod configuration commensurate with core position.

4. Stabilize in manual mode.

5. SCRAM the transient rod.

6. Raise the transient rod anvil to 100%.

7. Raise the transient rod anvil to the desired pulse position.

8. Let the power decay to approximately one watt.

9. Place power range select switch on the "3 MW-PULSE" position.

10. Place mode select switch in "PULSE HI" (greater than or equal to $2.15) or "PULSE LO" (less than $2.15) position, as appropriate.

11. Fire pulse by depressing "READY" button on the console (Note: Power must be below 1 KW).

12. Record data in the reactor operations logbook as follows:

    a. Insure the pulse detector selector switch is on Detector #2.

    b. Turn reactor pool lights and reactor room overhead lights OUT.

    c. NVT: Read from Safety Channel 2/NVT meter on right side of console. Multiply the Safety Channel 2/NVT meter reading by 1.45 to obtain actual NVT reading in MW-s. Full scale is 43.5 MW-s.

    d. NV: Read blue pen on chart recorder, center of console with the following equations:

        1. Pulse Hi switch: 2500 MW X (% of scale reading) X 1.45

        2. Pulse Lo switch: 500 MW X (% of scale reading) X 1.45

    e. Fuel Temperature: Read red pen of chart recorder, center of console, where 100% = 600 C.

13. Reset mode select switch to manual mode after reading are recorded

14. Reset R-1 and R-5 to their normal alarm points.

WEEKLY OPERATIONAL INSTRUMENT CHECKLIST

CHECKLIST No. ____________________ DATE ____________________

PERFORMED BY ____________________ REVIEWED BY ____________________

I. SAFETY CHANNEL ONE

A. Raise rod ≈2%, depress and release switch marked HV # 2 in left hand drawer, observe and reset scram ________

B. Rotate operate switch to zero, check meter for zero, reset scram ________

C. Rotate operate switch to calibrate, check meter for 100 %, reset scram ________

II. OPERATIONAL CHANNEL HIGH VOLTAGE

Depress and hold in switch HV # 1 in left hand drawer. Attempt to raise control rod ________

III. SAFETY CHANNEL TWO

A. Raise rod ~2%, depress and release switch HV trip test in right hand drawer. Observe and reset scram ________

B. Rotate operate switch to zero, check meter for zero, reset scram ________

C. Rotate operate switch to calibrate, check meter for 100 %, reset scram ________

IV. NV-NVT

A. NV - In manual mode, set 20% on safety channel # 2 with trip test knob. Switch to pulse HI, chart recorder should read 20%. Repeat for 40%, 60%, 80%, and 100%. Check scram at 110% ________

B. NVT - Check NVT circuit by procedure 3.3.5.2 through 3.3.5.4 in console maintenance manual. ________

V. FUEL TEMPERATURE NO. 1

Rotate switch to calibrate position, observe 100% on meter, reset scram ________

VI. FUEL TEMPERATURE NO. 2

Rotate switch to calibrate position, observe 100% on meter, reset scram ________

VII. WATER LEVEL INDICATOR

A. In pool, east side, depress float on water level indicator ________

B. Observe scram on console, (scram indication should reset automatically) ________

VIII. WATER CONDUCTIVITY

List resistivity readings for previous week from daily startup checklists. Determine that average at each point is greater than 0.5 $M\Omega$-cm.

	MON	TUE	WED	THU	FRI	*AVE*
Monitor Box						
DM 1						
DM 2						

IX. RADIATION MONITORS

A. Test alarm functions for high level and failure

Monitor	*Failure Alarm Functional*	*HIGH - Level Alarm Functional*
R-1		
R-2		
E-3		
E-6		
R-5 (criticality)		
Reactor Rm APM		
Gas Stack Monitor		

B. Reset Alarms ________

AFRRI FORM 66 (PREVIOUS EDITIONS OF THIS FORM ARE OBSOLETE.)
19 OCT 84

DAILY OPERATIONAL SHUTDOWN CHECKLIST

Checklist No. ______________ Date ______________________________
Time Completed ______________ Performed 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	________
5. Print out an hourly report from the Stack Gas Monitor	________

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. Dorr 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

1. Lobby alarm audio ON .... ________

VI. REACTOR CONTROL ROOM (Room 3160)

1. Reactor tank lights OFF .... ________
2. Timer OFF .... ________
3. TV monitors OFF .... ________
4. Console LOCKED .... ________
5. Diffuser and secondary pumps OFF .... ________
6. Purification and primary pumps ON. .... ________
7. Beeper system turned OFF .... ________
8. Reactor monthly usage summary completed .... ________
9. Exposure room camera power supply turned OFF ..... ________
10. 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	

REACTOR MONTHLY USAGE SUMMARY

	K-EXCESS AT POSITION 833	NUMBER START-UPS	NUMBER OF PULSES	PULSES ≥ $2.00	POSITION STEADY STATE/PULSES					TOTAL STEADY STATE POWER	TOTAL PULSE POWER	TOTAL POWER	UNPLAN SHUT-DOWNS
		TOTAL	TOTAL	TOTAL	ER NO.1 (231)	CENTER (567)	TUBES (833)	ER NO.2 (983)	OTHER ( )			TOTAL TO DATE	TOTAL
CARRY-OVER													
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AFRRI FORM 74
6 JULY 82

MONTH ______

Dec '87

TAB K: STACK GAS MONITOR PROCEDURE

GENERAL: This procedure specifies all the requirements for operation of the Stack Gas Monitor (SGM) in the reactor room. This instrumentation is used to monitor, measure, and record the amount of Argon-41 released to the environment through the reactor stack.

SPECIFIC:

1. QUALITY ASSURANCE PROCEDURES: A quality assurance check is performed daily, prior to reactor operations, as part of the reactor start-up. This check is performed in the following manner:

   a. The particulate filter is changed and the old filter is discarded as radioactive waste.

   b. The date and time of this quality assurance check is entered in the log.

   c. The Detector voltage system set point is checked and recorded in the log:

   d. The air sampling flow rate (should be greater than 3.5 cubic feet per minute) is recorded in the log.

   e. 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 setpoint. The audible alarm can be silenced by pushing the red button on the front of the SGM cabinet.

   f. After 5 minutes have elapsed, an historical one minute report printout should be made and the counts per minute for the third minute of high level readings should be recorded in the log.

   g. The alarms should be acknowledged in the log with the initials of the person performing the check and on the SGM by pushing the "acknowledge" button on the keyboard.

   h. The check source should be removed and the detector cover should be replaced.

i. Finally, the counts per minute reading (paragraph 2f) should be checked against the plot of counts per minute versus julian date to determine if it falls within a plus or minus 5% deviation for the detector and check source. This check provides the necessary quality assurance for the SGM system prior to conducting any reactor operations for the day.

2. HISTORICAL REPORT REQUIREMENTS: There are two historical report requirements for proper documentation of argon-41 releases.

   a. The SGM automatically prints out the amount of argon-41 released every 6 hours. Each of these 6 hour reports must be taped into the historical release data log.

   b. Finally, at the end of each day, as part of the reactor shut-down, an historical one hour report should be printed and taped into the historical release data log. This report contains the average counts per minute for each of the 30 previous hours. This information can be used to calculate the amount of argon-41 released through through the reactor stack.

3. SYSTEM CHANGES OR OBSERVED ABNORMALITIES: Any changes to the system set points or observed abnormalities should be reported immediately to the Reactor Facility Director and to the Safety & Health Department.

Revised: Dec 1986

PROCEDURE IX 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.

2. Mercury thermometers are not allowed in the reactor room at any time.

ATTACHMENT C
SAFETY ANALYSIS OF MODIFICATIONS TO UPGRADE THE REACTOR FACILITY AT THE ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE

10 CFR 50.59 SAFETY EVALUATION REPORT OF THE NEW REACTOR INSTRUMENTATION AND CONTROL SYSTEM AT THE ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE

11 MAY 1988

Mark Moore
Ken Hodgdon
Angela Munno

ABSTRACT

This report describes changes to the reactor facility at the Armed Forces Radiobiology Research Institute (AFRRI) in Bethesda, Maryland. This Safety Evaluation Report (SER) meets the requirements of Title 10, Code of Federal Regulations, Part 50.59 (10 CFR 50.59), and provides the basis for the conclusion that the changes to the facility involve no unreviewed safety questions and, in fact, are improvements in the facility design at AFRRI. In order to accomplish these changes, the Facility Safety Analysis Report (SAR) must be modified. The body of this report contains a description and safety analysis of the SAR changes. Excerpts from the SAR and the proposed changes are included as appendices.

Note: Under 10 CFR 50.59, a licensee may make changes to its facility provided that no changes are made to the Technical Specifications, and that there are no unreviewed safety questions. The conditions for unreviewed safety questions are outlined in 10 CFR 10.59.a.2, and are summarized below:

If the affected equipment is related to safety:

i. The probability of occurrence or the consequences of an accident or equipment malfunction shall not be increased.

ii. The possibility for an accident or malfunction of a different type than previously evaluated in the SAR shall not exist.

iii. The margin of safety as defined in the Basis for any Technical Specification shall not be reduced.

TABLE OF CONTENTS

INTRODUCTION

Present conditions at the Armed Forces Radiobiology Research Institute (AFRRI) require that modifications be made to upgrade the reactor facility. The changes being made to the Facility Safety Analysis Report (SAR) include: The installation of a new Reactor Instrumentation and Control System and the installation of three new stepping-motor standard control rod drives.

AFRRI's current reactor instrumentation system is a 1972 vintage unit (hereafter, refered to as the current (present), old, or 1972 console) salvaged from the 1977 decommissioning of the Diamond Ordnance Radiation Facility and was installed at AFRRI in 1978. The design life of this unit is 10 years. Because this console is now 16 years old, maintenance down time has increased and is expected to continue to increase over the next five years.

The console's functional utility is now continuously diminishing due to the progressive obsolescence of many of its electronic components. Although the obsolescence of these components does not effect the nuclear safety of the system, it is a problem operationally. Many of these electronic components are no longer manufactured; consequently, direct replacements are unobtainable. Redesign of selected circuits to use currently available electronic components would require, in each case, a safety review by the reactor safety committee and possible review and approval by the NRC.
Estimated hardware costs to entirely redesign, replace, and upgrade AFRRI's existing console exceed the cost of buying a new instrumentation system.

Failure analyses of current console components indicate that, under normal circumstances, AFRRI has sufficient spare parts to sustain its present operational capability for less than 2 years. Then it is expected that AFRRI would become involved in serious down time problems.

AFRRI's control rod drive system also suffers from the same progressive obsolescence, increasing maintenance down time, and spare parts unavailability as the control console.

Acquiring a new state-of-the-art console and control rod drive system using integrated circuits and microprocessor technology will resolve these problems and provide for reliable operation of the AFRRI Reactor Facility through the year 2000.

This new state-of-the-art microprocessor-based instrumentation and control system will replace the current control console while improving the existing operational capabilities and safety characteristics. The new system will increase reactor operational performance through increased productivity, improved efficiency, increased reliability, improved experiment reproducibility, and increased maintainability. Productivity will be improved through increased reactor operating time due to the system performing automatic self-checks of daily instrumentation checkouts, and through decreased operator training time - operators will become proficient in a much shorter length of time. The new system will increase efficiency in reactor operators' time by automatically logging reactor data or allowing keyboard entry of nonoperational but essential information pertinent to reactor operations. Experiment reproducibility will be improved through increased pulse accuracy and repeatability and through improved Auto Mode capabilities. In Pulse Mode, the system will provide prompt waveform analysis: peak power, energy, half power width, reactivity insertion, minimum period, and peak fuel temperature are measured and calculated automatically and reported promptly to the operator in either graphic or nongraphic mode. In Automatic Mode, the operator will select the desired power level, run duration (SCRAM time), and which rods will be servoed, then position the banked rods, select the Automatic Mode and let the Reactor Control System perform the run. The new system will increase maintainability through state-of-the-art system maintenance design and layout, line replaceable units and on-line system diagnostics. System safety will also be improved through the performance of periodic self-diagnostics that determine if the unit is in a safe operational status. These diagnostics will display error messages reporting failures to the operator and will automatically place the reactor in a safe neutronic configuration. Additionally, the system will have improved Electromagnetic Interference (EMI) protection through shielding, optical isolation, and digitizing data at near core locations, and will reduce cabling requirements by collecting data in the reactor room and then routing that data to the Control Console Computer via serial data trunks.

The Code of Federal Regulations (Title 10, Part 50.59) requires that modification of a portion of a licensed facility as described in the facility SAR be documented with a written safety evaluation. Such documentation provides the basis for determining that the change does not involve an unreviewed safety question. An unreviewed safety question according to 10 CFR 50.59 involves (1) the increase of probability of occurrence or the increase of consequences of an accident or malfunction of equipment important to safety compared to that situation previously evaluated in the SAR, or (2) the possibility for an accident or malfunction of a different type than previously analyzed in the SAR, or (3) the reduction in margin of safety as defined in the SAR.
Based on the analyses in this Technical Report, it has been determined that the proposed changes to the Reactor Facility do not involve any unreviewed safety questions and will actually improve the facility design at AFRRI.

This technical report describes changes and modifications made to the AFRRI reactor facility as depicted in the facility's SAR. These changes have been reviewed by the Reactor Facility Director and found to contain no unreviewed safety questions. This report is submitted to the Reactor and Radiation Facility Safety Committee (RRFSC) for their concurrence that conditions of 10 CFR 50.59 are met. These conditions are that no unreviewed safety questions are present and that the changes made do not increase the probability of occurrence or the consequences of an accident or malfunction.

The proposed modifications require minor changes to the SAR. The body of this report contains a description and safety analysis of the 10 CFR 50.59 SAR changes. Appendix A contains a specific page/section index of all of the SAR changes. Appendix B contains excerpts from the SAR, for each of these 10 CFR 50.59 modifications.

The new Digital Reactor Instrumentation and Control System has been designed to be safer than the present AFRRI control system which has been evaluated in the AFRRI TRIGA Mark F Reactor SAR. This has been accomplished by continuing to hardwire all safety circuits in a redundant, fail safe configuration. These safety circuits are completely independent of the data acquisition computer (DAC) and the control system computer (CSC). This means that if either or both computers were to fail, the failure cannot prevent the reactor from scramming. On the other hand, critical functions of the computers are monitored by "watch-dog-timers". If the computers fail to update the timers in a predetermined fashion, the redundant, hardwired watch-dog-timers will scram the reactor.

As a result, the new Digital Reactor Instrumentation and Control System has equal or greater safety built-in than the present AFRRI control system, which has SAR approval.

FACILITY MODIFICATIONS SAFETY EVALUATION

The installation of the new Reactor Instrumentation and Control System at the AFRRI TRIGA Mark F reactor facility will provide equal or greater operational and safety capabilities with a higher degree of reliability than the current instrumentation.

OVERVIEW

The basic elements of the new Reactor Instrumentation and Control System (see Figure 1) will consist of a Control Console, a Data Acquisition and Control Unit (DAC), two independent Power Monitor and Safety Systems, an Operational Channel, and a Pulse Channel. This system was design and built in accordance with ANSI/ANS-15.15-1978 "Criteria For The Reactor Safety Systems of Research Reactors".

The Control Console will be a desk-type unit located in the AFRRI Reactor Control Room. Operators will conduct reactor operations using a set of control switches and a keyboard located on the console, and the operators will receive feedback information through a high-resolution color monitor, a status monitor, indicators, and annunciators.

The heart of the control console will be the Control System Computer (CSC). Operators will adjust the rod positions by issuing commands to the CSC, which will transmit these commands to the DAC. The DAC will reissue the commands to the drive mechanisms. During reactor operations, the CSC will receive raw data from the DAC, process this data, and present the data in meaningful engineering units and graphic displays on a number of peripheral systems.

The CSC will operate two color CRT monitors. A high-resolution color graphics CRT (Reactor Control CRT) will provide the operator with a real-time graphic display of the reactor status. This CRT will display the important operational parameters using bar graphs and digital readouts and will alert the operator to any abnormal or dangerous conditions. A Reactor Status CRT will display pertinent diagnostic messages, reactor status, and facility status information.

The CSC will also interface with a near-letter-quality printer, allowing the logging of reactor information as required by the reactor operator. Historical data will be saved in the CSC's internal memory and on command from the operator be replayed, printed, or transferred to removable disks for permanent storage. This will provide the capability to maintain records of pertinent reactor statistics and to replay reactor operational records for training and analysis. In addition, the CSC will operate a color graphics printer capable of printing steady-state and pulse mode data as well as producing point-line plots. Finally, the CSC will interface to real-time recorders of reactor power and fuel temperature.

The DAC will be located in the AFRRI Reactor Room adjacent to the reactor and will provide high-speed data acquisition and control capability. The DAC will monitor the two independent Power Monitor and Safety Systems, the Operational Channel, the Pulse Channel, the fuel temperature, water level and temperature, and control rod positions. The DAC will, on command from the CSC, reissue the commands to raise and lower the control rods or scram the reactor. The DAC will communicate with the CSC via serial data trunks. The secondary trunk will serve as a backup should the primary trunk fail. These serial data trunks will drastically reduce the wiring requirements between the Reactor Room and the Control Console.

The Power Monitor and Safety Systems will monitor the power from 1% to 120% of full power (1.0 megawatts) and shut the reactor down (SCRAM) in the event of an overpower condition. The Operational Channel will monitor the power from source level to full power and the rate of power change (from -30 to +3 second period) in the steady state modes.

The Pulse Channel will monitor the power level up to 5000 megawatts in the pulse mode. This channel will use an ion chamber, a photo diode detector, or some other acceptable pulse monitoring detector. The DAC will collect information from the pulse channel and transmit the data to the CSC for processing.

The control console will have 8 Hardwired (Analog) LED Bargraph indicators which are located on the left side of the console. These hardwired channels include the two High Flux Safety Channels, the two Fuel Temperature Safety Channels, the Operational Wide-Range Log Channel, the Period Channel, and the Pulse NV and NVT Channels. Located below these analog bargraphs are the Operational Multirange Linear Channel and Fuel Temperature Channel strip chart recorders. These items are all hardwired and are completely independent of the CSC and DAC computers, and therefore, will provide information to the reactor operator at all times, even should the CSC and DAC computers fail.

AFRRI is also replacing its three 1960 vintage Standard Control Rod Drives with three new Standard Control Rod Drives using pulsed motor drive systems. These stepping motors operate on phase-switched dc power. These motors drive a pinion gear (connected to the Magnet Draw Tube) and a 10-turn positive feedback potentiometer via a chain and pulley gear mechanism. Except for the drive motors, the new control rod drive assemblies will be the same as the current control rod drive assemblies.

REACTOR SAFETY SYSTEM DESCRIPTIONS

HIGH FLUX SAFETY CHANNELS ONE AND TWO

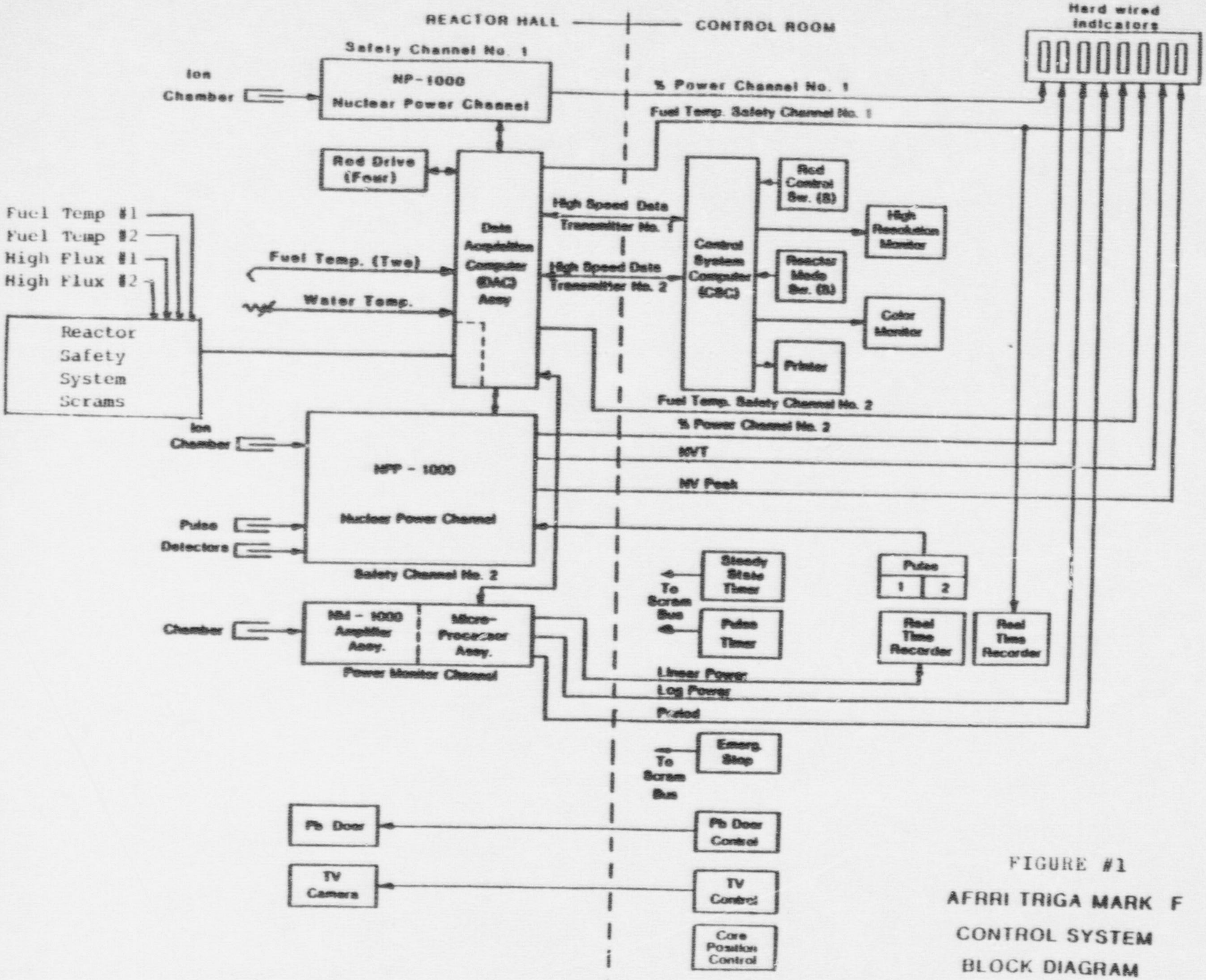


FIGURE #1

AFRRI TRIGA MARK F

CONTROL SYSTEM

BLOCK DIAGRAM

2/88

High flux safety channels one and two report the reactor power level as measured by two ion chambers and a pulse detector placed above the core in the neutron field. Each safety channel is a part of one multifunction NP-1000 neutron power channel. For safety reasons (simple redundancy) two independent NP-1000's are used and they operate identically during steady state operation. Each channel consists of an ion chamber placed above the core and the associated NP-1000 electronics. The steady state power level is displayed on two separate LED bargraph indicators and on the reactor control CRT.

During pulse operation, high flux safety channel one is shunted and the sensor for high flux safety channel two is switched to a third, independent pulse detector placed above the core. High flux safety channel two measures the peak power level achieved during the pulse (NV) and the total integrated power produced by the pulse (NVT) and is therefore specified as an NPP-1000 instead of an NP-1000. However, it should be noted that both safety channels operate with identical NP-1000 circuitry. Calibration of the NP-1000's is done automatically during the Daily Startup Checklist when the operator initiates the "pre-checks" by activation of the Prestart Check Switch on the control console's Mode Control Panel. Any failures detected during the prechecks will be automatically reported to the operator via the reactor status CRT.

The high flux safety channels (NP-1000's) form part of the scram logic circuitry. When the steady state reactor power level, as measured by either high flux safety channel, reaches the maximum power level specified in the technical specifications, a bistable trip circuit is activated which breaks the scram logic circuit, causing an immediate reactor scram. Similarly, when the reactor power level during pulse operation, as measured by high flux safety channel two, reaches the maximum pulse power level specified in the technical specifications, a bistable trip circuit is activated which causes an immediate reactor scram.

FUEL TEMPERATURE SAFETY CHANNELS ONE AND TWO

Fuel temperature safety channels one and two are independent of one another but operate in identical manners (simple redundancy). One thermocouple from each of the two instrumented fuel elements, one in the B-ring and one in the C-ring, provide inputs to fuel temperature safety channels one and two, respectively. The two fuel temperature signals are amplified and displayed on two separate bargraph indicators located on the reactor console and on the reactor control CRT. The fuel temperature safety channels have internal compensation for the chromel-alumel thermocouples and high noise rejection. Calibration of the Fuel Temperature Channels is done automatically during the Daily Startup Checklist when the reactor operator initiates the "pre-checks" by activation of the Prestart Check Switch on the control console's Mode Control Panel. Any failures detected during the prechecks will be automatically reported to the operator via the reactor status CRT.

In addition to providing information to the reactor operator on fuel temperature, the fuel temperature safety channels also form part of the scram logic circuitry. When the fuel temperature, as measured by either fuel temperature safety channel, reaches the maximum allowable fuel temperature specified in the technical specifications, a bistable trip circuit is activated which breaks the scram logic circuit, causing an immediate reactor scram. The operational fuel temperature limit is usually set below the technical specifications limit to assure an adequate degree of reactor protection.

The combination of the two independent High Flux Safety Channels and the two independent Fuel Temperature Safety Channels provides both simple redundancy and functional redundancy in terms of insuring that the Reactor Safety Limit as specified in the Technical Specifications is never reached.

SCRAM SYSTEMS

The scram logic circuitry (see Figure 2) assures that a set of reactor core and operational conditions must be satisfied for reactor operation to occur or continue in accordance with the technical specifications. The scram logic circuitry involves a set of open-on-failure logic relay switches in series: any scram signal or component failure in the scram logic, therefore, results in a loss of standard control rod magnet current and a loss of air to the transient rod cylinder, resulting in a reactor scram. The time between activation of the scram logic and the total insertion of the control rods is limited by the technical specifications to assure the safety of the reactor and the fuel elements for the range of anticipated transients for the AFRRI TRIGA reactor. The scram logic circuitry causes an automatic reactor scram under the following circumstances:

- The steady state timer causes a reactor scram after a given elapsed time, as set on the timer, when utilized during steady state power operations.

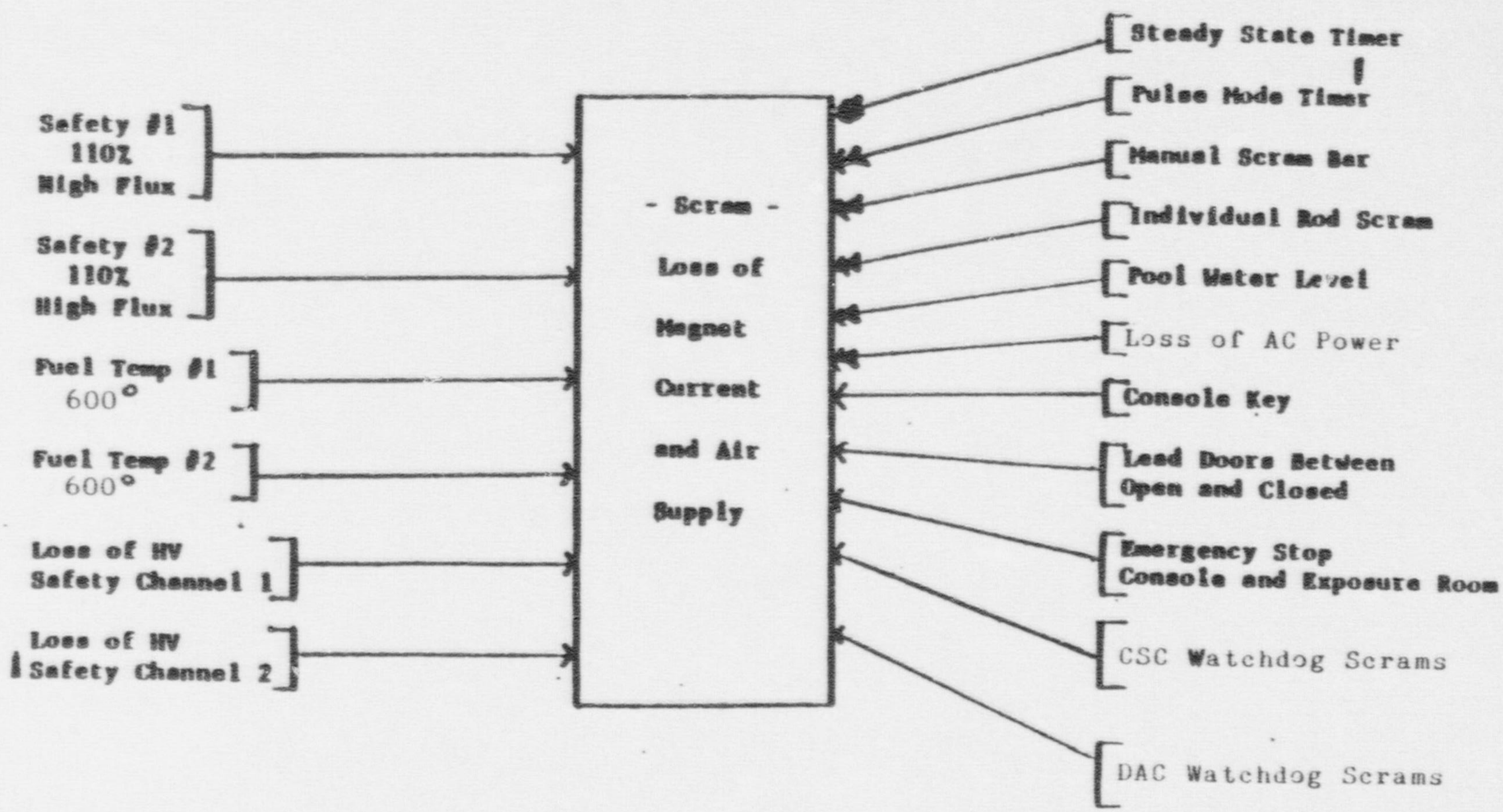
- The pulse timer causes a reactor scram after a given elapsed time, as set on the timer (in accordance with the limit specified in the technical specifications), during pulse power operations.

- The manual scram button located on the reactor console, allows the Reactor Operator to manually scram the reactor.

- Movement of the console key to the OFF position causes a reactor scram.

- The reactor tank shielding doors in any position other than fully open or fully closed will cause a reactor scram (this is part of the facility interlock system).

FIGURE #2: AFRRI REACTOR SCRAMS



- Activation of any of the emergency stop buttons in either exposure room or on the console causes a reactor scram.

- A loss of AC power to the reactor causes a reactor scram.

- High flux safety channel one causes a reactor scram at a reactor power level specified in the technical specifications for steady state modes of operation. This may be operationally set more conservative than the technical specifications limit.

- High flux safety channel two causes a reactor scram at a reactor power level specified in the technical specifications for steady state modes of operation. This may be operationally set more conservative than the technical specifications limit.

- A loss of high voltage to either of the detectors for high flux safety channels one and two causes a reactor scram.

- Fuel temperature safety channels one and two will each initiate a reactor scram if the fuel temperature, as measured independently by either channel, reaches 600°C (technical specification limit). This assures that the AFRRI safety limit (core temperature) of 1,000°C for AFRRI stainless steel clad cylindrical TRIGA fuel elements, as stated in the AFRRI technical specifications, is never approached or exceeded. The actual operational limit for the fuel temperature safety channels may be set lower than the technical specifications limit of 600°C.

- A loss of reactor pool water which leaves less than or equal to 14 feet of pool water above the core (technical specifications limit) causes a reactor scram. The actual operational limits for the pool water level may be set more conservatively than the technical specifications limit.

- One watchdog timer on the data acquisition computer and another one on the control system computer are required to be reset periodically by a program routine as a safeguard against computer component failures either in hardware or software. If the required response is not received within a definite time period, redundant normally open (fail safe) contacts interrupt the scram loop dropping the rods and shutting down the reactor. These watchdog timers are additional safety devices.

SINGLE FAILURE CRITERIA ANALYSIS

ANSI/ANS STD 15.15-1978 "Criteria for Reactor Safety Systems of Research Reactors" specifies that a Single Failure Criteria Analysis be performed on all non-redundant reactor safety systems. This analysis was performed by General Atomics for the new AFRRI TRIGA Reactor Instrumentation and Control System and is enclosed as Appendix C "AFRRI TRIGA Console (Safety) Scram System Single Failure Criteria Analysis." This analysis demonstrates that, except for the Reactor Key Switch (which does not perform a safety function except to prevent unauthorized startup), the Mean Time Between Failure of any single element of the new instrumentation scram system greatly exceeds (the MTBF's range from 23 years to 125 years) the design life of the new console (15 years). This analysis was performed for any single failure of the reactor safety system.

TRIGA REACTOR SAFETY SYSTEM FAILURE ANALYSIS

Although not required, a Failure Analysis was performed by the University of Texas and General Atomics of the new Reactor Instrumentation and Control System. This analysis is enclosed as Appendix D "TRIGA - ICS Reactor Safety System Failure Analysis". This analysis looked at the probability of the Reactor Safety System failing to perform its intended function: no scram occurs during a scram situation. In order for this to occur there would need to be simultaneous failures of two or more components of the Reactor Safety System. This analysis demonstrates that the Probability of Failure of the new Reactor Safety System is $2X10^{-11}$ failures/hour, or a mean time between failures of $5X10^{6}$ years.

REACTOR OPERATIONAL INSTRUMENTATION SYSTEM DESCRIPTIONS

REACTOR OPERATIONAL CHANNELS

Multirange Linear Channel

The mulitrange linear channel is one of three channels included in the NM-1000.

The multirange linear channel reports reactor power from source level [$\sim 10^{-3}$ Wt (thermal watts)] to full steady state power (1 MWt). The output of a principle fission detector serves as the channel input. The channel consists of two circuit sections: the count rate circuit, and the campbelling circuit. At power levels less than 1 kilowatt(t) the count rate circuit is utilized. The count rate circuit generates an output voltage proportional to the number of neutron genrated pulses or counts received from the fission detector. Hence, the output is proportional to the neutron population and the reactor power level. For steady state power levels at or above 1 kilowatt(t) the campbelling circuit is utilized. The campbelling circuit generates an output voltage proportional to the reactor power level by a verified technique of noise envelope amplitude detection and measurement known as campbelling. The NM-1000's micro-processor converts the signal from these circuits into 10 linear power ranges. This feature provides for a more precise reading of linear power level over the entire range of reactor power.

The NM-1000's multirange linear channel output is displayed in two formats. These are a bargraph indicator on the Reactor Control CRT display and a strip chart recorder located on the left-hand vertical panel on the control console. As a performance check, the microprocessor automatically tests the channel for campbell circuit operability while the reactor is operating in the count rate range and vice verse when the reactor is in the campbelling range. The multirange ranging function is auto-ranged via the NM-1000 control system computer.

Wide Range Log Channel

The wide-range log channel like the multirange linear measures reactor power from source level ($\sim 10^{-3}$ Wt) to full steady state power (1MWt). It is a digital version of the General Atomics 10-decade log power system to cover the reactor power range and provide a period signal. For the log power function, the chamber signal from startup (pulse counting) range through the campbelling [root mean square (RMS) signal processing] range covers in excess of 10-decades of power level. The self-contained microprocessor combines these signals and derives the power rate of change (period) through the full range of power.

The wide-range log channel forms part of the rod withdrawal prevent (RWP) interlock system. The channel activates variable set point bistable trips in the rod withdrawal prevent interlock system if source level neutrons ($\sim 10^{-3}$ Wt) are not present, if the reactor power level is above 1 KWt when switched to pulse mode, if a steady state power increase has a period of 3 seconds or faster during certain steady state modes, or if high voltage is not supplied to the fission detector.

The wide-range log and period output are displayed on bargraph indicators which are both hardwired and on the Reactor Control CRT. The NM-1000's microprocessor, similar to the multirange linear channel, automatically tests the wide-range log channel for upper and lower decade operability.

REACTOR INTERLOCKS (ROD WITHDRAWAL PREVENTS)

A Rod Withdrawal Prevent (RWP) interlock stops any upward motion of the standard control rods and prevents air from being supplied to the transient control rod unless specified operating conditions are met. An RWP interlock, however, does not prevent a control rod from being lowered or scrammed. Therefore, any RWP interlock prevents any further positive reactivity from being inserted into the core until specific conditions are satisfied.

The system of RWP interlocks prevents control rod withdrawals under the following circumstances:

- RWP prevents air from being applied to the transient rod unless the reactor power level is under 1 KWt.

- RWP prevents any control rod withdrawal unless, as a minimum, source level neutrons (~10^{-3} Wt) are present.

- RWP prevents any further control rod withdrawal unless the power level is changing on a 3-second or longer period as measured by the wide-range log channel during certain steady state operations.

- RWP prevents any control rod withdrawal unless high voltage is being supplied to the fission detector for the multirange linear and wide-range log channels.

- RWP prevents any control rod withdrawal unless the bulk pool water temperature is less than 60°C (Technical Specification Limit).

SERVO CONTROLLER

The Servo Controller, in the Automatic and Square Wave Modes, controls the reactor power automatically to within +/-1% of the demand power level selected by the operator. Thumbwheel switches are provided on the Mode Control panel for the desired power selection. The Servo Controller will track and stabilize reactor power through the utilization of a PID algorithm (Proportional, Integral, Derivative). The console will be capable of servoing any combination of the three standard control rods (REG, SAFE, or SHIM). It will not, however, servo the Transient Rod in any mode. The operator will be able to select which combination of rods will be servoed via a Servoed Rod Selector Switch located on the Mode Control Panel of the new control console. The Servo Controller system utilizes the latest digital computer technology coupled with extensively developed software. The current console uses an analog computer to servo the rods while the new console uses a digital computer to servo the rods.

Reactor flux level and change is accurately and rapidly measured by an analog/digital input from the Operational (fission) Channel. The PID algorithm in the DAC then responds to this input as compared to the operator set Demand Power Level Setting through the servoed control rods which are powered by precise translator/stepping motor drives. The (operator selected) drive(s) will be driven up or down automatically to control the power level to within +/-1% of the Demand Power Level Setting.

The new console Servo Controller can drive all three standard control rods simultaneously (~$5.50) in the Automatic and Square Wave Modes versus the old console which can servo the Transient and the REG rods (~$5.50) simultaneously in the Square Wave Mode and which servoed the REG rod in the Automatic Mode; by technical specifications the maximum excess reactivity above cold critical is $5.00. A Ramp Accident Analysis was performed to insure that a runaway drive situation involving a two second full-insertion (this is faster than the maximum drive rate of the new drives) of all three standard control rod drives would not lead to an event. This analysis was performed by General Atomics under contract to AFRRI and is enclosed as Appendix E "Analysis of a Five Dollar Ramp Insertion Over a Two Second Interval in AFRRI TRIGA Reactor". This analysis demonstrates that the consequences of this accident scenario are trivial. The peak power level attained is 330MW and the maximum fuel temperature attained is 330°C. The AFRRI TRIGA Reactor routinely pulses to peak powers of up to 3300MW and the normal 1 MW steady state fuel temperature is approximately 420°C. This analysis demonstrates that there are no unreviewed safety questions.

ROD DRIVES

The rod drive mechanisms for each of the new Standard Control Rod Drives is an electric stepping-motor-actuated linear drive equipped with a magnetic coupler and a positive feedback potentiometer. The purpose of each of the rod drive mechanisms is to position the reactor control rod elements.

General Operational Description

A stepping motor drives a pinion gear and a 10-turn potentiometer via a chain and pulley gear mechanism. The potentiometer is used to provide rod position information. The pinion gear engages a rack attached to the magnet draw tube. An electromagnet, attached to the lower end of the draw tube, engages an iron armature. The armature is screwed and pinned into the upper end of a connecting rod that terminates at its lower end in the control rod.

When the stepping motor is energized (via the rod control UP/DOWN switch on the operator's console), the pinion gear shaft rotates, thus raising the magnet draw tube. If the electromagnet is energized, the armature and the connecting rod will raise with the draw tube so that the control rod is withdrawn from the reactor core. In the event of a reactor scram, the magnet is de-energized and the armature will be released. The connecting rod, the piston, and the control rod will then drop, thus reinserting the control rod into the core.

Stepping motors operate on phase-switched dc power. The motor shaft advances 200 steps per revolution (1.8 deg per step). Since current is maintained on the motor windings when the motor is not being stepped, a high holding torque is maintained.

The torque vs speed characteristic of a stepping motor is greatly dependent on the drive circuit used to step the motor. To optimize the torque characteristic vs motor frame size, a Translator Module was selected to drive the stepping motor. This combination of stepping motor and translator module produces the optimum torque at the operating speeds of the control rod drives.

REACTOR MODES OF OPERATION

There are four standard operating modes: manual, automatic, square wave, and pulse.

The manual and automatic modes apply to the steady-state reactor condition; the square-wave and pulse modes are the conditions implied by their names and require a transient (pulse) rod drive.

The manual and automatic reactor control modes are used for reactor operation from source level to 100% power. These two modes are used for manual reactor start up, change in power level, and steady-state operation. The square-wave operation allows the power level to be raised quickly to a desired power level. The pulse mode generates high-power levels for very short periods of time.

Manual rod control is accomplished through the use of push-buttons on the rod control panel. The top row of push-buttons (magnet) is used to interrupt the current to the rod drive magnets. If the rod is scrammed and the drive is above the down limit, the rod will fall back into the core and the magnet will automatically drive to the down limit, where it again contacts the armature.

The middle row of push-buttons (up) and the bottom row (down) are used to position the control rods. Depressing these push-buttons causes the control rods to move in the direction indicated. Several interlocks prevent the movement of the rods in the up direction under conditions such as the following:

1. Scrams not reset.
2. Magnet not coupled to armature.
3. Source level below minimum count.
4. Two UP switches depressed at the same time.
5. Mode switch in the pulse position.
6. Mode switch in automatic position (servoed rods only).
7. Period less than 3 seconds.

There is no interlock inhibiting the DOWN direction of the control rods except in the case of the servoed rods while in the AUTOMATIC mode. In all cases, however, the manual scram of any rod will result in the full insertion of the rod into the core.

Automatic (servo) power control can be obtained by switching from manual operation to automatic operation via operator activation of the Auto Mode Switch on the control console's Mode Control Panel. All the instrumentation, safety, and interlock circuitry described above applies and is in operation in this mode. However, the selected servoed rods are now controlled automatically in response to a power level and period signal. The reactor power level is compared with the demand level set by the operator and is used to bring the reactor power to the demand level on a fixed preset period. The purpose of this feature is to automatically maintain the preset power level during long-term power runs. Options are available to the operator to maintain power by movement of a single rod or by bank operation of selected rods. The rods to be servoed are selected by the operator via the Servoed Rod Selector Switch on the control console's Mode Control Panel.

In a square-wave operation, the reactor is first brought to a critical condition below 1 KW, leaving the transient rod partially in the core. All of the steady-state instrumentation is in operation. The transient rod is ejected from the core by means of the transient rod FIRE push-button. When the power level reaches the demand level, it is maintained in the same manner as in the automatic mode.

Reactor control in the pulsing mode consists of establishing criticality at a flux level below 1 KW in the steady-state mode. This is accomplished by the use of the motor-driven control rods, leaving the transient rod either fully or partially inserted. The mode selector switch is then depressed. The Transient Rod Fire switch automatically connects the pulsing chamber to monitor and record peak flux (nv) and energy release (nvt). Pulsing can be initiated from either the critical or subcritical reactor state.

COMPARISON OF THE CURRENT AND THE NEW REACTOR SAFETY AND CONTROL SYSTEMS

REACTOR SAFETY SYSTEMS

The current console, which was designed and built in the early 1970's, has as its Reactor Safety Systems (See Table I) two hardwired independent analog High Flux Safety Channels, two hardwired independent analog Fuel Temperature Safety Channels, and a hardwired relay logic SCRAM circuitry. The High Flux safety Channels derive their signals from two Boron (neutron sensitive) Ion Chambers mounted above the core, and these channels have readouts located on the vertical panel of the control console in the form of analog meters. The Fuel Temperature Safety Channels derive their signals from two instrumented fuel elements, one located in the B-ring and one located in the C-ring. The Fuel Temperature Safety Channels also have readouts located on the vertical panel of the control console in the form of analog meters. The Scram circuitry has two independent relay contacts for each safety channel, one located in the supply side and one located in the return side of the magnet and solenoid power circuitry. Dropping any one of these numerous relays would cut power to the magnets and the air solenoid.

The new console, as with the old console, also has as its Reactor Safety Systems two independent hardwired analog High Flux Safety Channels, two independent hardwired analog Fuel Temperature Safety Channels, and a hardwired relay logic SCRAM circuitry. The High Flux Safety Channels, just like the old console, derive their signals from two Ion Chambers mounted above the core and have readouts located on the vertical panel of the control console. However, for the new console, these readouts take the form of LED bargraphs instead of meters. These new channels were designed to be the same as the old channels, only updated with current technology electronics. The Fuel Temperature Safety Channels will still derive their signals from the same two instrumented fuel elements located in the B-ring and in the C-ring. As with the High Flux Channels, the Fuel Temperature Channels have their readouts on the control console in the form of LED bargraphs instead of meters. It should be emphasized again, that these safety systems on the new consoles are independent hardwired analog channels just as those are on the old console. These systems are completely independent of the system's computers and will continue to function irregardless of the state these computers are in. This will insure safety system monitoring and control at all times. The Scram circuitry, again as with the old console, has two independent relays for each safety channel, one located in the supply side and one located in the return side of the magnet and solenoid power circuitry. Similar to the four safety channels, the Scram circuitry was designed to be the same as the old Scram circuitry only replaced with current technology electronics. Table 1 shows a comparison between the SCRAMS on the new and old consoles. The SCRAM circuitry on both systems is the same except for the Safety Channel Calibrate Scram on the old console and the Watchdog Scrams on the new console. The old console used to shunt the inputs to the safety channels while putting in calibration signals to the safety channels. This created the possibility of operating with a safety channel in the calibrate mode. To prevent this condition from occurring the old console had a relay which would scram the reactor if any of the safety channels were switched to the calibrate mode. In the new system, the calibration signals are additive to the normal safety channel signals (e.g. the safety channels are not shunted in the calibration mode). A calibration signal added to the normal safety channel signal is more conservative (will always provide a higher channel reading) and therefore does not require a calibrate scram. However, watchdog scrams, as described earlier, have been added to the new console scram circuitry. These watchdogs monitor the status of the DAC and CSC computers and should any of the four watchdogs (two in the DAC and two in the CSC) fail to be reset by the software, then the system would scram the reactor. This ensures that failure of either of these computers or of their software will cause a system scram.

REACTOR OPERATIONAL CONTROL AND MONITORING SYSTEMS

The 1972 console has an operational channel which derives its signal from a fission chamber and generates the Wide-Range Log and Multirange Linear monitoring channels. The operational channel combines the standard techniques of Count Rate and Campbelling in an analog computer to provide the capability to monitor 10 decades of power. The new console uses an

ANALOG (1972) vs DIGITAL (1988) CONTROL CONSOLES

OLD

SAFETY SYSTEMS	INTERLOCKS	CONTROL (OPS CHANNEL)	DRIVES
Hardwired Amp-BT circuit	Relay Logic	Analog Computer	Phase Interrupt

NEW

SAFETY SYSTEMS	INTERLOCKS	CONTROL (OPS CHANNEL)	DRIVES
Hardwired Amp-BT circuit	Firmware NM-1000 Relays & EPROM	Digital Computer	Stepping Motor (Digital)

CONSOLE REACTOR SAFETY SYSTEM COMPARISON

OLD	NEW
SAFETY CHANNELS	**SAFETY CHANNELS**
- 2 Percent Power	- 2 Percent Power
- 2 Fuel Temperature	- 2 Fuel Temperature
SCRAMS	**SCRAMS**
-TECH SPEC	-TECH SPEC
- 4 High Level Safety Trips	- 4 High Level Safety Trips
- Manual	- Manual
- 2 HV Loss % Power	- 2 HV Loss % Power
- Pulse Timer	- Pulse Timer
- Emergency Stop	- Emergency Stop
- Water Level	- Water Level
-SAR	-SAR
- Key Switch	- Key Switch
- Steady State Timer	- Steady State Timer
- Loss of AC	- Loss of AC
- Facility Interlocks	- Facility Interlocks
* Safety Channel Calibrate	* Watchdog - 2 Relays in both the DAC and the CSC
- Individual Rod SCRAM	- Individual Rod SCRAM

CONSOLE INTERLOCKS COMPARISON

OLD	NEW
-TECH SPEC	-TECH SPEC
- 1 kw	- 1 kw
- Source Level Neutrons	- Source Level Neutrons
- Mode I (no two rods)	- Mode I (no two rods)
- Mode III (no rod except TRANS)	- Mode III (no rod except TRANS)
-SAR	-SAR
- 3 second period	- 3 second period
- Ops Channel HV loss	- Ops Channel HV loss
- Bulk Water 60 C	- Bulk Water 60 C
* Ops Channel Calibrate	* (calibrate signal additive)

operational channel which was designed to be a digital version of the old system; it still combines the standard techniques of Count Rate and Campbelling to provide the capability to monitor 10 decades of power. The difference is that this function is now performed with a digital computer instead of a analog computer and uses current technology electronics. These two systems were demonstrated to be essentially equivalent during the manufactures test program when both the old and the new systems were operated in parallel.

The interlocks or Rod Withdrawal Prevents (RWPs) for both the new and old systems are shown in Table 2. Again, these interlocks are the same for both systems except for the Operational Channel Calibrate RWP on the old console. On the old console, the input signal to the operational channel would be shunted when the channel was placed in the calibrate mode. In order to prevent operation of the reactor in this configuration, an RWP was added to the system to prevent rod withdrawal with the operational channel in the calibrate mode. On the new console, the calibration signal is additive to the normal operational signal, and again is therefore more conservative and requires no RWP. The interlocks on the old console were all analog logic using relays. The interlocks on the new console use Digital Logic (Firmware).

STANDARD CONTROL ROD DRIVES

The three standard control rod drives will be replaced. The old drives used phase-interrupt (analog) motors while the new drives will use stepping (digital) motors (See Table 3). Only the drive motors are being changed, the remainder of the control rod drive assemblies will stay the same.

SAFETY EVALUATION CONCLUSION

The AFRRI TRIGA Reactor, NRC Facility License No. R-84, is classified as a "Negligible Risk Research Reactor (Pulsing)" in accordance with the NRC approved AFRRI TRIGA Reactor Facility Safety Analysis and as defined in ANSI/ANS 15.15-1978 "Criteria for the Reactor Safety Systems of Research Reactors". A "Negligible Risk Research Reactor (Pulsing)", as defined in ANSI/ANS 15.15-1978, is "a research reactor for which, in the postulated event of the complete failure of the reactor safety system coincident with the occurrence of the most adverse Design Basis Event, the radiological consequences would be negligible." Pulsing is defined as "a reactor that has been specially designed with an inherent shutdown mechanism sufficient to allow the reactor to accept large reactivity insertions without exceeding any safety limit."

In analyzing the safety of the AFRRI TRIGA Reactor, it is important to start with the inherent safety of the TRIGA Fuel, which is designed to operate with large positive step reactivity insertions. The inherent safety of the fuel element stems from its large prompt negative temperature coefficient of reactivity, which causes the automatic termination of a power excursion before any core damage results. The Prompt Negative Coefficient of Reactivity of the AFRRI TRIGA Reactor is - 0.0126 %deltaK/K per °C (-1.7cents/°C), while the Steady State Negative Coefficient of Reactivity is - 0.0051 %deltaK/K per °C (-.7cents/°C). Fuel elements with 8.45 wt.%U have been pulsed repeatedly in General Atomics' Advanced TRIGA Prototype Reactor (ATPR) to peak power levels of over 8,000 MW, and have been pulsed thousands of times to peak power levels greater than 2,000 MW. The AFRRI TRIGA Reactor is limited to a $4.00 step positive reactivity insertion (technical specification limit) which would yield a peak power level of approximately 4,700 MW.

The AFRRI Facility Safety Analysis Report has analyzed two Design Basis Accidents. The first Design Basis Accident, called the "Fuel Element Drop Accident," involved the postulated occurrence of a cladding failure of a fuel element after a 2-week period where the saturated fission product inventory of a 1 MW steady state operation has been allowed to decay after being taken out of the operating core and placed in storage; the saturated fission product inventory is obtained after 100 hours of continuous reactor operation at full power (1 MW). The cladding failure could occur when the fuel element is withdrawn from the reactor pool. While the fuel element is exposed to air, a cladding failure could occur coincidentally, or due to a drop. As the AFRRI FSAR explains, the probability of such an accident is considered to be extremely remote. The second Design Basis Accident, called the Fuel Element Cladding Failure Accident, involved the postulated occurrence of a cladding failure of a fuel element during a pulse operation or inadvertent transient following a steady state operation of 1 MW. Again, it was assumed a saturated fission product inventory which occurs after 100 hours of continuous reactor operation at full power (1 MW), and a pulse operation with an integrated energy of 40 MW-sec. A 40 MW-sec pulse operation is roughly equivalent to a step positive reactivity insertion of approximately $4.50. The maximum worth of the AFRRI TRIGA Pulse Rod (Transient Rod) is approximately $3.75, and as such a 40 MW-sec pulse operation is an extremely conservative assumption. The AFRRI FSAR again explains that the probability of such an accident is considered to be extremely remote.

The analysis in the AFRRI FSAR shows that "... the consequences from the Design Basis Accident of a fuel element drop accident or a fuel element clad failure accident were insignificant." Therefore, it was "... concluded that the operation of the AFRRI reactor in the manner authorized by Facility License No. R-84 does not represent an undue risk to the health and safety of the operational personnel or the general public."

Both of these Design Basis Accidents (DBAs) were postulated on the occurrence of one or two predetermined, deliberate man-made events. In the first DBA, the scenario required that the reactor be operated continuously for 100 hours at full power to build up a saturated fission product inventory. In the second DBA, the scenario again requires a saturated fission product inventory followed by a step positive insertion of reactivity that produces 40 MW-sec of integrated energy. AFRRI has never operated at full power for 100 hours continuously, nor will probably ever operate in this manner under normal operating conditions. Both of these DBAs require fuel cladding failures following a set of specific man-made conditions and are not a result of any failures on the part of the Reactor Safety Systems. It was shown previously that the new console has a MTBF of the Reactor Safety System of 5 X 10^6 years. Failure of the Reactor Safety System would not initiate a Design Basis Accident. Even should the Reactor Safety System suffer a complete failure at the same moment as a DBA, the consequences would be negligible.

It was determined during the design of the new Reactor Instrumentation and Control System that no technical specification changes would be required. There are no technical specification changes associated with the installation or operation of AFFRI's new Reactor Instrumentation and Control System.

The new Reactor Instrumentation and Control System will offer a dramatic improvement in operational productivity, system reliability, and system maintainability.

The new Digital Reactor Instrumentation and Control System has been designed to be safer than the present AFRRI control system. This has been accomplished by continuing to hardwire all safety circuits in a redundant, fail safe configuration. These safety circuits are completely independent of the data acquisition computer (DAC) and the control system computer (CSC). This means that if either or both computers were to fail, the failure cannot prevent the reactor from scramming. On the other hand, critical functions of the computers are monitored by "watch-dog-timers". If the computers fail to update the timers in a predetermined fashion, the redundant, hardwired watch-dog-timers will scram the reactor. As a result, the new Digital Reactor Instrumentation and Control System has equal or greater safety built-in than the present AFRRI control system, which has SAR approval.

Based on the analyses in this technical report, it has been determined that the proposed changes to the Reactor Facility do not involve unreviewed safety questions and, in fact, are improvements in the facility design at AFRRI.

This technical report describes changes and modifications made to the AFRRI reactor facility as depicted in the facility's SAR. These changes have been reviewed by the Reactor Facility Director and found to contain no unreviewed safety questions. This report is submitted to the Reactor and Radiation Facility Safety Committee (RRFSC) for their concurrence that conditions of 10 CFR 50.59 are met. These conditions are that no unreviewed safety questions are present and that the changes made do not increase the probability of occurrence or the consequences of an accident or malfunction.

APPENDIX A

Listing of Corrections to be made to the SAR

Page	Section	Change
4-16	4.10	This change will clarify the difference in the type of drive used for the standard and transient rods.
4-16,17	4.10.2	The paragraph is modified to reflect the new stepping motors used in the control rod drives.
4-16b	Figure 4-8	The figure has been updated to depict the new control rod drives on the standard control rods.
4-22	Section 4.11	The phrase "three ion chambers" has been changed to "two ion chambers and a pulse detector" to allow a Cherenkov detector or an ion chamber to be used for pulse operations.
4-22	Section 4.11	A paragraph describing the NM-1000 has been added to the SAR.
4-22	Section 4.11.1	The section describing the Multirange Linear Channel has been updated to reflect changes incurred by the new console.
4-23	Section 4.11.2	The section describing the Wide-Range Log Channel has been updated to reflect changes incurred by the new console.
4-24	Section 4.11.3	Portions of the section describing High Flux Safety Channels One and Two have been modified to reflect changes incurred by the new console.

Page	Section	Change
4-29	Section 4.11.4	Portions of the section describing Fuel Temperature Safety Channels have been modified to reflect changes incurred by the new console.
4-27	Section 4.12	The RWP associated with the wide-range log channel in any mode other than OPERATE is no longer required. See 10 CFR 50.59 writeup.
4-27	Section 4.12	The SCRAM associated with any of the safety channels in any position other than OPERATE is no longer required. See 10 CFR 50.59 writeup.

APPENDIX B

Specific SAR word changes for the previously discussed Facility Modification Safety Analyses

1. <u>REACTOR CONTROL COMPONENTS</u> (Section 4.10)

<u>CURRENT SAR WORDING:</u>
"Control rod movement within the core is accomplished using rack and pinion electromechanical drive for the transient control rod."

<u>PROPOSED SAR WORDING:</u>
"Control rod movement within the core is accomplished using rack and pinion electromechanical drives for the standard control rods, and pneumatic-electromechanical drive for the transient control rod."

2. <u>STANDARD CONTROL ROD DRIVES</u> (Section 4.10.2)

a. <u>CURRENT SAR FIGURE:</u>
Figure 4-8

<u>PROPOSED SAR FIGURE:</u>
Figure 4-8 (modified to reflect new control rod drives)

b. <u>CURRENT SAR WORDING:</u>
"The standard drive consists of a two-phase motor, a magnetic coupler, a rack and pinion gear system, and a potentiometer used to provide an indication of rod position, which is displayed on the reactor console."

<u>PROPOSED SAR WORDING:</u>
"The standard drive consists of a stepping motor, a magnetic coupler, a rack and pinion gear system, and a potentiometer used to provide an indication of rod position, which is displayed on the reactor console CRT."

c. <u>CURRENT SAR WORDING:</u>
"Clockwise rotation of the motor shaft raises the draw tube assembly."

<u>PROPOSED SAR WORDING:</u>
"When the stepping motor is energized, the pinion gear shaft rotates, thus raising the magnet draw tube."

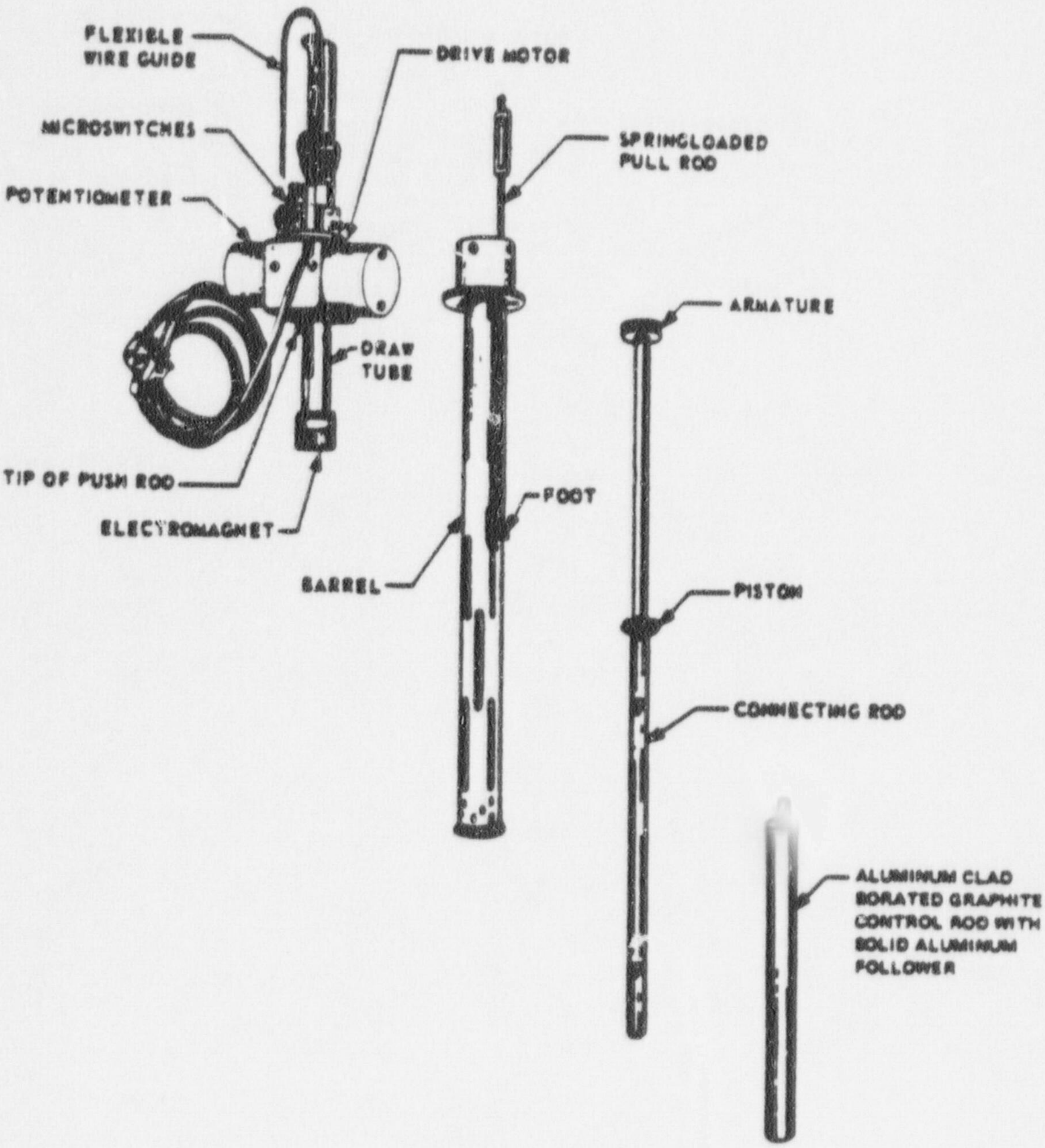


FIGURE 4-8
STANDARD CONTROL ROD DRIVE
FOR SAFETY AND SHIM RODS*

*Regulating Rod Drive Similar Except Drive Motor Contains Tachometer

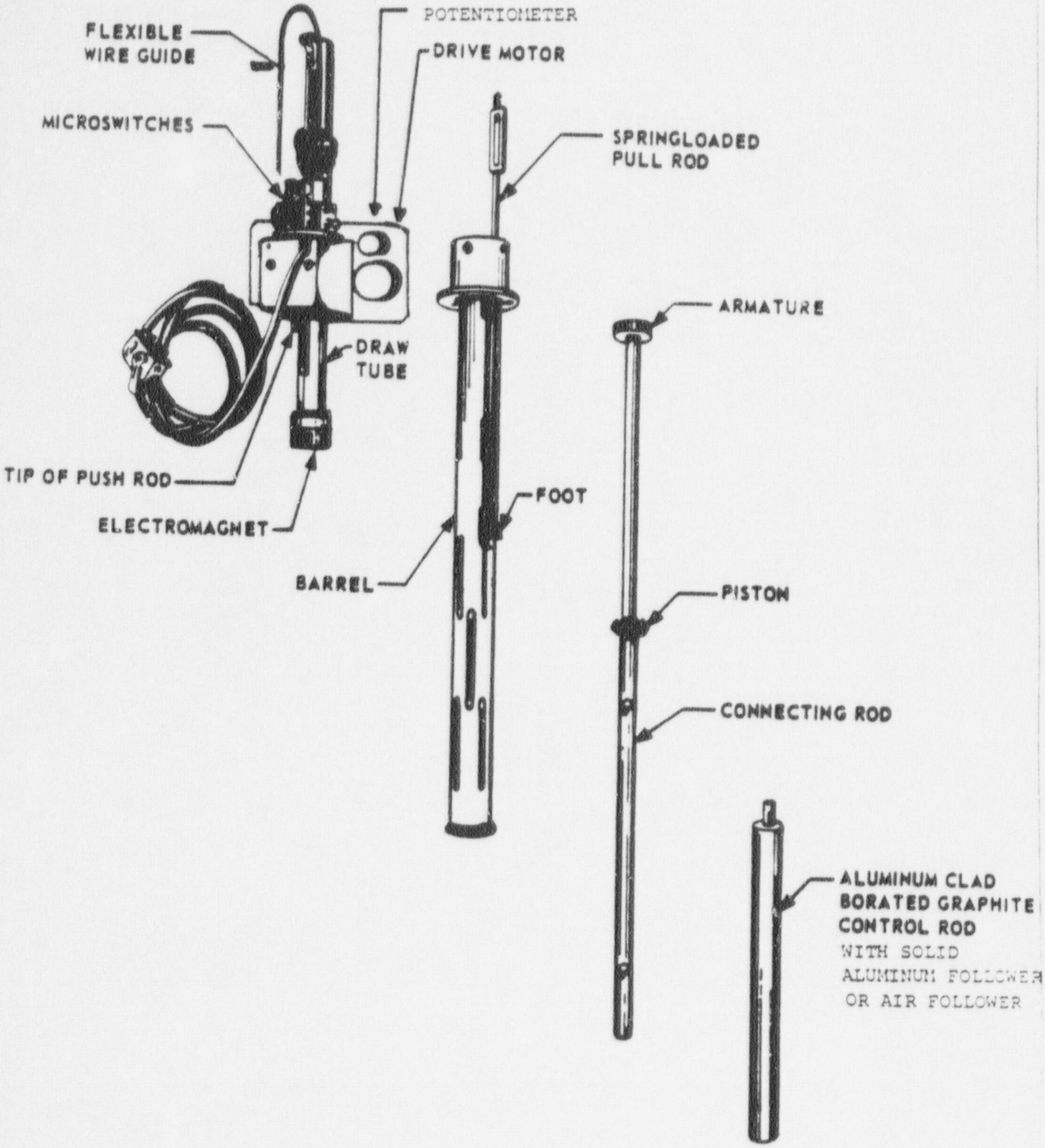


FIGURE 4-8
Standard Control Rod Drives

3. REACTOR INSTRUMENTATION (Section 4.11)

CURRENT SAR WORDING:
"A fission detector and three ion chambers comprise the remaining detectors."

PROPOSED SAR WORDING:
"A fission detector, two ion chambers, and a pulse detector comprise the remaining detectors."

4. NM-1000

ADD TO THE SAR: (at Section 4.11)
"The NM-1000 system, which includes the Multirange Linear Channel and the Wide-Range Log Channel, is contained in two National Electrical Manufactures Association (NEMA) enclosures, one for the amplifier and one for the processor assemblies. The amplifier assembly contains modular plug-in subassemblies for pulse preamplifier electronics, bandpass filter and RMS electronics, signal conditioning circuits, low voltage power supplies, detector high-voltage power supply, and digital diagnostics and communication electronics. The processor assembly is made up of modular plug-in subassemblies for communication electronics (between amplifier and processor), the microprocessor, a control/display module, low-voltage power supplies, isolated 4 to 20 mA outputs, and isolated alarm outputs. Communication between the amplifier and processor assemblies is via two twisted-shielded-pair cables."

5. MULTIRANGE LINEAR CHANNEL (Section 4.11.1)

CURRENT SAR WORDING:
"The multirange linear channel reports reactor power from source level ($\sim 10^{-3}$ thermal watts) to full steady state power (1 MWt). The output of the fission detector, fed through a preamplifier, serves as the channel input. The multirange linear channel consists of two circuits: the count rate circuit, and the campbelling circuit. For power levels less than 1 kilowatt(t), as selected on the power range select switch, the count rate circuit is utilized. The count rate circuit generates an output voltage proportional to the number of pulses or counts received from the fission detector. Hence, the output is proportional to the neutron population and the reactor power level. For steady state power levels at or above 1

kilowatt(t), as selected on the power range select switch, the campbelling circuit is utilized. The campbelling circuit generates an output voltage proportional to the reactor power level by a verified technique of noise envelope amplitude detection and measurement known as campbelling. The output from the appropriate circuit is fed to an amplifier which supplies a signal to the strip chart recorder located on the reactor console. The power level is scaled on the strip chart recorder between 0 and 100 percent of the power indicated by the power range select switch on the console. The strip chart records this output for all steady state modes of operation but not during pulse operation.

PROPOSED SAR WORDING:

"The multirange linear channel reports reactor power from source level (~10^{-3} thermal watts) to full steady state power (1 MWt). The output of the fission detector, fed through a preamplifier, serves as the channel input. The multirange linear channel consists of two circuits: the count rate circuit, and the campbelling circuit. For power levels less than 1 kilowatt(t), the count rate circuit is utilized. The count rate circuit generates an output voltage proportional to the number of pulses or counts received from the fission detector. Hence, the output is proportional to the neutron population and the reactor power level. For steady state power levels at or above 1 kilowatt(t), the campbelling circuit is utilized. The campbelling circuit generates an output voltage proportional to the reactor power level by a verified technique of noise envelope amplitude detection and measurement known as campbelling. The NM-1000's microprocessor converts the signal from these circuits into 10 linear power ranges. The multirange linear channel output is displayed in two formats. These are a bargraph indicator on the Reactor Control CRT display and a strip chart recorder located on the left-hand vertical panel on the control console. The power level as displayed on the CRT bargraph and the strip chart recorder is scaled between 0 and 100 percent for each of the 10 linear power ranges. The multirange function is auto-ranged via the NM-1000 control system computer. The multirange linear output on the CRT bargraph is displayed for all steady state modes of operation, but not during pulse operation.

6. WIDE-RANGE LOG CHANNEL (Section 4.11.2)

CURRENT SAR WORDING:
"The outputs of these two circuits are log amplified and then summed in a summing amplifier. The summing amplifier supplies a signal to the strip chart recorder located on the reactor console. The power level is indicated on a 10 decade log scale (10^{-3} watts(t) to 1 MW(t)). The strip chart records this output for all steady state modes of operation but not during pulse operation.

During certain steady state modes, the wide-range log channel also measures the rate of change of the power level, which is displayed on the period/log meter located on the reactor console."

PROPOSED SAR WORDING
"The outputs of these two circuits are digitally combined and processed to provide the power rate of change (period) and the power level indicated on a 10 decade log scale (10^{-3} watts(t) to 1 MW(t)). The wide-range log and period outputs are both displayed on bargraph indicators on the Reactor Control CRT and on hardwired vertical LED bargraphs on the left-hand side of the Reactor Control Console. The outputs on the CRT bargraphs are displayed for all steady state modes of operation but not during pulse operation."

7. HIGH FLUX SAFETY CHANNELS ONE AND TWO (Section 4.11.3)

a. CURRENT SAR WORDING:
"High flux safety channels one and two report the reactor power level as measured by three ion chambers placed above the core in the neutron field."

PROPOSED SAR WORDING:
"High flux safety channels one and two report the reactor power level as measured by two ion chambers and a pulse detector placed above the core."

b. CURRENT SAR WORDING:
"The steady state power level, as measured by the two high flux safety channels, is displayed on two separate meters located on the reactor console."

PROPOSED SAR WORDING:
"The steady state power level, as measured by the two high flux safety channels, is displayed on two separate bargraphs located on the reactor console."

c. CURRENT SAR WORDING:
"During pulse operation, high flux safety channel one is shunted and the sensor for high flux safety channel two is switched to a third, independent ion chamber placed above the core."

PROPOSED SAR WORDING:
"During pulse operation, high flux safety channel one is shunted and the sensor for high flux safety channel two is switched to a third, independent pulse detector placed above the core."

d. CURRENT SAR WORDING:
"The NV channel output is displayed on the strip chart recorder located on the reactor console. The NVT channel output is displayed on the reactor console NVT meter."

PROPOSED SAR WORDING:
"The NV and NVT channel outputs are displayed on two separate bargraph indicators located on the left-hand side of the console."

e. CURRENT SAR WORDING:
"Knobs for each channel, located on the reactor console, allow the channels to be checked for calibration. Switching these knobs to any mode from operate (i.e., to the zero or calibrate positions) causes an immediate reactor scram."

PROPOSED SAR WORDING:
"Calibration of each safety channel is done automatically when the operator initiates the "pre-checks" by activation of the Prestart Check Switch on the control console's Mode Control Panel. Any failures detected during the prechecks will be automatically reported to the operator via the reactor status CRT. This calibration can only be performed while the reactor is in the SCRAMMED mode."

f. <u>CURRENT SAR WORDING:</u>
"A trip test knob for each safety channel ..."

<u>PROPOSED SAR WORDING:</u>
"A trip test switch for each safety channel ..."

8. <u>FUEL TEMPERATURE SAFETY CHANNELS</u> (Section 4.11.4)

a. <u>CURRENT SAR WORDING:</u>
"The two fuel temperature signals are amplified and displayed on two separate meters located on the reactor console. During pulse operation, the output of fuel temperature safety channel one is also recorded on the reactor console strip chart recorder."

<u>PROPOSED SAR WORDING:</u>
"The two fuel temperature signals are amplified and displayed on two separate bargraphs indicators located on the reactor console and on the reactor control CRT."

b. <u>CURRENT SAR WORDING:</u>
"A trip test knob for each fuel temperature safety channel, located on the reactor console, provides a means of testing the scram capability of each channel without having to actually reach or exceed the technical specifications limit on allowable fuel temperatures."

<u>PROPOSED SAR WORDING:</u>
"Calibration of the Fuel Temperature Channels is done automatically when the reactor operator initiates the "pre-checks" by activation of the Prestart Check Switch on the control console's Mode Control Panel. Any failures detected during the prechecks will be automatically reported to the operator via the reactor status CRT."

9. <u>ROD WITHDRAWAL PREVENT (RWP) INTERLOCKS</u> (Section 4.12)

<u>CURRENT SAR WORDING:</u>
"RWP prevents any control rod withdrawal if the wide range log channel is in any mode (i.e. position) other than OPERATE."

<u>PROPOSED SAR WORDING:</u>
-This requirement is deleted (See document for analysis).

10. SCRAM LOGIC CIRCUITRY (Section 4.14)

CURRENT SAR WORDING:
"Any of the safety channels (fuel temperature safety channels and high flux safety channels) in any position other than OPERATE (i.e., CALIBRATE or ZERO) causes a reactor scram."

PROPOSED SAR WORDING:
-This requirement is deleted (See document for analysis).

APPENDIX C

AFRRI TRIGA Console (Safety) Scram System Single Failure Criteria Analysis

AFRRI TRIGA Console (Safety) Scram System
Single Failure Criteria Analysis

REFERENCES:

1. IEEE 279-1971 Criteria for Protection Systems for Nuclear Power Generating Stations.

2. IEEE 379-1977 Application of the Single-Failure Criteria to Nuclear Power Generating Station Class IE Systems.

The following analysis is postulated upon the principle [explained in Reference 2, Section 6.1(4)] that redundancy of protection devices provides complete assurance of safety in operation with regard to the parameter monitored by the device. For example, the failure of a fuse to blow when subjected to its designed rating of overload current is a credible possibility, but the failure of two identical fuses in series to blow simultaneously is not a credible possibility.

1. The steady steady-state timer scrams the reactor after an elapsed time and no redundancy is provided. The probability of the failure of this device is estimated as follows:

   Mean Time Between Failure (MTBF) of the electronic circuitry is about 200,000 hours based upon parts count and stress factor per MIL-HDBK-217B. At 200 hours per month this is one failure in 83 years.

   The electronic timing circuits operate relay contacts whose failure rate is expressed in operation cycles rather than MTBF. A conservative estimate based on manufacturers specifications is 25,000 operating cycles. At two cycles per day and 5 days per week, this is one failure in 48 years. The most likely failure is increased contact resistance rather than welded contacts so that an unsafe condition probably is not credible in less than 100 years of operation. The steady state timer is not a required safety system component.

2. The pulse timer scrams the reactor after completion of a power pulse and no redundancy is provided. The rated life of this device is 250,000 electrical operations which exceeds the probable number of pulses to be produced.

   The probability of random failure calculated as MTBF per MIL-HDBK-217B based upon parts count and stress factor is greater than 300,000 hours. At 200 hours per month, this is equivalent to one failure in 125 years.

3. The manual scram button is used to shut down the reactor manually. The specified life is 100,000 cycles of operation. At 15 manual scrams per day this would be one failure in 25.6 years. However, this is a normally closed switch with a direct acting operator. The most likely failure mode is a broken switch structure which would result in failure to reset after a scram. Welded contacts would be separated by mechanical force of the direct action operator. Redundancy for a manual scram exists in the console operator key switch and power on switch.

4. The console key switch de-energizes the magnet supply as well as other circuitry. The estimated life is 10,000 operations. At 15 operations per day, this is a failure rate of one every 2.6 years. However, the key switch is not depended upon to perform a safety function except to prevent unauthorized startup. The manual scram button provides shutdown redundancy so that an unsafe failure is not credible.

5. All reactor tank shielding door interlock switches and emergency stop buttons remain from the existing system and are unaffected by the new hardware. The emergency stop switch and all other switches on the new console use the same actuator and switching element as are used on the existing system.

6. The loss of AC power causes the magnet supply to be de-energized which in turn produces the same response as a manual scram, dropped rods.

7. The high level trips in the two power safety channels are redundant and therefore do not present a credible mode for failure. All non-safety outputs are physically separated and isolated to prevent common mode failures which may otherwise invalidate the single failure criterion. A minimum separation of six inches, or a metallic flame barrier exists between all safety and non-safety circuits. A minimum isolation voltage of 1500 volts RMS or DC applies to both optical and transformer isolation.

   The MTBF of the two NP1000 safety modules is greater than 20,000 hours based upon component failure rate data taken from MIL-HDBK-217B. The bistable trip portion of the NP1000 has an MTBF greater than 200,000 hours. Because the NP1000's operate independently, each with its own detector from the existing system complete redundancy exists.

8. The detector high voltage is interlocked by trip circuits in the power and safety channels and the redundant circuitry makes unsafe failures not credible. Separation and isolation criteria of item 6 above apply.

9. The two fuel temperature safety channels are high reliability modular signal conditioner/limit alarm devices each with calculated MTBF figures exceeding 200,000 hours. The channels are redundant with separation criteria applied to the wire harness therefore an unsafe failure is not credible.

10. The magnet supply ground fault detector uses a high reliability modular signal conditioner/limit alarm. The signal conditioner module has an MTBF of greater than 200,000 hours. The limit alarm uses a relay rated for more than 25,000 operations. There is a pushbutton switch which is used to test the operability of the ground fault detector on a daily

basis. Because the relay only operates during testing and fault conditions the end of life cannot be reached. Therefore the probability of an undetected ground fault is the probability of random failure in the signal conditioner which is less than one in 23 years.

11. Pool Level Monitor - Pool water level is monitored with redundant float operated switches and redundant relays with contacts in the scram circuits.

    The switches and relays have failure rates of less than one in 10^6 hours but redundancy makes a water level monitor failure not a credible failure mode.

12. Watchdog Scrams - A watchdog timer on the data acquisition computer and another on the control system computer are required to be reset periodically by a program routine as a safeguard against computer component failures either in hardware or software. If the required response is not received within a definite time period, redundant normally open (fail safe) contacts interrupt the scram loop dropping the rods and shutting down the reactor. The watchdog timer is an additional safety device.

APPENDIX D

Scram Circuit Safety Analysis for the University of Texas TRIGA Reactor

COLLEGE OF ENGINEERING
THE UNIVERSITY OF TEXAS AT AUSTIN
Department of Mechanical Engineering · Nuclear Engineering Program · Austin, Texas 78712 · (512) 471-5136

April 22, 1988

Mr. Junaid Razvi
General Atomics
P.O. Box 85608, Ms/21
San Diego, CA 92138

Dear Junaid:

As per our discussion at the TRIGA meeting, I have enclosed a copy of the complete safety circuit evaluation we developed from the available GA information. I hope that this analysis might provide valuable support for your analysis of the new console installation. A review by knowledgeable persons should be made to ascertain that our understanding and evaluation of the documents is correct. I believe that although the system has evolved from some of the documentation we had available, the changes to the analysis are not likely to be significant. An effort was made in the method of presentation to demonstrate various conditions.

Please review and return comments. Other persons have also expressed an interest in the analysis but I'd prefer to have General Atomics comments to make available on final document.

Thank you for your help in this matter.

Sincerely,

Thomas L. Bauer

Thomas L. Bauer
Assistant Director
Nuclear Engineering
Teaching Laboratory

TLB:dlw
Enclosure

The University of Texas at Austin

Scram Circuit Safety Analysis for The University of Texas TRIGA Reactor

Prepared by:

Dr. Thomas Bauer
Professor of Mechanical Engineering

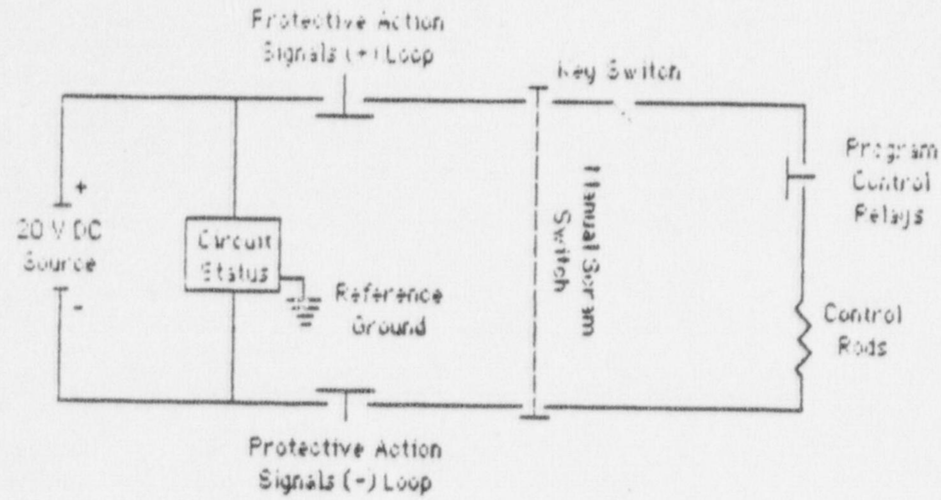
David Goff
Engineering Science Student

April 22, 1988

TRIGA-ICS
Reactor Safety System

Protective actions of the Reactor Safety System (RSS) are provided by several parameter measurement channels and a control-rod power circuit (scram circuit). Each measurement channel controls operation of the scram circuit by means of a relay in the circuit. When any one of these relays is tripped, it cuts power to the control rods.

The scram circuit design is comprised of four functional sections. These represent the protective action monitoring of the system, monitors of the system's operability, a software and manual section, including the key switch and manual scrams, and the physical circuit itself, including the ground fault and power supply monitors. These sections are shown in the diagram below.



RSS Functional Diagram

The following analysis first looks at the basics of the system in steady-state operation. After a general failure model is developed, the analysis expands to look at the calibration checks, the bypass relay used in pulse mode, and monitor channel failures outside the scram circuit itself.

RSS Failure Analysis

The RSS scram circuit supplies power to the control rods and hence is the point at which all scrams occur, or fail to occur. Its proper function is therefore imperative to safe operation of the reactor. In analyzing the scram circuit, as many potential failure modes as possible were examined to estimate the probability of a circuit failure. The ultimate failure consequence was that the control rods were not inserted and no scram occurred during a scram situation. In order to examine the way in which individual failures in the circuit might lead to a non-scram, a fault tree was constructed based on an analysis of the scram circuit.

The first step in the RSS failure analysis involved identifying the various ways in which the RSS could fail. These include:

1) Physical System Failure
2) Limiting Safety System Setting (LSSS) Failure
3) System Operable Failure
4) Computer/Manual Control Failure

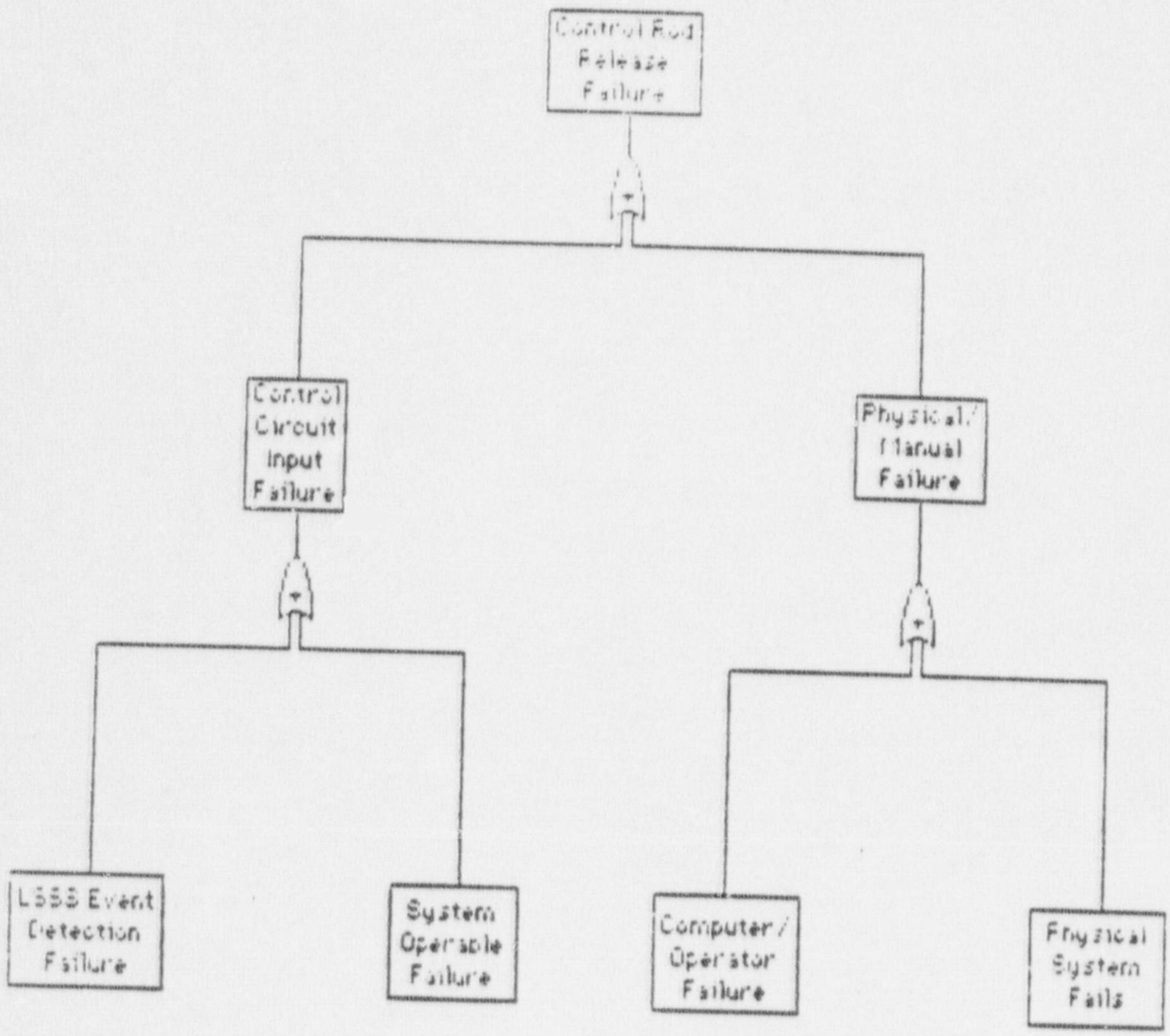
The Physical System failures include wire breaks, shorts, and failure of the ground fault detect and voltage detect circuits. The LSSS failures are those which would cause loss of the ability to detect an unsafe condition. These elements include the Fuel Temperature monitors and the Percent Power monitors in the NM-1000, NP-1000 and NPP-1000. System Operable failures are those which cause loss of the ability to monitor the operable condition of other systems, for instance the high voltage monitors. Finally, Computer/Manual Control failures are those associated with the program relays or the manual scram and key switch.

The failure analysis is based on a fault tree approach in which the probability of a particular failure is broken down into component parts which are either added or multiplied together depending on whether the components function in an "or" or an "and" manner respectively. The general equation for the fault tree is:

$$P_{Failure} = P_{Analog} + P_{LSSS} + P_{Diodes} + P_{Comparator} \qquad (1)$$

Where $P_{Failure}$ is the overall probability of the circuit failing to scram in a scram situation and the P_i's are the probability of each of the failure modes described above.

FAULT TREE OVERVIEW



$$P_{Failure} = P_{LSSS} + P_{SysOp} + P_{Comp/Man} + P_{PhySys} \qquad (1)$$

Physical System

There are many potential failures in the physical system. Fortunately, most result in loss of power to the control rods and hence, a scram situation. The possible failure modes are:

Short to line (supply to return)
Power loss
Short to power (20V DC and + to + or - to -)
Short to line (supply to supply or return to return)
Short to ground
Ground detect circuit failure
Short to power (+ to - or not 20V DC)
Power fluctuation
Voltage detect circuit failure

The first two failure types inherently scram the system by cutting off power to the control rods. Therefore, they are not of concern for this analysis. A short along either the supply or return train or to a power supply which is similar to that supplied to the scram circuit would not be detected by the scram circuit. Such a short would negate the safety relays before the short if it were in the supply train or those after a short in the return train. However, for this to lead to an unsafe failure, such shorts would have to occur on both the supply and return trains because all safety monitors are duplicated on both trains. This redundancy structure is shown in the fault tree and makes this a non-single failure mode.

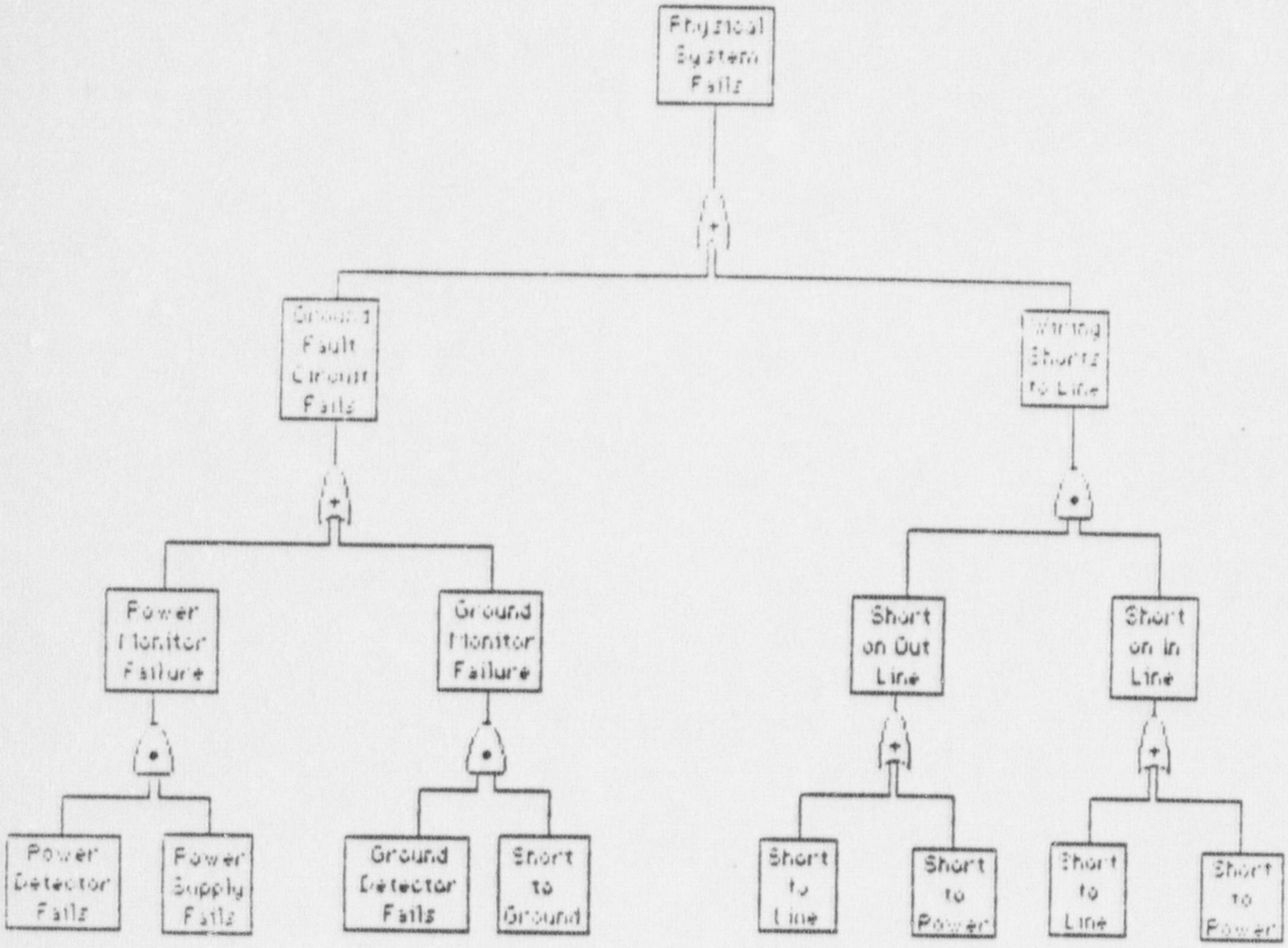
The other branch of the fault tree shows the probabilities associated with faults in the ground detect and voltage detect circuits. For these to cause a potential non-scram situation, however, a short to ground must occur as well as the ground detect failure. Similarly for the voltage detect circuit, only a sensor monitor failure coupled with irregular voltage can cause a potential non-scram situation.

The equation for this segment of the fault tree, then, is:

$$P_{PhysSys} = P_{Cr\ Fault} * P_{Cr\ Detect} + P_{V\ Fault} * P_{V\ Detect} + (P_{Sh\ Line} + P_{Sh\ Power})^2 \quad (2)$$

Where the squared term indicates that either a short to power or along the line must occur on both the supply and return lines. N.B. $P_{V\ Short}$ is the probability of a short to power which is different from the power supply and hence, detectable by the voltage monitoring circuits while $P_{Sh\ Power}$ is the probability of a short to power indistinguishable from the power supply. $P_{PhysSys}$ can be substituted into Equation 1 as part of the overall failure probability.

PHYSICAL SYSTEM FAULT TREE



$$P_{PhySys} = P_{Gr.Fault} + P_{Gr.Det} + P_{Ufault} + P_{UDet} + (P_{Sh.Line} + P_{Sh.Pwr})^2 \qquad (2)$$

Limiting Safety System Setting

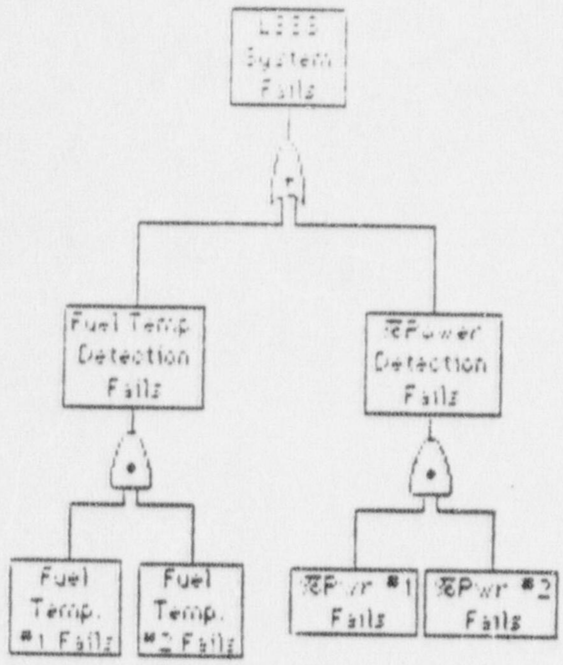
The LSSS consists of the fuel temperature monitors and the percent power monitors. For either high fuel temperature or percent power to cause a non-scram situation, relays on both the supply and return trains must fail. This is because there are two independent fuel temperature monitors, one connected to each line of the scram circuit. Similarly, there are 2 percent power monitors independently connected to the scram circuit so that in order for a failure to occur, both would have to fail. This is clearly a non-single failure mode.

The equation for the probability of LSSS failure as shown in the fault tree is:

$$P_{LSSS} = (P_{F\,Temp})^2 + (P_{\%Pwr})^2 \qquad (3)$$

P_{LSSS} may be plugged into Equation 1 as part of the overall failure probability equation.

LSSS FAULT TREE



$$P_{LSSS} = (P_{FTemp})^2 + (P_{\%Pwr})^2 \qquad (3)$$

System Operable Failure

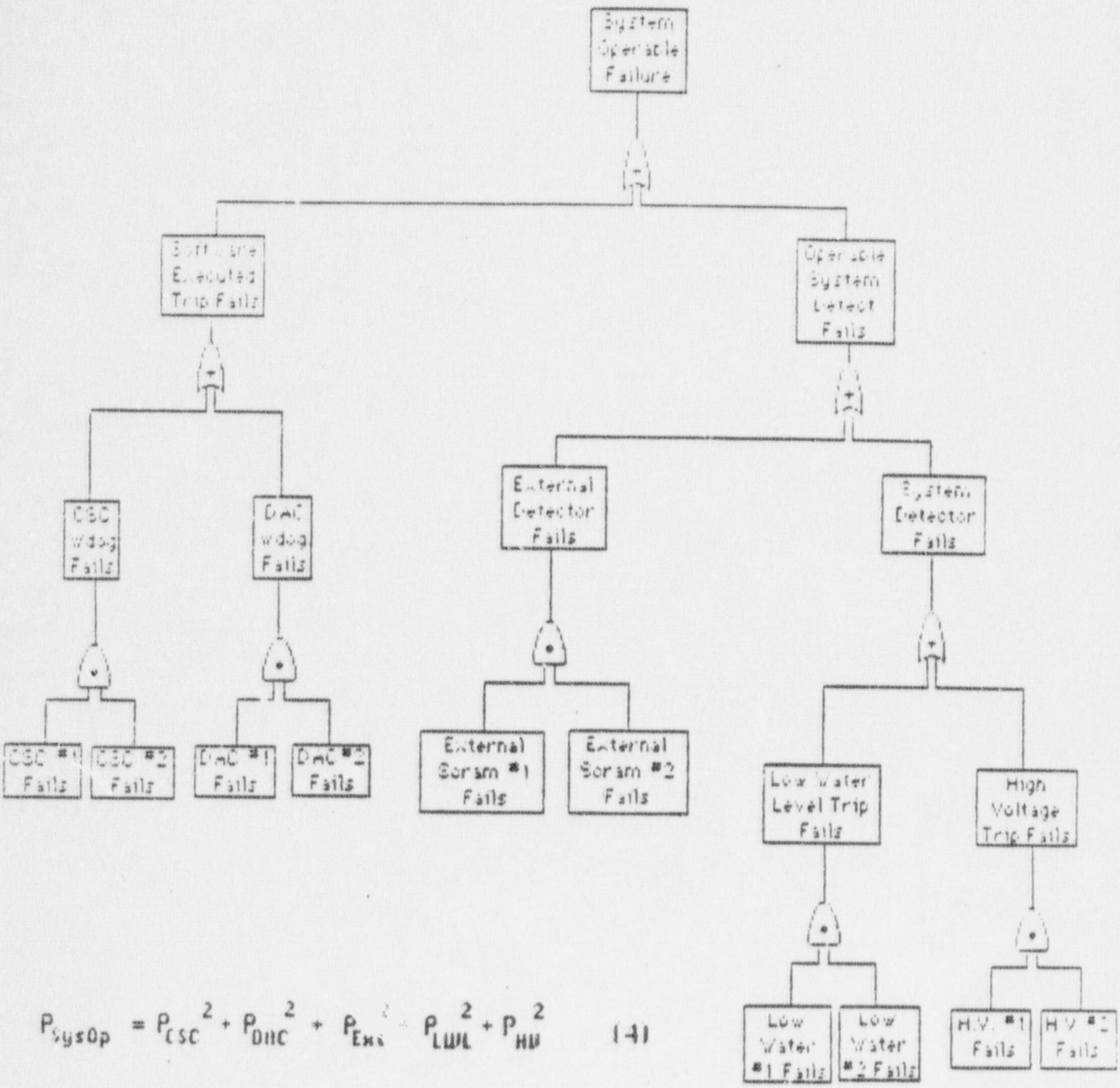
The system operable components are the low-water level, high voltage, watchdog, and external scram relays. Each of these has independent sensors wired into both the supply and return lines and so is a non-single failure mode. The low-water level monitors water level in the tank. High voltage checks the voltage on the percent power monitors and the external scram insures that all external conditions are met, if applicable. There are two pairs of watchdog relays, one for the CSC and one for the DAC. They monitor the software and will scram if not reset every five seconds by their computer.

The equation governing the probability associated with the system operable segment of the fault tree is:

$$P_{SysOp} = (P_{HV})^2 + (P_{LWL})^2 + (P_{ExtScr})^2 + (P_{CSC})^2 + (P_{DAC})^2 \qquad (4)$$

Where the squared terms are due to the redundancy in the system. P_{SysOp} can be plugged into Equation 1 as part of the overall failure probability.

SYSTEM OPERABLE FAULT TREE



$$P_{SysOp} = P_{CSC}^2 + P_{DAC}^2 + P_{Ext}^2 + P_{LWL}^2 + P_{HV}^2 \qquad (4)$$

Computer/Manual Control

This section describes the probability of failure of the program relays and an operator scram. Since the program relays are identical, the possible failures are that one relay fails to open on command, or that two, three or all four fail. If only one relay fails, insertion of the three remaining rods will shut down the reactor so this is not an unsafe failure mode. If any two, three or all four relays fail to open, the reactor will not shut down. It is easily demonstrated with a probability tree analysis that the probability of failure of 2, 3, or 4 of the relays is $6P_f^2 + 4P_f^3 + P_f^4$ where P_f is the probability of a single relay failure. This expression will clearly be dominated by the first term for small P_f so the cube and fourth power terms will be disregarded in further analysis.

The operator scram is normally initiated with the manual scram switch. In the case of a switch failure, however, the operator has other means of shutting down the reactor. These include the key switch and the individual rod controls. The expression for rod control failure is based on the same three-out-of-four logic as the program relays as again, only three rods must be inserted to shut the reactor down.

The expression, then, for the probability of failure of these subsystems is.

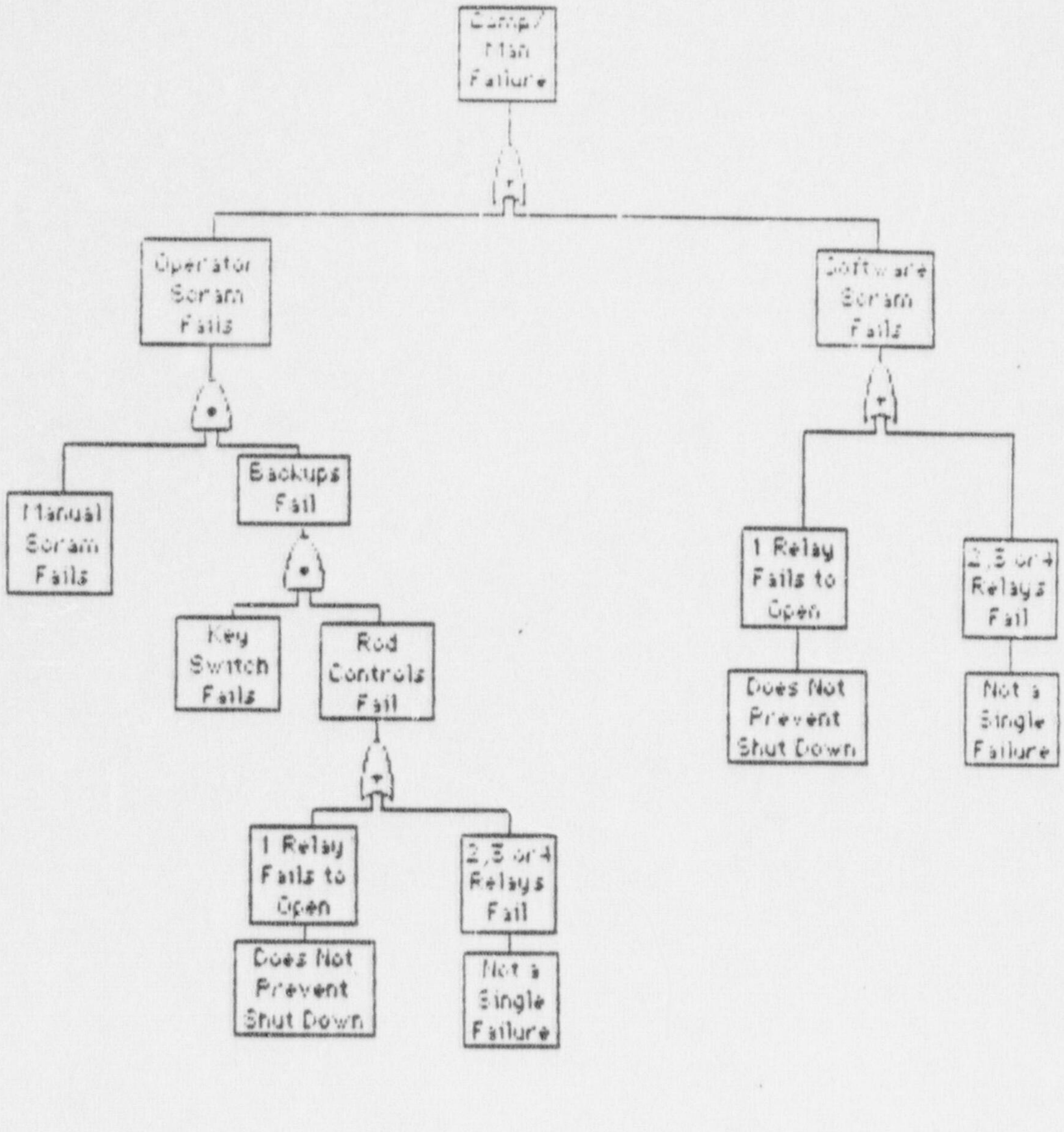
$$P_{Comp/Man} = 6P_{Pr\,Relay}^{\;2} + (P_{Man\,Scr} * P_{Key} * 6P_{RodCtrl}^{\;2}) \qquad (5)$$

Note that the operator has three independent methods to scram the system, all of which must fail for a non-scram situation to arise. This is highly unlikely as the switches themselves are redundant. The manual scram switch, for example, is wired directly into the rod control circuit at two places. Both of which must fail for the manual scram to fail. Similarly, the key switch is wired directly into the scram circuit and also will send a power off signal to the CSC. This signal stops the CSC from updating the watchdog timers and after five seconds, they will time out, scramming the circuit if the direct relay failed to do so. Finally, there are the individual rod controls. These are run through the CSC and so demand that the software be operating properly; however, the watchdog relays are designed to scram the circuit in the event of a software failure. Assuming then that the software is running, only three of the four rod controls must function properly to shut down the reactor, i.e. here again there must be two failures for the system not to scram. Overall, then there must be several catastrophic failures all occurring simultaneously, none of which is caused by an event which would trigger other safety systems, for the operator not to be able to scram the system.

Clearly, the expression is dominated by the chance of a program relay failure and the probability of the operator being unable to scram the system is vanishingly small.

COMPUTER / MANUAL

FAULT TREE



$$P_{Comp/Man} = 6\,P_{Pr.Relay}^{2} + (P_{ManScr} * P_{KeySw} * 6P_{RodCtrl}^{2}) \qquad (5)$$

Failure Analysis

Many of the relays in the scram circuit are of the same type and hence have identical failure probabilities. The high voltage, percent power, low water, watchdog, fuel temperature, external scram and program relays are all similar. An expression for the estimated failure rate for relays is found in <u>Military Handbook 217 Revision E</u>. It is based on the environment, cycles per hour that the relay is expected to operate and of course, relay type.

The Handbook gives the expression for failure as

$\lambda_r = \lambda_b (P_L * P_e * P_c * P_{cyc} * P_f * P_q)$ failures/10^6 hrs (6)

Assuming a double pole, single throw, solenoid relay operating at less than one cycle per hour, carrying less than five Amps, the literature gives the modification factors as:

P_e = 4.6 : Environmental Factor
P_c = 1.5 : Contact Type Factor
P_{cyc} = 1 : Cycle Rate Factor
P_f = 12 : Family Construction/Application Factor
P_q = 1.5 : Quality Rating Factor
P_L = 1.28 : Load Factor
λ_b = .006 : Base Relay Failure Rate

Equation 6 then gives λ_r = 1 failure/10^6 hrs. If P_R is the probability of a relay failure per hour, then $P_R = 1 \times 10^{-6}$ failures/hr.

For the manual scram, control rod and key switches, a similar expression applies:

$\lambda_s = \lambda_b (P_e * P_c * P_{cyc} * P_L)$ failures/10^6 hrs (7)

Where:
P_e = 2.9 : Environmental Factor
P_c = 2.0 : Contact Type Factor

P_{cyc} = 1.0 : Cycle Rate Factor
P_L = 4.77 : Load Factor
λ_b = .054 : Base Switch Failure Rate

Then λ_s = 3 failures/10^6 hrs and P_s = 3×10^{-7} failures/hr. Note that this is only the probability of a physical failure of the switch itself. However, because of the redundancy in the operation of the switches, as described in the section on operator scrams, this probability is much larger than that of the switch operating properly, but failing to scram the system due to internal system failure.

For the conductors in the circuit, data is given by the <u>IEEE Guide to the Collection and Presentation of Electrical, Electronic, Sensing Component and Mechanical Equipment Reliability Data for Nuclear Power Generating Stations</u>. The Guide suggests from empirical data that for a short to ground, the probability is $P_g = 1 \times 10^{-7}$ failures/hour/10 circuit feet. The probability of a short to power is $P_{pwr} = 6 \times 10^{-8}$ failures/hour/ 10 circuit feet. It is assumed that a short to line is similar in probability to a short to power.

The ground and voltage detect circuits were assumed to have the same failure rate as a sensing instrument overall. This is a rather conservative number then, as the detect circuits are much simpler than most sensing instruments and have fewer failure modes. <u>Reliability and Risk Analysis</u> suggests a failure rate for a sensing instrument as: $P_{inst} = 1 \times 10^{-6}$ failures/hour.

The probabilities calculated in the fault tree analysis then, give:

$$P_{SysOp} = 5 * P_R^2 = 5 \times 10^{-12}$$
$$P_{LSSS} = 2 * P_R^2 = 2 \times 10^{-12}$$
$$P_{Comp/Man} = 6P_R^2 + 4P_S^4 = 6 \times 10^{-12}$$

$$P_{PhySys} = P_{a} * P_{inst} + P_{pwr} * P_{inst} + (2P_{pwr})^2 = 2 \times 10^{-13}$$

Using these numbers in Equation 1, we see that

$P_{Failure} = 1 \times 10^{-11}$ failures/hr, or a mean time between failures of 1×10^{7} years. For the failures considered. It is important to note that this is not the expected time for the circuit to go without failure, the long lifetime is rather indicative of the inherent design if the system in that all single failures will cause a scram condition, therefore, only two or more failures occurring simultaneously can lead to a potentially unsafe failure. The improbability of this happening is reflected in the low failure probability

Bypass Relay

The bypass relay is used to cut the NP-1000 out of the scram circuit upon entering pulse mode. When this occurs, only one monitor for percent power remains able to scram the system. The preceeding analysis on failure modes shows that one of the reasons for the extreme safety of the system is the redundancy inherent in all monitoring systems. This redundancy is compromised when the reactor goes into pulse mode. Fortunately, the reactor normally stays in pulse mode for a very short time so the chance of a failure at that instant is very small.

A potential problem could arise, however, if the bypass relay [illegible] failed and the system did not return from pulse mode. In that event, the [illegible]em could operate for an extended period without the NP-1000 to provid[illegible] the extra safety factor. If the bypass relay does fail, however, this failure will be apparent on the operator's display. The percent power indicator for the NP-1000 will remain blank because the CSC will not be receiving any information from it. It is, therefore, important that the operator check the NP-1000 display each time the reactor is pulsed to insure that the bypass relay has returned the system to steady-state operation.

Note that even if the bypass relay fails, the NPP-1000 is still monitoring the system and would be able to scram the system should the percent power exceed its limits. For the circuit to remain in operation and totally unmonitored, the NPP-1000 would also have to fail. This again creates a situation in which two failures must occur for an unsafe situation to arise.

The new probability equation for the LSSS due to the bypass relay is:

$$P_{LSSS} = (P_{FTemp})^2 + (P_{\%Pwr})^2 + (P_{\%Pwr} * P_{bypass}) = 3(P_R{}^2) = 3x10^{-12}$$

Instead of $P_{LSSS} = 2(P_R{}^2) = 2x10^{-12}$ as before.

This still gives an overall $P_{Failure} = 1x10^{-11}$ failures/hr, or a mean time between failures of $1x10^7$ years.

Appendix: Explanation of Equations

The equations given for switch and relay failure are of similar form. They include a base failure rate for the given component type (λ_g) and several modifications (p_i's) based on the individual component and the system in which it operates. The modification factors used are explained below.

p_e : Environmental Factor
p_c : Contact Type Factor
p_{cyc} : Cycle Rate Factor
p_f : Family Construction/Application Factor
p_q : Quality Rating Factor
p_L : Load Factor

Numerical values for the p_i's are given in Military Handbook 217-Rev. E and have been transcribed in part. Most of the modification factors depend on whether the component meets MilSpec standards or is considered "lower quality". In the interest of keeping failure estimates conservative, it is assumed that components are not MilSpec quality.

P_e is based on the environment and installation type. For a fixed ground installation, p_e is 2.9 for switches and 4.6 for relays.

P_C is the same for relays and switches and depends on the form and number of contacts. Values for P_C are shown in Table 1.

Table 1

Type	P_C
SPST	1.0
DPST	1.5
SPDT	1.75
3PST	2.0
4PST	2.5
DPDT	3.0
3PDT	4.25
4PDT	5.5
6PDT	8.0

Table 2

S	P_L
.05	1.02
.1	1.06
.2	1.28
.3	1.76
.4	2.72
.5	4.77
.6	9.49
.7	21.4

Table 3

Rating	P_q
R	.1
P	.3
M	1.0
L	1.0
Not Rated	1.5

For p_L, the load factor, values are determined by S, which is the ratio of the load current to the rated resistive load. P_L values for an inductance based solenoid relay are shown in Table 2 above. The relays are assumed to be rated for 120V which gives an S = .2.

For a switch, p_{cyc} is equal to the number of cycles per hour that the switch is operated (p_{cyc} = 1 if less than 1cycle/hr). For relays, p_{cyc} is 1.0 if the relay operates at less than 10 cycles per hour.

The quality factor, p_q, is shown in Table 3. The relay ratings are unknown and hence are assumed to be unrated.

Finally, p_{cyc} is shown for several relay construction types in Table 4 below.

Table 4

P_f	Contact Current	Construction Type
8	Signal Current	Armature
18	Low mVolt and	Dry Reed
3	mAmps	Hg Wetted
8		Magnetic Latch
14		Solenoid
6	0-5 Amps	Armature
10		Balanced Armature
12		Solenoid

These factors can be plugged into Equations 6 and 7 in the failure analysis to get:

$\lambda_r = \lambda_b$ (P_L * P_e * P_c * P_{cyc} * P_f * P_q) failures/10^6 hrs (6)

λ_r = .006 (1.28 * 4.6 * 1.5 * 1.0 * 12 * 1.5)

λ_r = 1 Failure/10^6 hrs

$\lambda_s = \lambda_b$ (P_e * P_c * P_{cyc} * P_L) failures/10^6 hrs (7)

λ_s = .034 (2.9 * 2.0 * 1.0 * 1.48)

λ_s = .3 Failures/10^6 hrs

Calibration Checks

~~At system startup, the~~ calibration of several systems is checked automatically. These systems are: high voltage monitors, percent power monitors, fuel temperature monitors, and the watchdog timers. The low water level, external scram settings, manual scram switch and key switch are not tested by the auto pretest and should be checked manually.

The percent power, fuel temperature, and high voltage monitors are checked by means of relays which switch from their normal positions to cut the monitors out of the system and allow a test current to be run through the trip section of the system. The CSC monitors when the system trips to insure that it is at the specified point. The relays then return the system to normal operating mode. To check the watchdog timers, the CSC sets each timer and makes sure that it times out at the appropriate time.

For the high voltage, percent power, and fuel temperature systems, if any relay fails to return to normal operating mode, no current from the detectors would reach the monitor circuits and this would result in a scram. If, however, an entire system e.g. the fuel temperature monitors, fails to return to normal mode and the calibration current remained on, the monitors would not scram but the detectors themselves would be completely cut out of the system. This is obviously an undesirable situation. Note that the only way for such a failure to occur is for the CSC to leave the calibration signal active and fail to return the calibration relays to their normal operating positions. Merely leaving the relays in the wrong positions will cause a scram when the calibration current is turned off.

If both of these failures occur in one of the high voltage/percent power monitors, the calibration voltage will be present and show up as variations in percent power and high voltage on the operator's display on the CSC (assuming that the calibration current does not exceed the system limits and cause a scram itself). Also, since the calibration of each monitor unit is checked independently, both must fail for the system as a whole to operate in an unmonitored mode. If the failures occur on the fuel temperature monitors, the CSC display should again show variations due to the calibration current. However, these units are checked all at once so if the system fails, there is no backup system and the fuel temperature remains unmonitored. If the voltage continues to ramp as it does during the calibration check, though, it should quickly trigger a scram on its own.

There are basically two failure modes associated with the watchdog timers: failure to reset and failure to time out. Both of these modes are tested in the pre-start calibration checks by simply setting the timer and letting it time out. Even if the CSC gets stuck in the calibration mode it is a safe failure as in this mode the CSC waits for a time out after setting the timer. Were the system in operation, the first such time out would cause a scram. The watchdog timers could also be reset by a random signal, but this is unlikely as two pairs of timers would require a reset. There are, then, no unsafe failures associated with the watchdog timers' calibration.

The additional failure probabilities for each subsystem due to calibration of the system are assumed to be those of the each subsystem failing all at once. Therefore, there are two terms to be added to the overall failure equation, one for the fuel temperature and one for the percent power/high voltage monitors. The temperature system has three relays which must fail simultaneously and each NP unit has two relays which must fail simultaneously.

$$P_{\%pwr/hv} = P_{unit1} * P_{unit2} = P_R^2 * P_R^2 = P_R^4 = 1x10^{-24} \text{ Failures/hr}$$

$$P_{F.T.} = P_R^3 = 1X10^{-18} \text{ Failures/hr}$$

Clearly both of these failure rates are orders of magnitude smaller than those for the system as a whole. They do not significantly affect the overall failure probability.

Monitor Channels

In addition to the scram circuit itself, safety system failures could occur in the monitors themselves. The monitor channels of specific import are the fuel temperature monitors and the NP-1000 and NPP-1000 percent power / high voltage monitors as these are critical to the safe operation of the system. For this analysis, the channels are all assumed to have the instrument failure rate shown in the above analysis and all failures are assumed to be unsafe. This is a conservative estimate as some common failure modes, e.g. loss of signal from the detector, would cause a scram.

The instrument failure rate is given by $P_{inst} = 1 \times 10^{-6}$ failures/hour. Note that this failure rate is the same as the failure rate used for the relays in the circuit itself. For an unsafe fuel temperature failure to occur, the analysis is identical to that for the scram loop itself i.e. both must fail for the system to be unsafe. This leads to several permutations of failures which are unsafe. However, all require at least two failures. The original expression was $P_{FTemp} = 1 \times 10^{-12}$. Now either the monitor or the relay can fail, but one must fail on each channel. Therefore:

$P_{FTemp} = (P_R + P_I)^2 = 4 \times 10^{-12}$ failures/hr.

Similarly, for the NP-1000 and NPP-1000, the added failure modes increase the number of possible failures, but the system redundancy still protects the system. For the NPP-1000, in addition to the monitor failure, a gain failure is considered. The NPP operates in a separate gain mode for pulse operation and were it to switch to pulse mode during steady state operation the NPP would essentially be useless as the trip point in pulse mode is much higher than for steady state. Since the percent power and high voltage failure rates are incorporated into different parts of the overall failure model and the percent power failure rates are also affected by the bypass relay, it is easiest to look here simply at the increase in failures caused by considering the monitor channel failures. A detailed analysis is presented in the following example. The additional failure probability, considering the interaction of the bypass relay and NPP gain turns out to be

$$P_{HP/NPPFail} = 6 \times 10^{-12} \text{ failures/hr.}$$

This is essentially an increase of 1.1×10^{-11} failures/hr and brings the overall failure rate, incorporating the bypass relay and instrument failures, to 2×10^{-11} failures/hr. This gives a mean time between failures of 5×10^{6} years. Note that this number is essentially double that for the basic system, which is to be expected as the instrument channels considered had similar failure rates to the relays in the circuit itself.

Analysis Example

The following is an example of the analysis used in this failure model. In looking at the percent power system, there are six failures which can cause an unsafe situation. These are failure of: the NP-1000 monitor, the NPP-1000 monitor, the NP-1000 percent power scram relay, the NPP-1000 percent power scram relay, the NPP-1000 gain mode relay, and the pulse mode bypass relay. In all cases, failure of two components is necessary to cause an unmonitored situation, but not all failure pairs will result in such a situation. Since the NP and NPP are on different lines, one component must fail in each i.e. an NP monitor and NP scram relay failure is a safe combination as the NPP-1000 is still fully functional. The table below illustrates the possible failure combinations.

	NPP-M	NPP-R	NPP-G	NP-M	NP-R	Bypass
NPP-M	–	S	S	U	U	U
NPP-R	S	–	S	U	U	U
NPP-G	S	S	–	U	U	U
NP-M	U	U	U	–	S	S
NP-R	U	U	U	S	–	S
Bypass	U	U	U	S	S	–

NPP-M: NPP-1000 Monitor **NPP-R: NPP-1000 Scram Relay** **NPP-G: NPP-1000 Gain**
NP-M: NP-1000 Monitor **NP-R: NP-1000 Scram Relay** **Bypass: Bypass Relay**
S: Safe failure i.e. system still monitored ***U: Unsafe failure, system not monitored***

The table clearly shows the increase in failures from the original model, which had a percent power failure rate of 1×10^{-12} (NP-R and NPP-R in the table). There are nine unique failure modes shown above for the increase of 8×10^{-12} discussed in the monitor channel section.

Conclusion

As stated before, this analysis gives an overall failure probability of 2×10^{-11} failures per hour. This gives an approximate mean time between failures of 5×10^{6} years. Despite the seeming extremity of this number, it was attempted throughout the analysis to make all assumptions as conservative as reasonably possible. The inherent redundancy of the system simply makes it highly improbable that any failure would destroy the integrity of the safety system. For instance the actual operation of 8 hours per day would [illegible] these numbers a factor of 3.

At this point, a comparison of the safety system's reliability to that of the physical system itself might be of interest. Reliability and Risk Analysis gives the failure rate of an individual control rod physically sticking as 1×10^{-4} per day, i.e. 4×10^{-6} failures per hour ~~(This number is actually giving the control rod the benefit of the doubt, as it assumes the reactor operates 24 hours a day. Were the reactor assumed to operate only eight hours a day, the hourly failure rate would be three times higher.)~~ Using the three out of four logic that only three control rods must function in order to cause a scram, the probability of failure equation is identical to that shown for the program relays in the Computer / Manual section and is dominated by the term $6 * P_r^2$. This gives a failure rate for just the control rods as 1×10^{-10} failures per hour.

Granted that this number still provides a reassuringly long mean time between failures (1×10^6 years), the point is that this small section of the physical plant alone has a failure rate which is almost an entire order of magnitude greater than the failure rate for the entire Reactor Safety System. Clearly, the Reactor Safety System is one of the more reliable parts of the reactor design and is not likely to be responsible for any system failures to scram.

APPENDIX E

Analysis of Five Dollar Ramp Insertion Over a Two Second Interval in the AFRRI TRIGA Reactor

Revised
4/26/88

ANALYSIS OF 5 DOLLAR RAMP INSERTION OVER 2 SECOND INTERVAL IN AFRRI TRIGA REACTOR

Work Performed for

ARMED FORCES RADIOBIOLOGICAL RESEARCH INSTITUTE
Bethesda, Maryland

by

GENERAL ATOMICS

under

Contract DNA004-86-C-0011
Amendment P00005

April 14, 1988

AFRRI RAMP ACCIDENT

Summary - With the computer controlled TRIGA Mark F reactor the control rods can be operated in a bank which makes it possible to add large amounts of reactivity in one action. The speed at which the rods can be withdrawn is a variable parameter. An accident scenario is postulated such that during a startup, the following sequence of events occurs:

1. The transient rod is fully withdrawn preparatory to going to a steady state power;

2. The shim, safety and regulating rods are then withdrawn to establish criticality;

3. This withdrawal occurs at a speed which would withdraw the total rod bank in two seconds from a sub-critical condition; and

4. The safety systems terminate the excursion by scramming the reactor at 110% power, i.e., 1.1 MW.

The consequences of this accident are trivial. The maximum fuel temperature is about 330°C. Although the excursion results in a peak power of 340 MW, the reactor power is below 1 MW in less than 1 sec after the initiating event, i.e., the beginning of the rod withdrawal. In Fig. 1 there are shown the results of this accident.

Analysis - Use was made of the computer program BLOOST3, a lumped parameter neutron kinetics, thermal-hydraulic program. This program has been used extensively in the analyses of reactor transients in which reactivity changes are rapid and the event is of short duration.

In Table 1 there are listed the reactor parameters used in the analysis.

TABLE 1

Reactor Parameters

Initial Conditions:

No. of Fuel Elements	87
Core/Coolant Temperature	25°C
Initial Power	0.01 watts
Cold, clean excess	3.5% $\delta\kappa/\kappa$ (\$5.00)

Rod Worths

Transient	2.56% $\delta\kappa/\kappa$	(\$3.66)
Shim	1.30	(1.85)
Safety	1.30	(1.86)
Regulating	1.27	(1.82)
Prompt neutron lifetime	39 μsec	
Fuel element specific heat ($C+\gamma T$)		
C	821.7 joule/°C	
γ	1.67 joule/(°C)^2	

Core water specific heat (per element)

C_w	860 joule/°C

Delayed Neutron Data

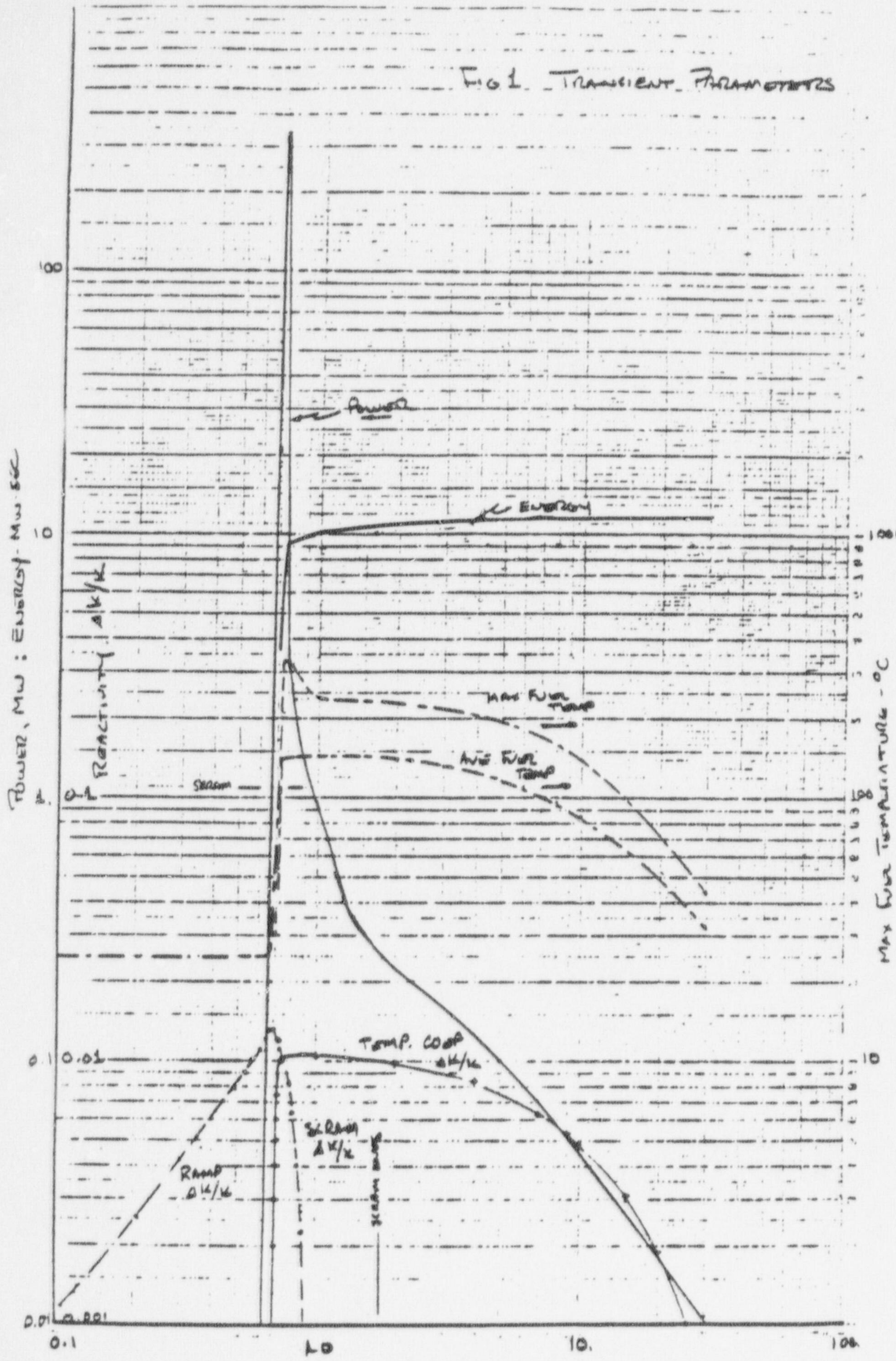
I	β	λ (sec^{-1})
1	2.310×10^{-4}	1.244×10^{-2}
2	1.528×10^{-3}	3.051×10^{-2}
3	1.372×10^{-3}	1.114×10^{-1}
4	2.765×10^{-3}	3.013×10^{-1}
5	8.049×10^{-4}	1.1362×10^{0}
6	2.940×10^{-4}	3.0135×10^{0}

The integral fuel temperature coefficient is shown in Fig. 2. The coefficient itself is approximately 1×10^{-4} $\Delta\kappa/\kappa$°C. The coolant temperature coefficient was assumed to be zero since it is relatively small and, also, because in the excursion little heat is transferred to the water.

With only the transient rod withdrawn the reactor is subcritical by 0.37% $\delta\kappa/\kappa$ (\$0.53). The withdrawal of approximately 10% of the rod bank occurs before criticality is achieved (based on a normalized s-curve for worth vs length withdrawn) so the 3.5% $d\kappa/\kappa$ (\$5.00) insertion occurs in 1.8 secs instead of 2 secs. In Fig. 3 the reactivity inserted as a function of time from the point at which $\kappa = 1.0$ is shown.

Since the transient is terminated when the reactor power is 1.1 MW (110% full power) only a portion of the 3.5% $d\kappa/\kappa$ is inserted at the time of the scram. A problem was run to determine how far the rod bank was withdrawn when the scram occurred. The reactivity inserted in the ramp was 1.305% $d\kappa/\kappa$ (\$1.86). This represents about 34% of the rod length. To this must be added the 10% withdrawn before criticality was achieved. Thus 44% of the rod bank length is out of the core and now participates in the scram. This portion of the length represents 40% of the worth of the bank, or 1.55% $d\kappa/\kappa$ (\$2.21). The total scram activity is, then 1.55% + 2.56% $d\kappa/\kappa$, or 4.11% $d\kappa/\kappa$ (\$5.87) total with the pulse rod worth added to the banked rods. The rods fall under the influence of gravity in 1 sec from full out to full in, following a delay time of .015 secs to allow the magnetic field to decay. Since the rods are also influenced by the resistance implied by the passage through the water, the rate of insertion is not as the second power of time. If there was no resistance the rods would fall from full out to full in in less than 0.3 sec. By assuming a resistance term that is proportional to velocity and that the drop time from full out is 1 sec, the reactivity inserted as a function of time from first motion is shown in Fig. 4.

Conclusions. The postulated accident scenario in which a bank of rods worth 3.87% of $d\kappa/\kappa$ is withdrawn from the AFRRI TRIGA Mark F in 2 secs, with the safety system functioning, will cause no damage to the reactor or harm to any person .



REV 4/26/83 4/13/83

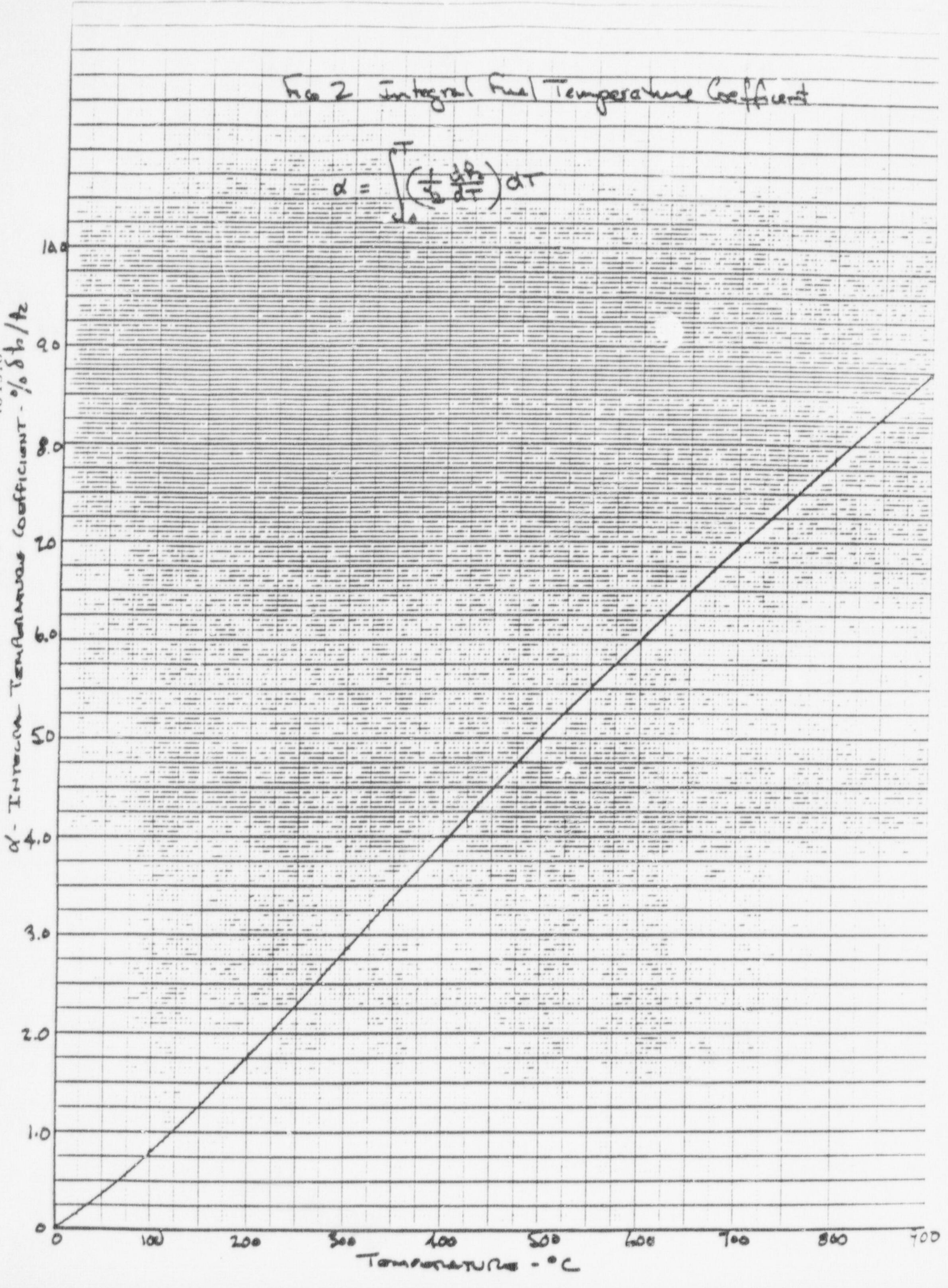


Fig 2 Integral Fuel Temperature Coefficient
$\alpha = \int_{0}^{T} \left(\frac{1}{k_0} \frac{dk}{dT} \right) dT$
α - Integral Temperature Coefficient - % $\delta k/k$
Temperature - °C
0
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0
9.0
10.0
100
200
300
400
500
600
700
800
900

461510

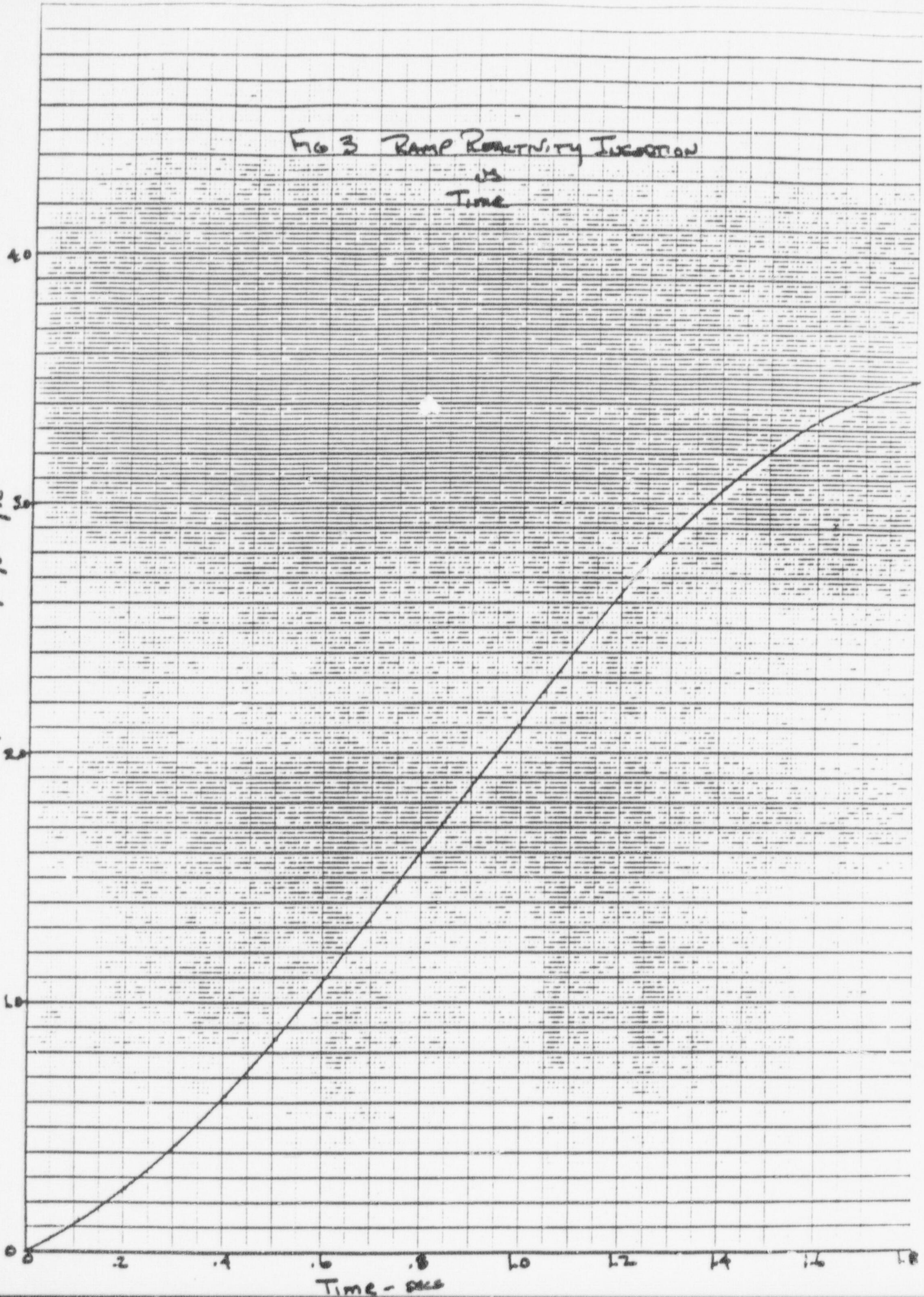
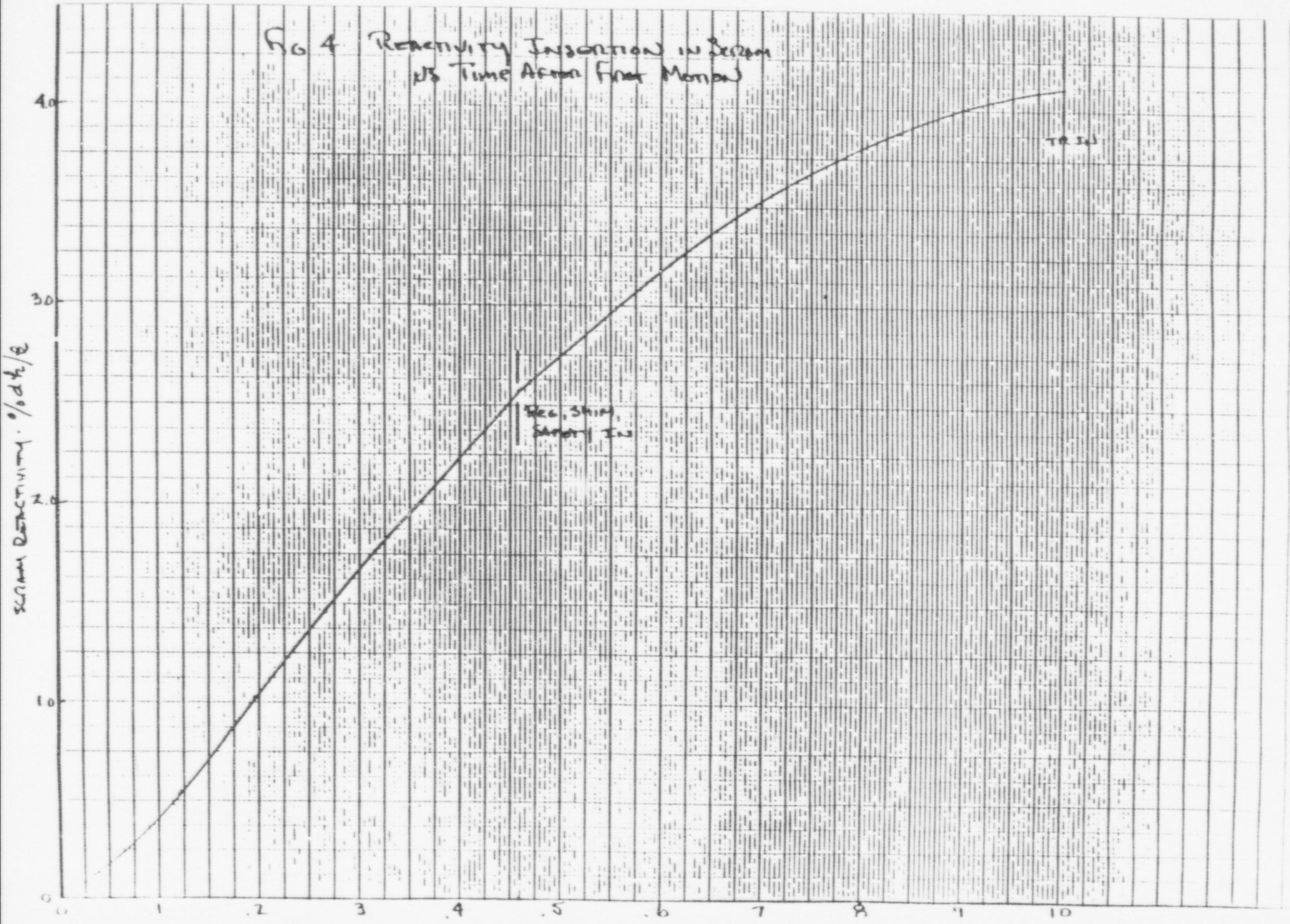


FIG 3 RAMP REACTIVITY INSERTION vs TIME

Fig 4 Reactivity Insertion in Scram vs Time After First Motion



DISPOSITION FORM

For use of this form, see AR340-15; the proponent agency is TAGO.

REFERENCE OR OFFICE SYMBOL	SUBJECT		
RSD	Movement of Equipment Room 3152 Roof Scuttle (Hatch)		
TO	**FROM**	**DATE**	**CMT 1**
Files	Chairman, RSD	1 December 1988 FELTY/jrf/51290	

I concur with the analysis provided herein and find that placement of the roof scuttle (hatch) in room 3152 in its new location poses no increased hazard for personnel who would use this hatch during a reactor emergency.

Mark H Moore Jr

MARK MOORE
Chairman, RSD
Reactor Facility Director

DA FORM 2496

DISPOSITION FORM

For use of this form, see AR340-15; the proponent agency is TAGO.

REFERENCE OR OFFICE SYMBOL	SUBJECT		
RSDR	Movement of Equipment Room 3152 Roof Scuttle (Hatch)		
TO	FROM	DATE	CMT 1
Maj Felty ROS	SFC Cartwright SRO	22 NOV 88 Cartwright, Munno	

1. The roof scuttle for Room 3152 is to be moved to a new location within Room 3152 (see Enclosure 1). This report discusses the impact of the relocation with respect to radiation doses recieved during an accident.

2. The imformation on which this summary is based is provided in three enclosures:

   a. Enclosure 1: Roof plan drawings from contractor.

   b. Enclosure 2: DF "TRIGA Emergencies - Dose Estimates", dtd 20 SEP 78

   c. Enclosure 3: Drawing of core in Northmost position (231) with dose estimates written in.

3. Dose estimates for TRIGA emergiences were determined by Mr. J. Arras (see Enclosure 2). This summary was reviewed to determine if the relocation of the roof scuttle within Room 3152 would significantly affect expected doses to personnel using the scuttle during a reactor emergency involving complete loss of coolant, with the core located in the Northern most portion of the pool.

4. Room 3152 is seperated from the reactor deck, therefore any fission products released during an accident would enter the room through leakage. The dose contribution from fission product release is independent of the location of the roof scuttle.

5. The direct view dose rates were calculated for points within the reactor room (see page 3 of Enclosure 2 of the report). The position of the the core for these estimates was 903, or the Southern most portion of the pool. The points selected at ceiling level (the same height as the roof hatch) were directly above the core, and the furthest point from the core while still in direct view of the core. There is no point determined within the equipment room that allows a direct path for gamma radiation from the core to the roof. The path from the core to the nearest point in the equipment room contains a minimum of five feet of concrete (estimated from drawings). To reduce the intensity of the gamma radiation to one tenth of its original value, at least 18" of concrete is required (Radiological Health Handbook, pg. 149). The five feet of concrete would provide a reduction of intensity by approximately 1000. Therefore the dose due to gamma radiation at the ceiling inside the equipment room is negligible, and the relocation of the roof hatch will not increase the dose received from gamma radiation from the core.

6. In the interest of completeness, dose rate estimates were performed with the core at position 231, or the northern most position of the pool where direct view of the core might be possible from parts of the Equipment Room roof. This required a new drawing (see enclosure 3) and an assumption that dose estimates results from enclosure 2 would be symmetric. The new drawing shows that concrete attentuation of the direct view of gamma radiation at the existing roof

scuttle and at the new roof scuttle installed at a right angle to the core in line with the old scuttle is over 12 feet. Close to half of the roof in room 3152 would be obscured from direct view of the core by over five feet of concrete. Therefore, the radiation dose attributable to direct view gamma would be approximately 0.0 R/hr across half the roof.

7. The scatter, or skyshine, at the roof would drop to a working level due to attenuation of the photons by the roof material. This skyshine level would be less than 0.1 R/hr on the back quarter of the roof.

8. The placement of the roof scutle at any point around the back quarter of the roof will not change any previously predicted levels during accident conditions.

Philip P. Cartwright

C Enclosures
as

Philip P. Cartwright
SFC, USA
Senior Reactor Operator

TO: RFD
Mark Moore

FROM: ROS
MAJ Felty

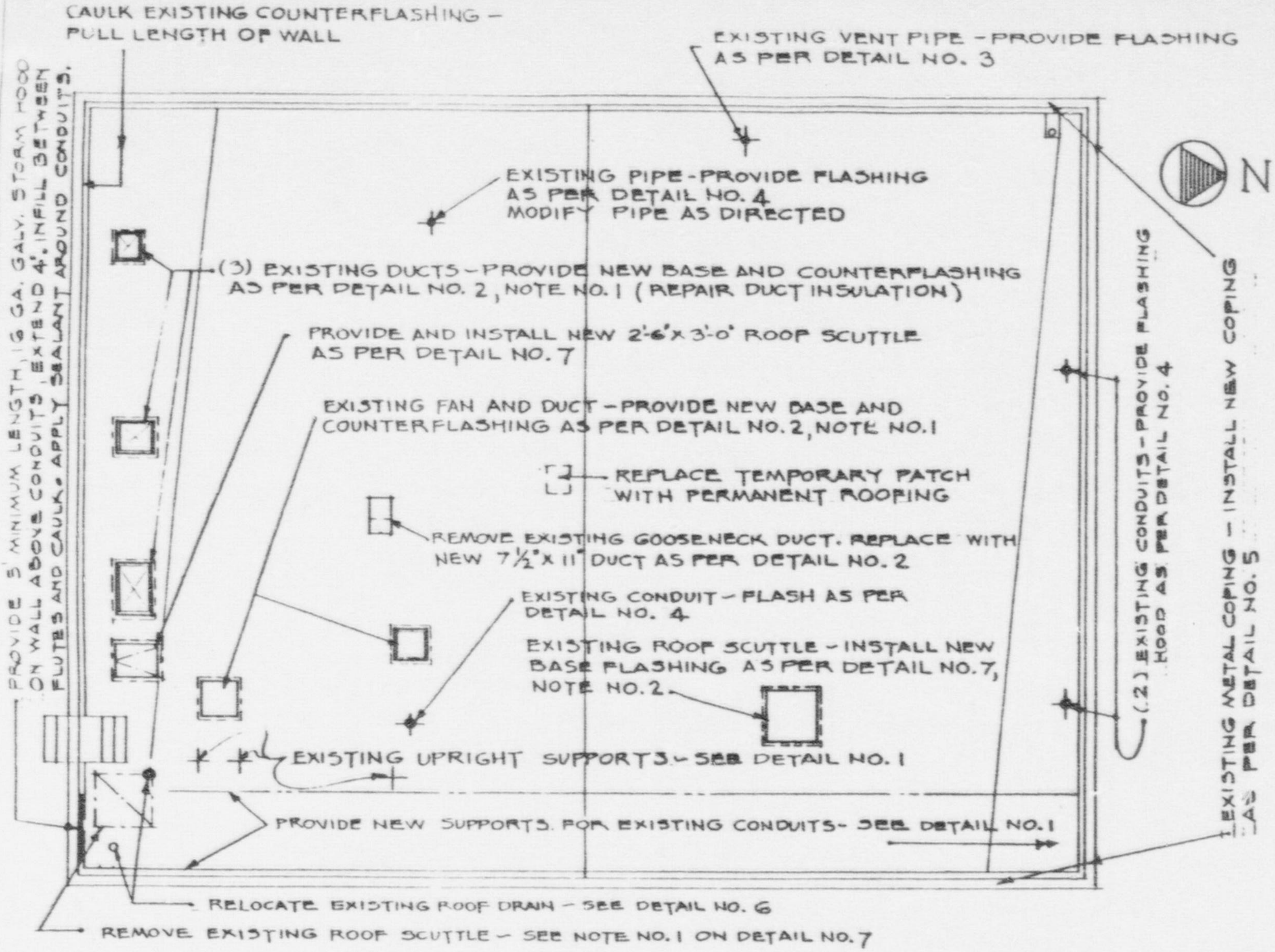
CMT 2

Based on this analysis, placement of the roof scuttle in the new location (as depicted in enclosure 1) presents no increased hazard for personnel who will use the scuttle in the event of a reactor emergency.

JR Felty

James R. Felty
Major, USA
ROS

Enclosure 1



BUILDING 42A - ROOF MODIFICATIONS PLAN

SCALE: 1/8" = 1'-0"

ROOF PLAN	DEPARTMENT OF THE NAVY NAVAL FACILITIES ENGINEERING COMMAND CHESAPEAKE DIVISION WASHINGTON, D.C.
DRWG. X-1	ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE NATIONAL NAVAL MEDICAL CENTER BETHESDA, MD. UPGRADE MECHANICAL/ELECTRICAL SYSTEMS
HENRY ADAMS, INC. Consulting Engineers Baltimore, Maryland	UPGRADE ROOF - BUILDING 42A MODIFICATION P00021
COCHRAN, STEPHENSON AND DONKERVOET - ARCHITECTS	CONSTRUCTION CONTRACT NO. N62477-81-C-0152

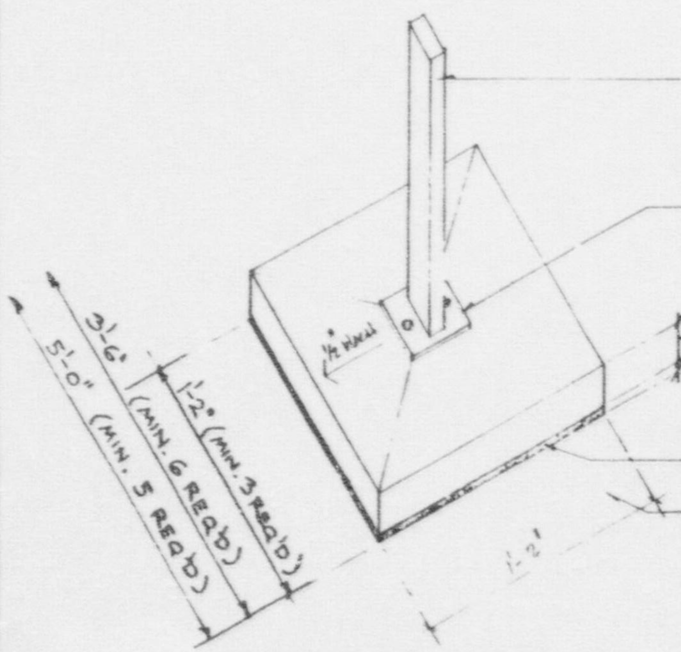
QUANTITIES:

1. PROVIDE 1'-2" X 1'-2" CONCRETE PADS UNDER DUCT AND ONE CONDUIT. (MIN. 3 REQUIRED)
2. PROVIDE 1'-2" X 3'-6" CONCRETE PADS UNDER CONDUITS AT NORTH END. (MIN. 6 REQUIRED)
3. PROVIDE 1'-2" X 5'-0" CONCRETE PADS UNDER CONDUITS AT SOUTH END. (MIN. 5 REQUIRED)
4. THE CONTRACTOR SHALL PROVIDE CONCRETE PAD SUPPORTS FOR ALL EXISTING PITCH POCKETS.

(QUANTITES ARE MINIMUM)

GENERAL:

1. PROVIDE TEMPORARY SUPPORT
2. CUT UPRIGHT SUPPORTS TO PROPER HEIGHT
3. REPAIR ROOF PLYS AND INSULATION (MINIMUM 1'-0" x 1'-0" SQUARE)
4. WELD PLATE TO UPRIGHT SUPPORT END
5. INSTALL ROOF PAD ON ROOF MEMBRANE
6. INSTALL CONCRETE DISTRIBUTION PAD
7. SECURE STEEL PLATE TO CONCRETE WITH BOLT
8. REPLACE AGGREGATE FINISH SURROUNDING CONCRETE PAD



BUILDING NO. 42A-TYPICAL MODIFICATION TO EQUIPMENT AND CONDUIT PITCH POCKET SUPPORTS

DETAIL NO. 1

DRWG. X-2

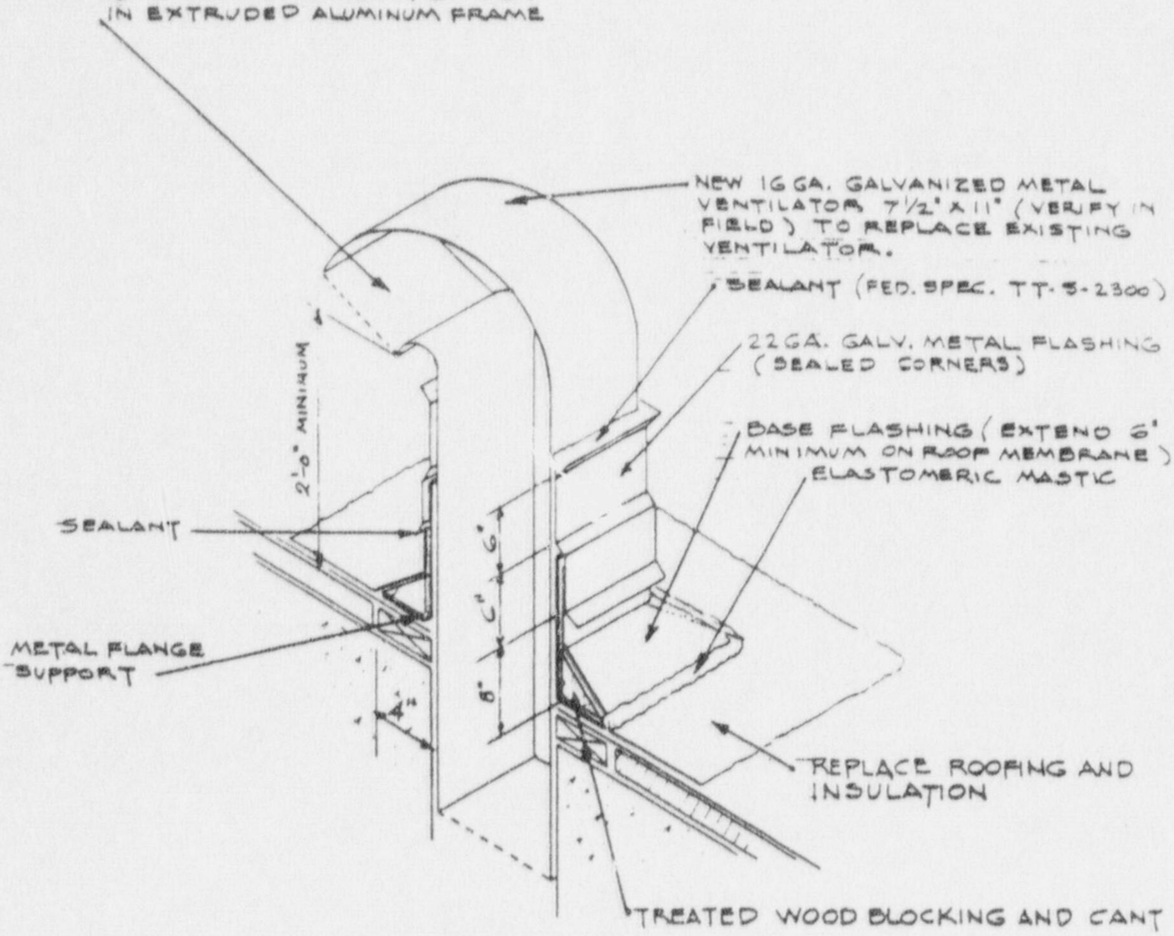
HENRY ADAMS, INC.
Consulting Engineers
Baltimore, Maryland

COCHRAN, STEPHENSON AND DONKERVOET ARCHITECTS

DEPARTMENT OF THE NAVY NAVAL FACILITIES ENGINEERING COMMAND
CHESAPEAKE DIVISION
WASHINGTON, D.C.

ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE
NATIONAL NAVAL MEDICAL CENTER BETHESDA, MD.
UPGRADE MECHANICAL/ELECTRICAL SYSTEMS

UPGRADE ROOF - BUILDING 42A
MODIFICATION P00021
CONSTRUCTION CONTRACT NO. N62477-81-C-0152



NEW GOOSENECK VENTILATOR NO SCALE

NOTE No. 1:
PROVIDE NEW BLOCKING, CANT, BASE FLASHING, COMPOSITION FLASHING, AND METAL FLASHING SIMILAR TO AS SHOWN ABOVE, FOR ALL DUCTS WHERE SHOWN ON THE PLAN. MODIFY HEIGHTS AS REQUIRED.
FAN SHALL BE RAISED MINIMUM OF 8" TO ACCOMODATE NEW FLASHING.

DETAIL NO. 2
DRWG. X-3

DEPARTMENT OF THE NAVY NAVAL FACILITIES ENGINEERING COMMAND
CHESAPEAKE DIVISION
WASHINGTON, D.C.

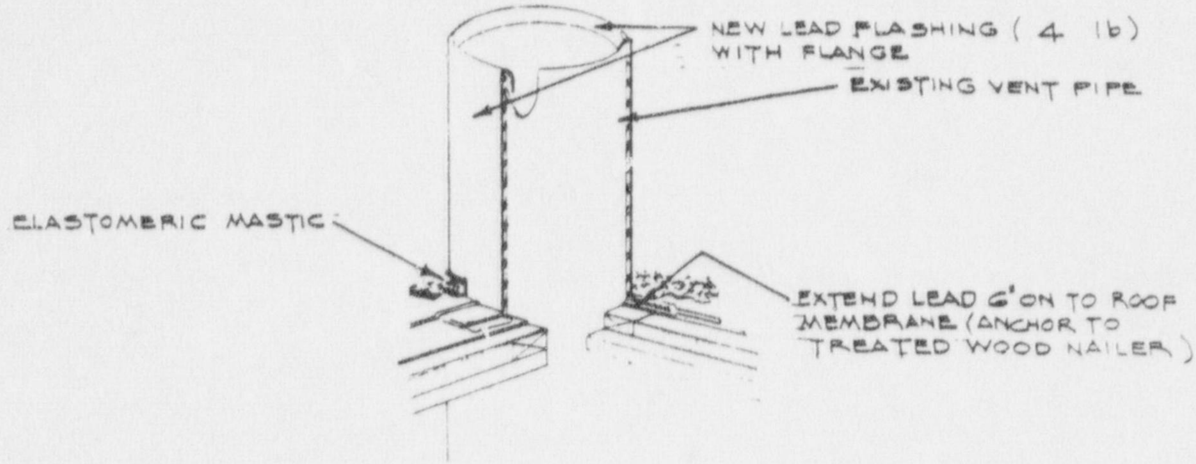
HENRY ADAMS, INC.
Consulting Engineers
Baltimore, Maryland

ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE
NATIONAL NAVAL MEDICAL CENTER BETHESDA, MD.
UPGRADE MECHANICAL/ELECTRICAL SYSTEMS

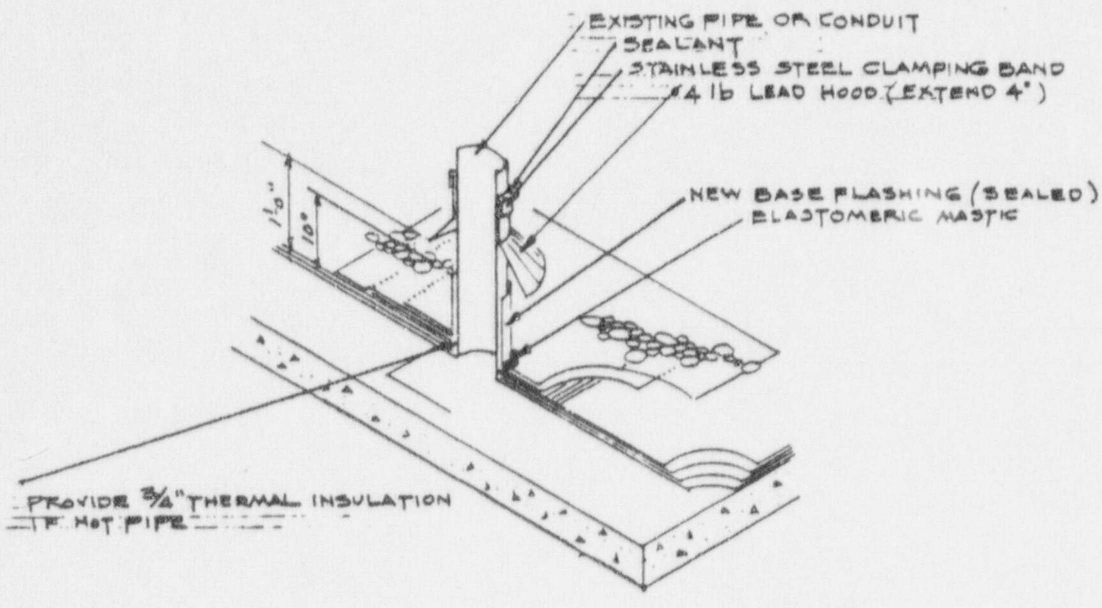
COCHRAN, STEPHENSON AND DONKERVOET — ARCHITECTS

UPGRADE ROOF — BUILDING 42A
MODIFICATION P00021

CONSTRUCTION CONTRACT NO. N62477-81-C-0152



VENT PIPE FLASHING DETAIL No. 3



TYPICAL CONDUIT OR PIPE FLASHING - DETAIL No. 4

DETAIL No. 3 & 4

DRWG. X-4

HENRY ADAMS, INC.
Consulting Engineers
Baltimore, Maryland

COCHRAN, STEPHENSON AND DONKERVOET - ARCHITECTS

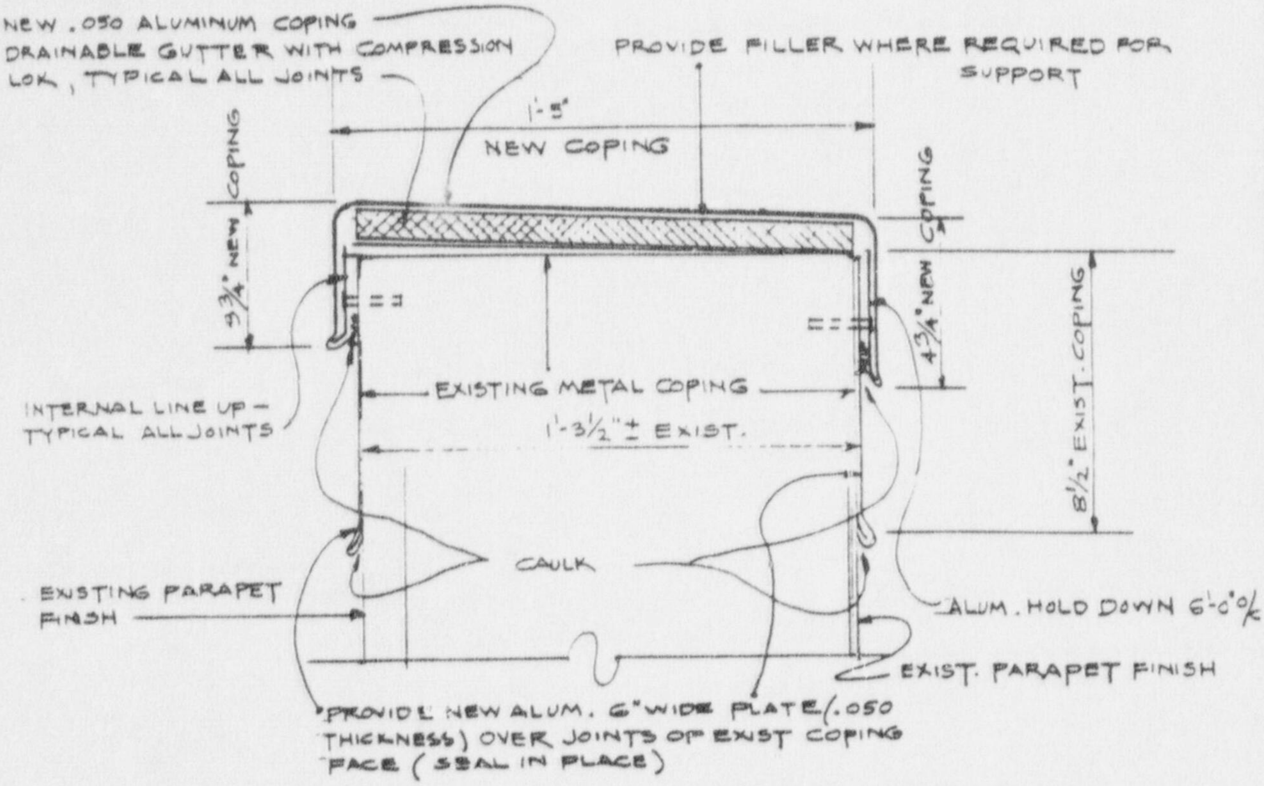
DEPARTMENT OF THE NAVY NAVAL FACILITIES ENGINEERING COMMAND
CHESAPEAKE DIVISION
WASHINGTON, D.C.

ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE
NATIONAL NAVAL MEDICAL CENTER BETHESDA, MD.
UPGRADE MECHANICAL/ELECTRICAL SYSTEMS

UPGRADE ROOF - BUILDING 42A
MODIFICATION P00021

CONSTRUCTION CONTRACT NO. N62477-81-C-0152

NOTE: REMOVE EXISTING CONDUITS, ETC. AND REINSTALL AS DIRECTED.



DETAIL No. 5 : NEW METAL COPING INSTALLED ON ALL EXISTING METAL COPING

SCALE: 3"=1'-0"

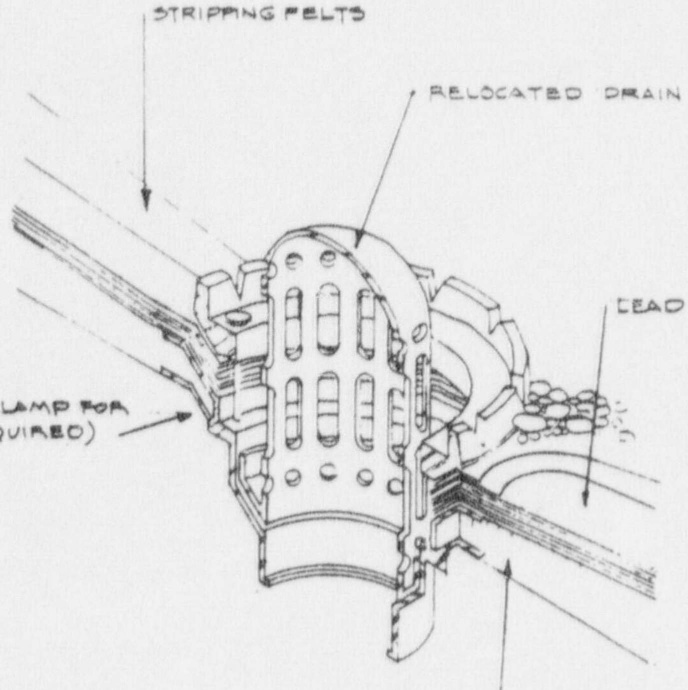
NEW COPING SHALL BE STYLE SL COPING AS MANUFACTURED BY CONSTRUCTION SPECIALTIES, INC. COPING SHALL BE 17" WIDTH, .050" THICKNESS STUCCO ALUMINUM FINISHED IN KYNAR 500 STANDARD COLOR COATINGS. COPING TO BE INSTALLED WITH BUTT JOINTS, 12'-0" O.C. WITH A 3/8" MINIMUM ALLOWANCE FOR EXPANSION. INTERNAL FACE LINE-UP SPLICES TO BE PROVIDED AT ALL JOINTS. INSTALLATION SHALL BE ACCOMPLISHED WITH 6" WIDE RETAINER CLIPS AND DUAL DUROMETER COMPRESSION PADS FASTENED AT ALL JOINTS. 2" WIDE ALUMINUM HOLD DOWN PLATE AND DUAL DUROMETER PADS SHALL BE INSTALLED AT 6'-0" CENTERS BY USE OF SCREWS OR NYLON TAPPITS, PER JOB CONDITION. COMPRESSION PADS TO BE ALIGNED AND MECHANICALLY SECURED PRIOR TO COPING BEING SNAPPED IN PLACE. COPING CORNERS AND VERTICAL AREAS SHALL BE MITERED AND CONTINUOUSLY WELDED PRIOR TO IN-PLANT FINISHING.

DETAIL NO. 5	DEPARTMENT OF THE NAVY NAVAL FACILITIES ENGINEERING COMMAND CHESAPEAKE DIVISION WASHINGTON, D.C.
DRWG. X-5	
HENRY ADAMS, INC. Consulting Engineers Baltimore, Maryland	ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE NATIONAL NAVAL MEDICAL CENTER BETHESDA, MD. UPGRADE MECHANICAL/ELECTRICAL SYSTEMS
COCHRAN, STEPHENSON AND DONKERVOET — ARCHITECTS	UPGRADE ROOF - BUILDING 42A MODIFICATION P00021
	CONSTRUCTION CONTRACT NO. N62477-81-C-0152

MINIMUM 30" SQUARE 4 lb. LEAD FLASHING : SET ON FINISHED ROOFING FELTS IN MASTIC. PRIME TOP SURFACE BEFORE STRIPPING.

MEMBRANE PLIES, METAL FLASHING AND FLASH-IN PLIES : EXTEND UNDER CLAMPING RING.

STRIPPING FELTS : EXTEND 6" BEYOND EDGE OF FLASHING SHEET.



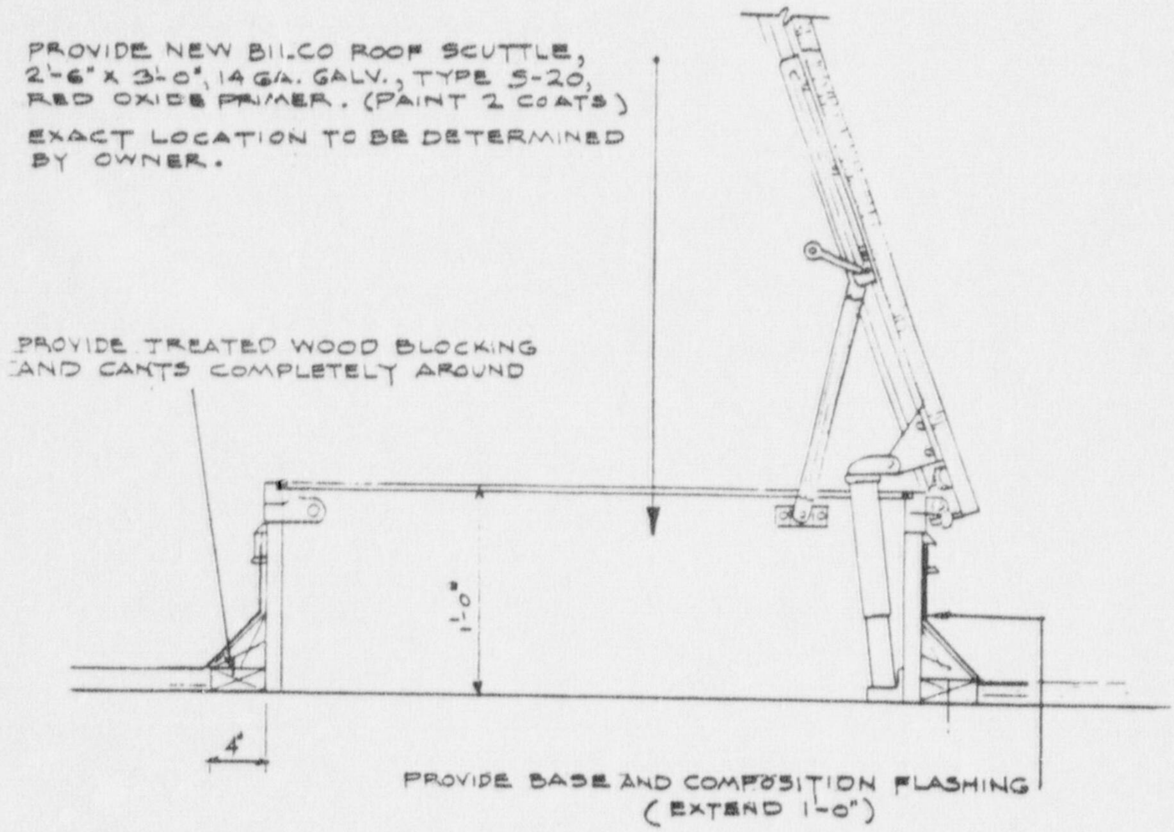
PROVIDE APPROX. 9' STEEL PIPING TO EXISTING VERTICAL PIPE. (INSULATE ALL HORIZONTAL RUNS)

PROVIDE SLOPE IN EXISTING ROOFING FROM SE CORNER TO DRAIN.

DETAIL NO. 6 : RELOCATION OF EXISTING ROOF DRAIN

NO SCALE

DETAIL NO. 6	DEPARTMENT OF THE NAVY NAVAL FACILITIES ENGINEERING COMMAND CHESAPEAKE DIVISION WASHINGTON, D.C.
DRWG. X-6	
HENRY ADAMS, INC. Consulting Engineers Baltimore, Maryland	ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE NATIONAL NAVAL MEDICAL CENTER BETHESDA, MD. UPGRADE MECHANICAL/ELECTRICAL SYSTEMS
COCHRAN, STEPHENSON AND DONKERVOET — ARCHITECTS	UPGRADE ROOF - BUILDING 42A MODIFICATION P00021
	CONSTRUCTION CONTRACT NO. N62477-81-C-0152



NOTE: PROVIDE (4) 3"x3"x 5/16" STEEL ANGLES AND 1½"x1½"x3/16" STRUTS SIMILAR TO EXISTING HATCH AND AS SHOWN ON CONTRACT DRWG. S-16, DETAIL "B".
RELOCATE EXISTING STEEL LADDER.

DETAIL NO. 7: NEW ROOF SCUTTLE

NOTE NO. 1: REMOVE EXISTING ROOF HATCH AND CLOSE OPENING AS SHOWN ON CONTRACT DRWG. S-16, DETAIL "A". REPAIR ALL ROOFING AS REQUIRED TO MATCH EXISTING.

NOTE NO. 2: EXISTING ROOF SCUTTLE TO REMAIN. PROVIDE NEW BLOCKING, CANT, BASE FLASHING AND COMPOSITION FLASHING AS PER THIS DETAIL.

DETAIL NO. 7	DEPARTMENT OF THE NAVY NAVAL FACILITIES ENGINEERING COMMAND
DRWG. X-7	CHESAPEAKE DIVISION WASHINGTON, D C
HENRY ADAMS, INC. Consulting Engineers Baltimore, Maryland	ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE NATIONAL NAVAL MEDICAL CENTER BETHESDA, MD. UPGRADE MECHANICAL/ELECTRICAL SYSTEMS
COCHRAN, STEPHENSON AND DONKERVOET – ARCHITECTS	UPGRADE ROOF – BUILDING 42A MODIFICATION P00021
	CONSTRUCTION CONTRACT NO. N62477-81-C-0152

DISPOSITION FORM

For use of this form, see AR 340-15, the proponent agency is TAGCEN.

REFERENCE OR OFFICE SYMBOL	SUBJECT
SAHP	TRIGA Emergencies - Dose Estimates

TO SAF *JWS.* FROM SAHP DATE 20 Sep 78 CMT 1

Mr Arras/bsm/50351

1. This is a summary of expected sources and levels of external and internal radiation dose, in the event of a TRIGA reactor emergency. Exposures are estimated, where applicable, for both restricted areas and nearby unrestricted areas. Most estimates, as noted, are for the maximum credible incident, which is, for the AFRRI-TRIGA, complete loss of pool water; i.e., coolant.

2. The information on which this summary is based is provided in four enclosures:

    a. Enclosure 1: Excerpts from Final Safeguards Report.

    b. Enclosure 2: Gamma Exposure Rate (Loss-of-Coolant).

    c. Enclosure 3: Fission Product Release (Loss-of-Coolant).

    d. Enclosure 4: TRIGA Emergency Rules-of-Thumb.

Each enclosure is expected to stand alone, and may be attached to any appropriate document.

3. Maximum gamma levels for the reactor area, excerpted from Table 2-2, Enclosure 2, are (approximately):

    a. At the top of the pool, 145 R/h,

    b. On the reactor building roof, 36 R/h,

    c. Outside the reactor building, 1.6 R/h, and

    d. In the control room, 0.6 R/h.

4. Maximum unrestricted area doses from released fission products, extracted from Table 3-4, Enclosure 3, are (approximately):

    a. 1.4 rem/day to total body, and

    b. 3.7 rem/day to thyroid.

John M. Arras

4 Enclosures
as

JOHN M. ARRAS
Head, Operational Health Physics Division

Enclosure 2

Excerpts from: AFRRI-TRIGA Final Safeguards Report
Revised, March 1962, Chapter VI: "Hazards Analysis"

1. General.

a. Only that information deemed relevant, to the purpose of this study, has been extracted from the Final Safeguards Report. The evaluations and conclusions stated is this enclosure, unlike those presented in subsequent enclosures, are those presented directly in the Final Safeguards Report (FSR).

b. Specific hazards have been evaluated and classified in three hazards categories; i.e., these for which one of the following statements is considered valid, according to the FSR:

(1) Physical reactor parmeters and/or interlocks eliminate significant hazard to workers or the general public. Code I.

(2) Administrative procedures are required and followed. A somewhat hazardous condition is possible, but constraints, in the physical system, prevent any serious hazard. Code II.

(3) There is a possibility of a serious hazard. Any such hazard will be discussed later in this report, in detail. Code III.

2. Summary of Hazards

Hazard Considered in FSR	Code (see paragraph 1.b.)
Improper fuel loading	I
Variation in excess radioactivity	I
Malfunction of experiments	II
Reactivity changes	I
Loss of coolant	III
Argon activation	I
Fuel element cladding failure	II
Atmospheric releases	II
Soil activation	I

3. Loss-of-Coolant.

a. Assuming, very conservatively, a virtually immediate water loss ("impossible as per FSR), convection would still adequately remove the after-heat, after either (1) continuous operation at 250 kW or 20 minutes operation at 1 MW. The original assumption is based on aluminum cladding, and the current (stainless steel) cladding should accomplish this at least as well.

b. The maximum release in the event of a cladding failure is estimated as: (1) 5.1 Ci of iodine isotopes, (2) 5.0 Ci of xenon isotopes, and (3) 2.9 Ci of krypton isotopes.

Enclosure 1

c. The FSR does not provide direct information on fission product inventory or dose rates related to loss-of-coolant. Information given in subsequent enclosures is extrapolated from information in the FSR and other references.

d. Loss of moderator, caused by loss-of-coolant, would cause a reactor shutdown, and would not result in any criticality incident. Therefore, no significant neutron dose is expected as a result of this postulated incident.

GAMMA EXPOSURE RATES: Loss-of-Coolant

1. General.

a. The Final Safeguards Report (reference 2.a) does not provide direct information applicable to exposure rates, resulting from loss of the TRIGA reactor pool water, and applicable to typical TRIGA operations. None of the other references obtained provided specific information applicable to such infrequent, short-term, operations. Information estimating gamma-emitting fission product inventory were obtained from references 2.b. and 2.c., and adjusted for reactor runs of approximately one hour.

b. The radionuclide inventory, listed in Table 2-1, was derived from three postulated sources:

(1) Fission products produced by a one medawatt run of one hour duration; i.e., 3600 MW-s,

(2) Fission products produced by a one kilowatt steady-state operation of 100 days duration; i.e., 8640 MW-s, and

(3) Activation products, produced primarily from the steady-state operations.

All three sources are assumed to contribute to any emergency condition.

c. It is assumed, conservatively, that self-absorption within the reactor core approximately equals backscatter and other albedo effects. Therefore, the core is treated as approximately a point source, for any receptor located as much as five meters from it.

d. Most of the special radionuclide data used in this report were obtained from reference 2.d.

Table 2-1. Gamma-Emitting Radionuclide Inventory

Radionuclide Group	Radionuclide Specific	Curies Produced	$\frac{R\text{-}m^2}{Ci\text{-}h}$	Exposure Rate, R/h @ 1m t+10m (*1)	Exposure Rate, R/h @ 1m t+24h (*1)
FS(*2)	Sr-91	321	0.39	125	23
	Y-92	8237	0.125	1030	9
	Y-93	335	0.05	17	3
	Zr-95	213	0.38	81	30
	Nb-97	90	0.35	32	0
	Ru-105	123	0.34	42	1
	Sb-129	90	(2.0)(*3)	180	6
	I-132	219	1.18	256	1

Table 2-1. Gamma-Emitting Radionuclide Inventory
Continued

Radionuclide Group	Specific	Curies Produced	$\frac{R\text{-}m^2}{Ci\text{-}h}$	Exposure Rate, R/h @ 1m t+10m (*1)	t+24h (*1)
FS(*2)	I-133	177	0.26	46	21
	I-135	14	0.99	14	1
	Xe-135	658	0.14	92	15
	(*4)	800	0.7	560	280
FL(*5)	Zr-95	32.9	0.41	13	13
	Nb-95	20.9	0.42	9	9
	Ru-103	25.1	0.26	7	7
	I-131	25.2	0.32	6	6
	Te-132	36.9	0.22	8	8
	Ba-140	51.5	1.24	64	64
	La-140	51.3	1.13	58	58
	Ce-141	43.0	0.04	2	2
	(*4)	200	0.7	140	140
AP(*6)	Na-24	4.3	1.84	8	3
	Mn-56	19.7	0.83	16	1
	Fe-59	0.3	0.64	1	1
	Co-60	1.8	1.30	3	3
	Cu-64	13.8	0.12	2	1
	Co-58	6.5	0.55	4	1
	(*4)	50	1.0	50	50
			Total exposure rate:	2866	807

(*1) t= time of shut-down (t+10 minutes, t+24 hours)
(*2) fission products from 1MW square wave, for one hour
(*3) conservatively estimated from published data (reference 2.d.)
(*4) contributions from radionuclides, not listed because of very low gamma levels and/or small amount produced.
(*5) steady-state, 1kW, 100 days, fission products produced.
(*6) activation products (assume 100 kg of steel in neutron field).

2. Exposure Rate Discussion.

a. All exposure rates are derived from a postulated level of 3000 R/h, at 1m. Figures 1A and 1B show the postulated positions of exposed personnel.

b. Assumed Area Parameters:

(1) reactor located 16.5 ft. below reactor deck, with maximum pool width of 12 feet.

(2) gamma energy = 1 MeV, so tha μ/ρ=0.0637 cm^2/g, buildup factors as follows (for concrete):

μx:	1	2	4	7	10	15	20
BF:	2.0	3.5	6.8	14	23	40	61

(3) density (ρ) = 2.3 (concrete), and 1.7 (soil) g/cm^3, so that the number of mean free paths; i.e., the magnitude of ux, for one foot is: 4.47 (concrete) and 3.30 (soil); see reference 2.d.

c. Accurate skyshine estimates were not obtained. Therefore, conservative estimates were used for skyshine contributions, based on reference 2.e., and with the following assumptions:

(1) energy is roughly equivalent to cobalt-60 (1.25 MeV),

(2) a source located at a distance of 7 feet from a receptor, located on the opposite side of a barrier which is 4 feet high, and

(3) essentially all exposure results from secondary photons produced by compton scatter with air molecules ("skyshine").

The resulting skyshine is, therefore, estimated as 2% of what the direct radiation would be at 7 feet, if there were no barrier.

3. Summary of Exposure Rate Levels.

a. Exposure rates are postulated as arising either from unattenuated radiation which has traversed part of the biological shielding material of the reactor, labeled as DIRECT, or from radiation scattered from the air located above the reactor deck, including that outside the building, labeled as SKYSHINE.

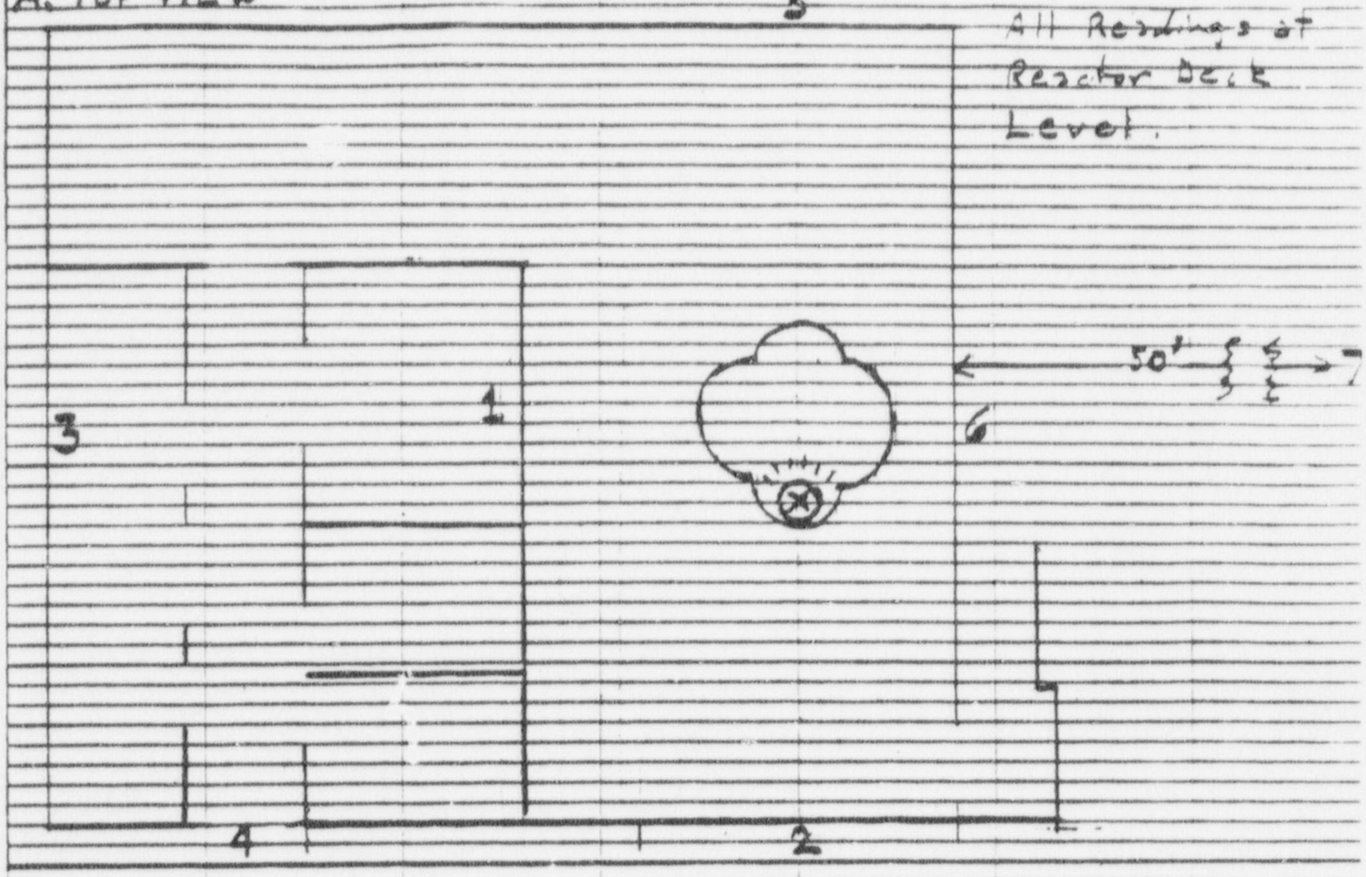
b. Exposure rate data are summarized in Table 2-2. All locations referenced are shown in Figure 1, with the "top view points coded A, and "side view" Points coded B.

Table 2-2. Reactor Deck Area Exposure Rates

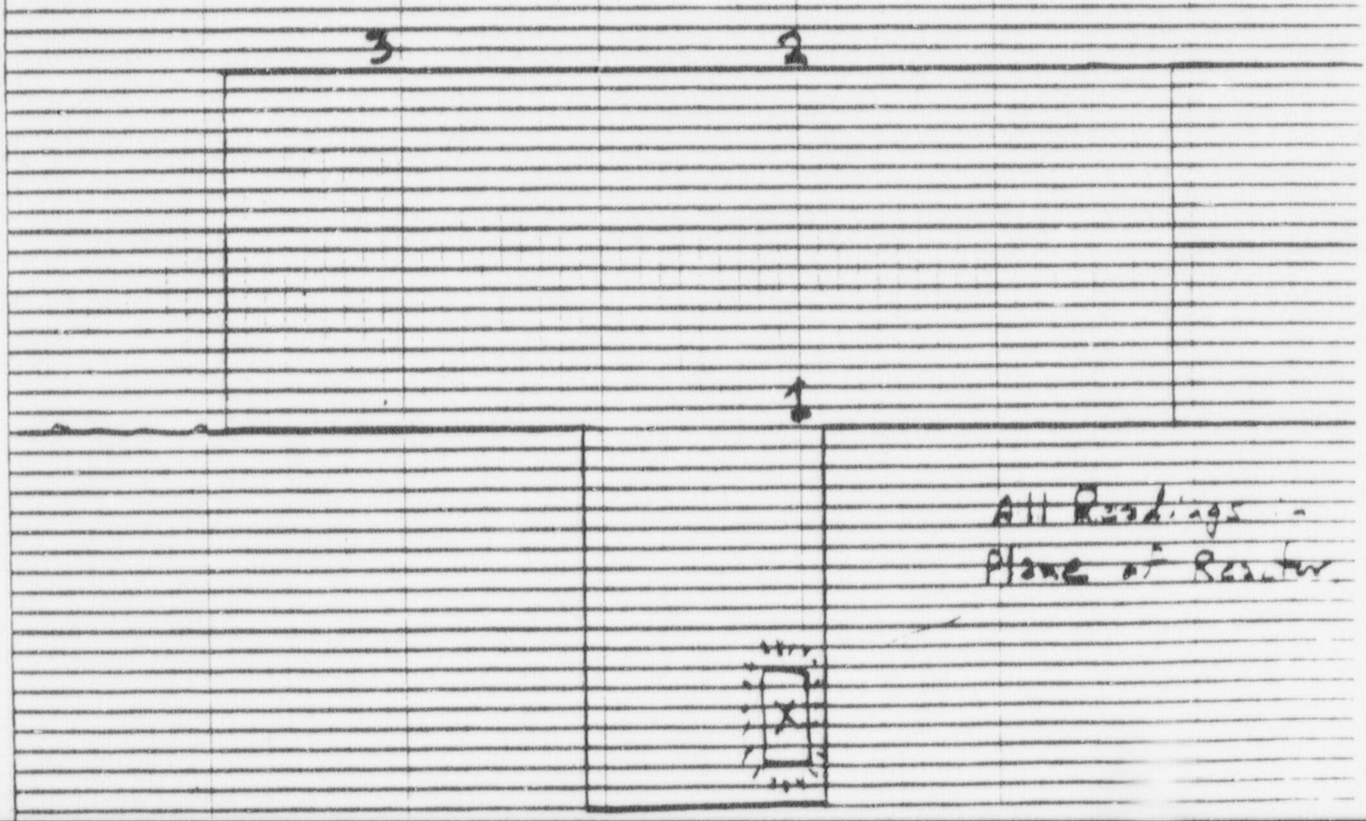
Location	Dist. (feet)	Shielding (ux)	R/h DIRECT	R/h SKYSHINE
A-1	23	33	≈0	0.6
A-2	24	56	≈0	0.2
A-3	40	99	≈0	0.1
A-4	39	105	≈0	<0.1
A-5	30	49	≈0	0.2
A-6	19	8	<0.1	1.5
A-7	61	58	≈0	<0.1
B-1	15	--	133	≈12
B-2	30		33	≈3
B-3	35		25	≈3

FIGURE I. REACTOR DECK RADIATION LEVELS: LOSS OF COOLANT

A. TOP VIEW



B. SIDE VIEW



4. <u>References</u>.

2.a. AFRRI-TRIGA Final Safeguards Report (revised March 1962), Chapter VI, "Hazards Analysis".

2.b. D. H. Slade, Meteorology and Atomic Energy: 1968, U.S.A.E.C., page 314 (Table 7.1).

2.c. Etherington (edit.), Radiation Hygiene Handbook, pp. 2-2 to 2-5, 7-16.

2.d. U.S.P.H.S., Radiological Health Handbook, (both 1960 and 1970 editions), numerous pages.

2.e. Price, Horton & Spinney, Radiation Shielding, Pergamon, 1957, pp. 56-60.

FISSION PRODUCT RELEASE: Loss-of-Coolant

1. General.

a. The expected fission product inventory (FPI) after 100 days operation is given in Table 3-1, based of data from Reference 3.a. This quantity, divided by the appropriate maximum permissible concentration in air (MPC_a) is the hazard index (HI).

b. The radionuclides in Table 3-1 include more than 99.9% of the total HI. Those radionuclides with HI less than $2x10^{10}$ are not included in the final release calculations of Table 3-2.

2. Total Release.

a. The activity released from the fuel elements is assumed to be one percent of the total; this represents 87% of the total activity in one fuel element, 9% of the activity in 10 elements, etc. In view of the statements made in Reference 3.b., this is a conservative assumption.

b. Release from the building is assumed to have occurred, on the average, 24 hours after shutdown. The reactor building provides confinement, but not containment. It is assumed that 100% of gases, and 10% of other fission products, will escape. Given the release conditions, and the information provided in Reference 3.c., this is a conservative assumption. It is also consistent with the conservative assumptions required by USNRC license R-84, for the TRIGA reactor.

c. The release rate, in Ci/s, is averaged over 24 hours, as permitted by 10CFR20.403 (see reference 3.d.).

Table 3-1. TRIGA Fission Product Summary

Radionuclide	% of F.P.(*1)	MPC_a (µCi/ml)(*2)	kγ(*3)	Half-life	Inventory(*4)
Sr-89	5.39	3 (-10)	--	52.7d	1005
Sr-90	0.08	3 (-11)	--	27.7y	14.4
Y-90	0.07	3 (-9)	0.050	64h	14.4
Y-91	6.41	1 (-9)	0.137	58.8d	1240
Zr-95	6.08	1 (-9)	0.366	65.5d	1175
Nb-95	3.93	3 (-9)	0.035	35.0d	1175
Ru-103	4.63	3 (-9)	0.250	39.5d	896
Rh-103	4.63	2 (-6)	0.023	57m	896
Ru-106	0.14	6 (-9)	--	368d	26.9
Rh-106	0.14	2 (-9)(*5)	0.503	2.2h	26.9
Ag-111	0.03	8 (-9)	0.014	7.5d	5.4
Sn-125	0.02	3 (-9)	0.165	9.4d	3.6
Sb-127	0.14	6 (-9)(*5)	0.457	3.8d	2810

Enclosure 3

Table 3-1. TRIGA Fission Product Summary
Continued

Radionuclide	% of F.P.(*1)	MPC_a (μCi/ml)(*2)	kγ(*3)	Half-life	Inventory(*4)
Te-127	0.17	3 (-8)	0.004	9.4d	3400
Te-129	5.01	1 (-7)	0.135	68.7m	50.4
I-131	4.46	1 (-10)	0.192	8.05d	900
Xe-131	0.04	4 (-7)	0.002	11.8d	8.9
Te-132	6.57	4 (-9)	0.117	77.7h	1320
I-132	6.57	3 (-9)	0.904	2.26h	1320
Xe-133	9.81	3 (-7)	0.016	5.27d	1970
Cs-137	0.06	5 (-10)	0.333	30.0y	10.7
Ba-140	9.15	1 (-9)	0.209	12.8d	1840
La-140	9.10	4 (-9)	1.023	40.2h	1830
Ce-141	7.63	5 (-9)	0.038	32.5d	1535
Pr-143	8.00	6 (-9)	----	13.6d	1610
Cr-144	1.75	2 (-10)	0.014	284d	352
Pr-144	1.75	5 (-9)(*5)	0.018	17.3m	352
Nd-147	3.87	8 (-9)	0.069	11.1d	779
Pm-147	0.23	2 (-9)	----	2.62y	46.1
Eu-156	0.02	3 (-9)	0.895	15.2d	3.9

(*1) From Reference 3a., supplemented by comparison data from Reference 3.b., 3.c.
(*2) (-n) $= x10^{-n}$; for unrestricted areas.
(*3) kγ= gamma radiation level from point source, rad-m^2/Ci-h.
(*4) Inventory at time of shutdown, curies.
(*5) Calculated by JMA; others taken from Reference 3.d.

Table 3-2. Postulated Fission Product Release

Radionuclide	Ci(0)(*1)	Ci(24)(*1)	[NI] FPI/MPC_a(*2)	Ci(R)(*3)	Ci(B)(*4)	Q(Ci/s)(*5)
Sr-89	1005	992	33.1(12)	9.92	0.992	1.15(-5)
Sr-90	14.4	14.4	4.80(11)	0.144	0.015	1.74(-7)
Y-90	14.4	14.4	4.80(9)	---	---	---
Y-91	1240	1225	1.23(12)	12.25	1.23	1.42(-5)
Zr-95	1175	1163	1.16(12)	11.63	1.16	1.34(-5)
Nb-95	1175	1163	3.88(11)	11.63	1.16	1.34(-5)
Ru-103	896	880	2.99(11)	8.80	0.88	1.02(-5)
Rh-103	896	880	4.40(8)	---	---	---
Ru-106	26.9	26.8	4.47(9)	---	---	---
Rh-106	26.9	26.8	1.34(10)	---	---	---
Ag-111	5.4	4.9	6.13(8)	---	---	---
Sn-125	3.6	3.3	1.10(9)	---	---	---
Sb-127	2810	2340	3.90(11)	23.40	2.34	2.66(-5)
Te-127	3400	580	1.93(10)	5.80	0.58	6.94(-6)
Te-129	50.4	10^{-4}	---	---	---	---
I-131	900	826	9.00(12)	8.26	8.26	9.56(-5)

Table 3-2. Postulated Fission Product Release
Continued

Radionuclide	Ci(0)(*1)	Ci(24)(*1)	FPI/MPC_a(*2)	Ci(R)(*3)	Ci(B)(*4)	Q(Ci/s)(*5)
Xe-131	8.9	8.4	2.10(7)	0.09	0.09	1.04(-6)
Te-132	1320	1066	2.67(11)	10.66	1.07	1.27(-7)
I-132	1320	1066	3.55(11)	10.66	10.66	1.23(-4)
Xe-133	1970	1727	5.76(9)	17.27	17.27	2.00(-4)
Cs-137	10.7	10.7	2.14(10)	0.107	0.01	1.16(-7)
Ba-140	1840	1743	1.74(12)	17.43	1.74	2.01(-5)
La-140	1830	1734	4.34(11)	17.34	1.73	2.00(-5)
Ce-141	1535	1503	3.01(11)	15.03	1.50	1.74(-5)
Pr-143	1610	1530	2.55(11)	15.30	1.53	1.77(-5)
Ce-144	352	351	1.76(12)	3.51	0.35	4.05(-6)
Pr-144	352	351	1.10(12)	3.51	0.35	4.05(-6)
Nd-147	779	732	9.15(10)	7.32	0.73	8.45(-6)
Pm-147	46.1	46.1	2.31(10)	0.46	0.05	5.32(-7)
Eu-156	3.9	3.7	1.23(9)	---	---	---

(*1) Number of curies present at shutdown (0) and 24 hours after shutdown (24).
(*2) Fission product inventory (FPI), in curies, divided by the appropriate MPC_a; known as <u>hazard index</u> (HI); (n)=$x10^n$.
(*3) Curies released from fuel elements, assuming loss of 1% of total.
(*4) Curies released from building, assuming 100% loss of gases and 10% loss of othe fission products.
(*5) Average release rate, over a 24-hour period; (-n)=$x10^{-n}$.

3. Atmospheric Dispersion.

a. The dispersion equation used (reference 3.f., 3.h.) assumes a release at an elevation below that of the top of the AFRRI stack. The parameter A, representing the effective cross-sectional area of AFRRI at the release point, is assumed to be approximately 200 m^2.

b. The equation is:

$X(Ci/m^3) = Q\ (Ci/s)/\pi\Sigma y\Sigma zU\ (m^3/s)$, with: χ the downwind concentration, U the effective wind speed, Q the average release rate (see Table 3-2) and $\Sigma y\Sigma z$ as defined below:

(1) $\Sigma_y^2 = \sigma_y^2 + 0.5\ A/\pi$

(2) $\Sigma_z^2 = \sigma_z^2 + 0.5\ A/\pi$

(3) For AFRRI conditions, $0.5A/\pi = 31.8$.

c. Dispersion coefficients for Type-F conditions, the worst credible 24-hour conditions, are given in Table 3-3, based on reference g.

Table 3-3. Dispersion Coefficients and Concentrations

Downwind distance x, m	Standard Disp. coeff., m σy	σz	Adjusted disp. coeff., m Σy	Σz	Rel./Conc.* Q/χ; for U=1, AFRRI conditions
50	2.5	1.0	6.2	5.7	110
100	4.0	2.3	6.9	6.1	132
200	7.5	4.1	9.4	7.0	206
400	14.4	7.1	15.5	9.1	441

* Ratio of release rate to ground level concentration (m^3/s)

4. Dose Estimates.

a. The maximum permissible dose (MPD), as specified below, is the dose commitment related to an exposure to a concentration of $1xMPC_a$ for one day. For the total body, MPD = 1.37 millirem, for the thyroid MPD = 4.11 millirem.

b. Table 3-4 summarizes significant releases considered possible in the event of loss-of-coolant, based on data presented in previous tables of this report.

Table 3-4. Dose Estimates for Released Radionuclides

Radionuclide Released	Avg. Q (Ci/s)	$X(Ci/m^3)$ at x=50m	MPC_a-Equivalents Released	Critical Organ
Xe-133	2.00(-4)	1.82(-6)	6	total body
I-132	1.23(-4)	1.12(-6)	373	thyroid
I-131	9.56(-5)	8.69(-7)	8690	thyroid
Xr-Nb-95	2.68(-5)	2.44(-7)	122	total body
Sb-127	2.66(-5)	2.42(-7)	2	total body
Ba-La-140	4.01(-5)	3.65(-7)	182	total body
Other F.P.	9.70(-5)	8.82(-7)	882	total body

c. Table 3-5. Summarized the unrestricted area dose possible to persons located 50 or 400 meters of AFRRI for a period of 24 hours following the release postulated above, assuming the wind blows toward those persons for the entire 24 hours.

Table 3-5. Maximum Credible Dose Summary

Critical Organ	mrem/da @ 50m	mrem/da @ 400m
Total body	1403	350
Thyroid	3724	929

5. References.

3.a. U.S.P.H.S., Radiological Health Handbook, 1960 Edition.

3.b. AFRRI-TRIGA Final Safeguards Report, Table VI 3.

3.c. J. H. Rust & L. E. Weaver, Nuclear Power Safety, Pergamon, 1976, pp. 101-153, "Release of Radioactive Materials from Reactors" (K. Z. Morgan).

3.d. Title 10, CFR, Part 20, "Standards for Protection Against Radiation", Appendix B.

3.e. IAEA Technical Report No. 152, Evaluation of Radiation Emergencies and Accidents: Selected Criteria and Data, E. J. Vallario, U.S.A.E.C., 1974.

3.f. U.S.N.R.C. Regulatory Guide 1.111, "Methods of Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors", March 1976.

3.g. D. H. Slade, Meteorology and Atomic Energy: 1968, U.S.A.E.C.

3.h. HPP 2-5, "Environmental Release Evaluations", September 1978.

TRIGA Emergency Rules of Thumb

1. <u>External Exposure Rate and Dose Commitment.</u>

a. The purpose of this paragraph is to provide a means of estimating internal dose commitment, based on:

(1) presence on the reactor deck following a major release of radioactivity from the TRIGA reactor, and

(2) an assumed airborne radioactivity mixture typical of TRIGA operations; i.e, primarily iodines, kryptons and xenons, with medium-to-short half-lives. (Reference 4.a. and 4.b.).

b. Assumptions include:

(1) radioactive aerosol volume approximating hemispherical shape, with no allowance for scattered gamma radiation (slightly conservative),

(2) effective MPC_a (I) of released fission products calculated as 3×10^{-8} (derived from reference 4.a., 4.b., 4.c., fairly conservative).

(3) effective gamma energy of 0.7 MeV, and gamma constant of 0.4 $R\text{-}m^2/Ci\text{-}h$.

c. External exposure rate.

(1) integrated from infinite cloud model (reference 4.d.), over the postulated reactor deck volume.

(2) the equation is:

$$\text{exposure rate (R/h)} = \int_{r=o}^{5m} \frac{\Gamma\chi(2\pi r^2)}{r^2} e^{-ur}\, dr; \quad E.R. = 2\pi\, \Gamma\, \chi\, (1-e^{-5\mu})/\mu$$

(3) $E.R. = 2\pi x 0.4 x 3 x 10^{-8} x 0.0474 + 0.0097 = 3.68 x 10^{-7}$ R/h

d. Dose commitment.

(1) based on a dose commitment of 5000 millirem in 2000 hours, the expected dose from 1 MPC_a(I) is $2.5 x 10^{-3}$ rem per hour of exposure to the contaminated atmosphere.

(2) therefore, dividing the dose commitment by the exposure rate yields the following:

1 mR/h is equivalent to 7 rem/h

Enclosure 4

e. Conclusion: An exposure for one hour, to a reactor deck concentration which results in an external exposure rate of 1 mR/h, will give a dose commitment of less than 10 rems.

2. <u>CAM Readings and Concentration</u>.

a. The purpose of this paragraph is to provide a means of estimating airborne concentrations, in fission product MPC-equivalents, from continuous air monitor (CAM) readings.

b. Assumptions:

(1) effective $MPC_a(I) = 3 \times 10^{-8}$ µCi/ml (see paragraph 1).

(2) efficiency of CAM detector is 5% (1cpm/20dpm) for mixed fission products

(3) no exposure of the detector to external radiation (conservative assumption)

(4) pump rate = 7 CFM (200,000 cm^3/minute)

(5) essentially no radioactive decay or removal from the collecting filter during the collection period.

c. Discussion: AFRRI CAM's are the fixed-filter type. Therefore, the amount of activity on the collecting filter is the product of the airborne concentration, air flow rate, and total time of buildup. For this reason, <u>all count rates must be interpreted in terms of total buildup time</u>.

d. Table 4-1 summarizes the estimated airborne radioactivity levels corresponding to a net count rate of <u>40,000</u> cpm (80% of full scale) for several buildup periods.

Table 4-1 <u>Buildup Time and Airborne Concentration</u>

Buildup Time (min.)	Increase in cpm/minute	Concentration (µCi/cm^3)	Fraction of MPC	Staytime, 50 mrem* (min.)
5	8000	3.6×10^{-7}	12	100
10	4000	1.8×10^{-7}	6	200
20	2000	9×10^{-8}	3	400
40	1000	4.5×10^{-8}	1.5	800

* Dose commitment (see paragraph 1), assuming <u>no</u> respirator.

3. References.

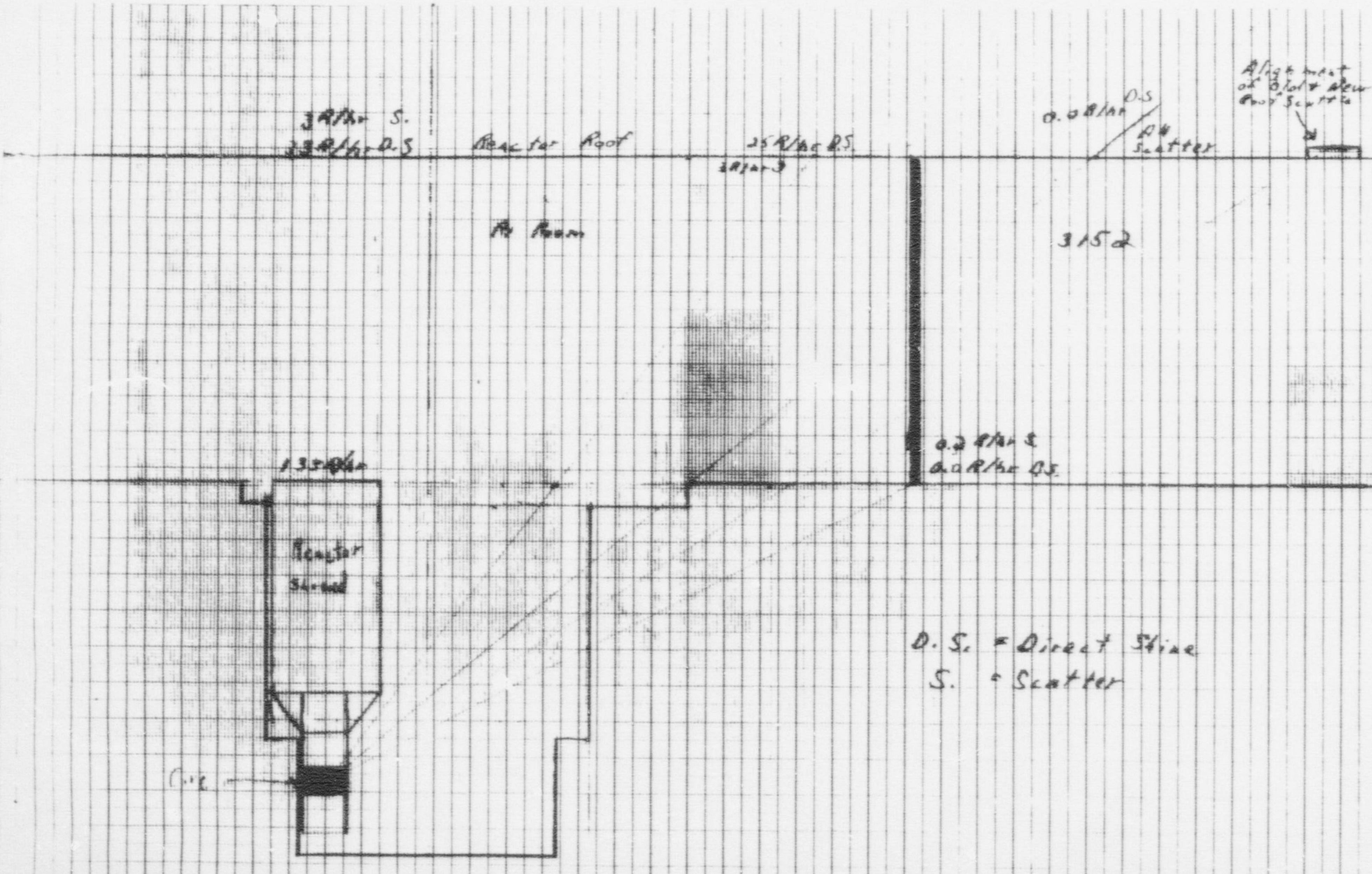
a. Title 10, CFR, Part 20, "Standards for Protection Against Radiation, Appendix B.

b. IAEA Technical Report #152, "Evaluation of Radiation Emergencies and Accidents: Selected Criteria and Data", E. J. Vallario, 1974.

c. N.C.R.P. Report #55, "Protection of the Thyroid Gland in the Event of Releases of Radioiodine", August, 1977.

d. D. H. Slade, Meteorology and Atomic Energy; 1968, U.S.A.E.C., Chapter 7.

Enclosure 3



3 R/hr S.
33 R/hr D.S
Reactor Roof
25 R/hr D.S.
3 R/hr S
0.08 R/hr D.S
R# Scatter
Alignment of Slot New Roof Scuttle
Rx Room
315 Ω
133 R/hr
0.2 R/hr S
0.0 R/hr D.S.
Reactor Shroud
Core
D.S. = Direct Shine
S. = Scatter

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

13 February 1989

TO: Reactor Modifications File

FROM: Reactor Facility Director

SUBJECT: Safety Analyses of Modifications to the Reactor Facility at the Armed Forces Radiobiology Research Institute (AFRRI)

1. In accordance with the provisions of 10 CFR 50.59 safety analyses have been performed on several modifications to the AFRRI TRIGA Mark-F research reactor. The Reactor Facility Director (RFD) has determined that in each case the modification involves no unreviewed safety questions or changes to the facility technical specifications. The Institute's Reactor and Radiation Facility Safety Committee has reviewed these and concurred with the RFD's conclusions.

2. These analyses are hereto appended for the record.

Mark Moore
Mark Moore
Reactor Facility Director

CF: Chief, RSDR

SAFETY ANALYSES OF MODIFICATIONS TO THE REACTOR FACILITY AT THE ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE

DECEMBER 1988

I have determined that modifications to the Reactor Facility, as described in this Technical Report (Safety Analysis Report per 10CFR50.59), involve no unreviewed safety questions and, in fact, are improvements to the operational capability of the facility and radiological safety at AFRRI. I submit this Technical Report to the Reactor Radiation and Facility Safety Committee (RRFSC) for review and concurrence.

Mark Moore

Mark Moore
Reactor Facility Director

The RRFSC has reviewed this Technical Report and concurs with the determination that the Modifications to the Reactor Facility, as described in this Technical Report, involve no unreviewed safety questions.

R I Walker

Richard I. Walker
CAPT, MSC, USN
Chairman, RRFSC

APPROVED

George W. Irving

George W. Irving, III
Colonel, USAF, BSC
Director

ABSTRACT

This report describes changes to the reactor facility at the Armed Forces Radiobiology Research Institute (AFRRI) in Bethesda, Maryland. Classified as a Safety Analysis Report (SAR) that meets the requirements of Title 10, Code of Federal Regulations, Part 50.59 (10 CFR 50.59), this document provides the basis for the conclusion that the changes to the facility involve no unreviewed safety questions and, in fact, are improvements in the operational capability of the facility and radiological safety at AFRRI. In order to accomplish these changes, the SAR must be modified. The body of this report contains a complete description and detailed safety analysis of each of the SAR changes. Excerpts from the SAR and the proposed changes are included as appendices.

Under 10CFR50.59, a licensee may make changes to its facility provided that no changes are made to the Technical Specifications, and that there are no unreviewed safety questions. The conditions for unreviewed safety questions are outlined in 10CFR50.59.a.2, and are summarized below:

If the affected equipment is related to safety:

i. The probability of occurrence or the consequences of an accident or equipment malfunction shall not be increased.

ii. The possibility for an accident or malfunction of a different type than previously evaluated in the SAR shall not exist.

iii. The margin of safety as defined in the Basis for any Technical Specification shall not be reduced.

TABLE OF CONTENTS

INTRODUCTION

Present conditions at the Armed Forces Radiobiology Research Institute (AFRRI) require that certain modification be made to improve the reactor facility. The changes being made to the Safety Analysis Report (SAR) include: the primary Continuous Air Monitor, the Stack Gas Monitor System, the Cerenkov detector, and the Digital Voltmeter.

The Code of Federal Regulations (Title 10. Part 50.59) requires that modification of a portion of a licensed facility as described in the facility SAR be documented with a written safety analysis. Such documentation provides the basis for determining that the change does not involve an unreviewed safety question. Based on the analyses in this Technical Report, it has been determined that the proposed changes to the Reactor Facility do not involve unreviewed safety questions and will actually improve the operational capability of the facility and radiological safety at AFRRI.

This technical report describes changes and modifications made to the AFRRI reactor facility as depicted in the facility's SAR. These changes have been reviewed by the Reactor Facility Director and found to contain no unreviewed safety questions. This report is submitted to the Reactor and Radiation Facility Safety Committee (RRFSC) for their concurrence that conditions of 10CFR50.59 are met. These conditions are that no unreviewed safety questions are present and that the changes made do not increase the probability of occurrence or the consequences of an accident or malfunction.

The proposed modifications require minor administrative changes in the SAR. The body of this report contains a complete description and detailed safety analysis of each of the 10 CFR 50.59 SAR changes. Appendix A contains a specific page/section index of all of the SAR changes. Appendix B contains excerpts from the SAR, for each of these 10CFR50.59 modifications, and the proposed changes to the SAR.

FACILITY MODIFICATIONS

SAFETY ANALYSES

1. Primary Continuous Air Monitor

A flashing visual light installed on the reactor auxiliary instrumentation console gives illumination when the primary reactor room continuous air monitor (CAM) is set in the TEST mode during testing. This flashing visual light is connected to the CAM through switches or relays and is designed to be activated when the CAM is in the TEST mode. Using an output signal from the Test Circuitry in the CAM to activate the flashing light does not degrade the CAM's ability to perform its intended function.

The Technical Specification Basis (3.5.1) for the radiation monitoring system is "... to characterize the normal operational radiological environment of the facility and to aid to evaluating any abnormal operations or conditions. The radiation monitors provide information to the operating personnel of any existing or impending danger from radiation, to give sufficient time to evacuate the facility and take necessary steps to prevent the spread of radioactivity to the surroundings".

A flashing visual light will alert the operator when the CAM is in TEST mode. The reactor is not permitted to operate with the CAM in any mode other than the OPERATE mode. The installation of this flashing visual light will improve the operational capability of the reactor facility without degrading the CAM's ability to perform its intended function. Therefore, there exists no unreviewed safety question for this item.

2. Stack Gas Monitor System

A flashing visual light installed on the reactor auxiliary instrumentation console gives illumination when the Stack Gas Monitor (SGM) pump motor is turned off or the SGM low count warning alerts. The flashing visual light is connected to the SGM through switches or relays and is designed to be activated under these two conditions. Using output signals from the SGM to activate the flashing light does not degrade the SGM's ability the perform its intended function.

The Technical Specification Basis (3.5.1) for the radiation monitoring system is "... to characterize the normal operational radiological environment of the facility and to aid to evaluating any abnormal operations or conditions. The radiation monitors provide information to the operating personnel of any existing or impending danger from radiation, to give sufficient time to evacuate the facility and take necessary steps to prevent the spread of radioactivity to the surroundings".

A flashing visual light will alert the operator when the SGM pump motor is turned off or when the SGM low count warning alerts. In the event of any of these two conditions, the reactor is not permitted to operate. The installation of a flashing visual light will improve the operational capability of the reactor facility without degrading the SGM's ability to perform its intended function. Therefore, there exists no unreviewed safety question for this item.

3. Cerenkov Detector

Sections 4.11 and 4.11.3 are being changed to include a Cerenkov detector. This change will improve the operational capability in accurately monitoring the output of peak power (NV) and integrated power (NVT) during pulse operations.

The Cerenkov detector is a photocell enclosed in an aluminum housing mounted just above the pool water level, directly above the reactor core. This detector was placed above the core in addition to other detectors located underwater just above the reactor core. The intercomparison of data between the Cerenkov detector and gamma ion chamber during pulse operations showed that the gamma ion chamber has a limited upper range due to its physical design limitations and that the Cerenkov detector performs quite well in these upper ranges. The addition of the Cerenkov detector allows the operator to measure the power output of larger pulse operation more accurately. With the Cerenkov detector optimally positioned to measure the larger pulses, the gamma ion chamber can be adjusted to increase the accuracy of measurement of power output of low pulse operations.

The Cerenkov detector and other ion chambers were evaluated in the AFRRI Final Safeguard Report dated March 1962 in conjunction with the issuance of AFRRI's first reactor license from the Atomic Energy Commission in 1962. These instruments were in operation for twenty years. In early 1983, the Cerenkov detector was taken out of the reactor core because it was not being used at that time. In 1988, the Cerenkov detector was put back into the reactor core and calibrated prior to its use for pulse operation.

The Technical Specification Basis (3.2.1) for the channels monitoring the reactor core is "... the power level channels assure that radiations indicating reactor core parameters are adequately monitored for both steady state and pulsing modes of operations". Since the Cerenkov detector has been evaluated in AFRRI Safeguard Report dated March 1962 and was in operation for twenty years prior to its removal in early 1983, the re-installation of this detector in 1988 simply restored the reactor to its original configuration. Therefore, there exists no unreviewed safety question for this item.

4. Digital Voltmeter

Section 4.11.1 of the SAR is being changed to include a digital voltmeter to give parallel readings, if necessary, to the strip chart recorder located on the reactor console. The SAR states that "The power level is scaled on the strip chart recorder between 0 and 100 percent of the power indicated by the power range select switch on the console." The digital voltmeter which gives a more precise output reading from the multi-range linear channel, is used to assist the operator in maintaining a particular reactor power level.

The digital voltmeter is connected into the test points on the linear amplifier circuit provided by the manufacturer for the purpose of measuring the output of the multi-range linear channel with better precision than can be obtained from the linear pen. In addition, this digital voltmeter is used by the operator to make precise measurements at any other set of test points in the console circuitry. Since the addition of a voltmeter, which is isolated, does not alter the multi-range linear channel output, and actually gives the operator a more precise reading when monitoring the reactor power, the addition of this voltmeter actually improves the operational capability of the reactor facility. Finally, the addition of a digital voltmeter is not used as any basis for any Technical Specification item. Therefore, there is no unreviewed safety item for this item.

Appendix A

Listing of Corrections to be made to the SAR

Page	Section	Change
3-39	3.6.2	Changed to include a flashing visual light activation when the primary CAM is in TEST mode. See 10 CFR 50.59 write-up.
3-43	3.6.3.3	Changed to include a flashing visual light activation when the stack gas monitoring system is in low count alert or its pump motor is turned off. See 10 CFR50.59 write-up.
4-22	4.11	Changed to include a Cerenkov detector. See 10 CFR 50.59 write-up.
4-24	4.11.3	Changed to include a Cerenkov detector. See 10 CFR 50.59 write-up.
4-23	4.11.1	Changed to include a digital voltmeter. See 10 CFR 50.59 write-up.

APPENDIX B

Specific SAR word changes based on the Analyses contained in this report.

1. PRIMARY CONTINUOUS AIR MONITOR

Section 3.6.2 of the SAR

CURRENT SAR WORDING:

"A description of the CAM's alarms, locations, and read-outs is given in Table 3-2 and Figures 3-12 through 3-14. The alarm set points can be found in the appropriate AFRRI internal documents.[4]"

PROPOSED SAR WORDING:

"A description of the CAM's alarms, locations and read-out is given in Table 3-2 and Figures 3-12 through 3-14. The alarm setpoints can be found in the appropriate AFRRI internal document.[4] Additionally, a flashing visual light on the reactor auxiliary instrumentation console in the reactor control room will be illuminated when the primary reactor room CAM is set in the TEST mode during testing."

2. STACK GAS MONITOR SYSTEM

Section 3.6.3.3 of the SAR

CURRENT SAR WORDING:

"The stack gas monitor system is capable of activating alarms at two levels."

PROPOSED SAR WORDING:

"The stack gas monitor system is capable of activating alarms at two levels. Additionally, a flashing visual light on the reactor auxiliary instrumentation console in the reactor control room will be illuminated when the stack gas monitoring system low count warning is activated or when the stack gas monitoring system pump motor is turned off."

3. CERENKOV DETECTOR

Section 4.11 of the SAR

CURRENT SAR WORDING:

"The AFRRI-TRIGA reactor core is monitored by six detectors. One thermocouple from each of the two instrumented fuel elements comprise two of the six detectors. A fission detector and three ion chambers comprise the remaining reactor detectors. These six detectors are utilized to provide six independent "channels" which monitor the power level and fuel temperature of the core."

PROPOSED SAR WORDING:

"The AFRRI-TRIGA reactor core is monitored by a variety of detectors. One thermocouple from each of the two instrumented fuel elements comprise two of the detectors. A fission detector, two or more ion chambers, and a Cerenkov detector comprise the remaining reactor detectors. These detectors are utilized to provide at least five independent 'channels' which monitor the power level and fuel temperature of the core during steady state operation and at least three independent 'channels' which monitor the power level and fuel temperature of the core during pulse operations."

Section 4.11.3 of the SAR

CURRENT SAR WORDING:

"High flux safety channels one and two report the reactor power level as measured by three ion chambers placed above the core in the neutron field.

High flux safety channels one and two are independent of one another but operate in identical manners during steady state operation. Each channel consists of an ion chamber placed above the core and the associated electronic circuitry. The steady state power level, as measured by the two high flux safety channels, is displayed on two separate meters located on the reactor console.

During pulse operation, high flux safety channel one is shunted and the sensor for high flux safety channel two is switched to a third, independent ion chamber placed above the core. High flux safety channel two measures the peak power level achieved during the pulse (NV channel)."

PROPOSED SAR WORDING:

"High flux safety channels one and two report the reactor power level as measured by independent power monitors (ion chambers or Cerenkov detector) placed above the core.

High flux safety channels one and two are independent of one another but operate in identical manners during steady state operations. Each channel consists of an ion chamber placed above the core and the associated electronic circuitry. The steady state power level, as measured by the two high flux safety channels, is displayed on two separate meters located on the reactor console.

During pulse operation, high flux safety channel one is shunted and the sensor for high flux safety channel two is switched to a third, independent detector. High flux safety channel two measures the peak power level achieved during the pulse (NV channel)."

4. DIGITAL VOLTMETER

Section 4.11.1 of the SAR

CURRENT SAR WORDING:

"The power level is scaled on the strip chart recorder between 0 and 100 percent of the power indicated by the power range select switch on the console. The strip chart records this output for all steady state modes of operation but not during pulse operation."

PROPOSED SAR WORDING:

"The power level is scaled on the strip chart recorder or indicated on a digital voltmeter for precise reading between 0 and 100 percent of the power indicated by the power range select switch on the console. The digital voltmeter is connected into the test points on the linear amplifier circuit provided by the manufacturer for the purpose of measuring the output of the multi-range linear channel. The strip chart records the output for all steady state modes of operation but not during pulse operation."

ATTACHMENT D

LICENSE EVENT REPORTS

DEFENSE NUCLEAR AGENCY
ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE
BETHESDA, MARYLAND 20814-5415

Memorandum 23 March 1988

LICENSEE EVENT REPORT (LER)

Abstract: The Reactor room primary Continuous Air Monitor (CAM) was accidently left in TEST mode during the reactor operations on 29 February 1988. Two sets of corrective action were taken to prevent reoccurance. One was to instruct the Safety and Health Department staff to be more attentive to their routine maintenance on the Reactor CAMs. The second was the installation of a flashing visual light on the reactor auxiliary instrumentation console in the control room. This flashing light is illuminated by a signal from the Primary Reactor Room CAM when the TEST circuit is activated. This will alert the operator when the CAM is in TEST mode.

Circumstances surrounding the event - 29 February 1988:

0640 Startup checklist completed, all required systems are operable.

0834 Console locked by reactor operator after two experimental runs; reactor operator proceeded downstairs for an Exposure Room opening.

0845 Routine maintenance of reactor CAM's in-progress by a member of the Safety and Health Department (SHD). During this maintenance, the Reactor Room Primary CAM was placed into TEST mode by a member of SHD, a non-licensed individual, and was accidently left in TEST mode upon completion of the routine maintenance (SHD routinely performs the maintenance of all CAM's in the Reactor Facility).

0921 Reactor operator returned to the control room to continue with the experimental runs; the reactor operator did not know that the SHD member had performed a daily check on the CAM, and that the CAM had been accidently left in TEST mode.

The reactor operations for the day, after 0921, included a series of one minute runs at low power of 1 Kw (five runs total), and a series of medium power pulses between $2.05 and $2.15 step reactivity insertions. During the pulse mode operations in the afternoon, the reactor operator on console first noticed above normal CAM readings on the readout meter in the control room, but attributed these above normal readings to those levels which are expected for medium power pulse operations.

At 1521, while doing the Shutdown checklist, the reactor operator printed out an hourly report from the Stack Gas Monitor, which is adjacent to the CAM on the Reactor deck. The Reactor Operator then noticed that the tracing on the strip chart on the CAM was not commensurate with the operations conducted during the day; the CAM trace on the CAM chart recorder was a straight line trace as would be expected from a test input signal, instead of a fluctuating trace as would be expected from a series of medium power pulse operations. Upon further inspection, it was discovered that the CAM was in TEST mode and had been accidently left in TEST mode by the SHD member upon completion of routine maintenance that morning.

This event was due to a Safety and Health Department staff member not following the proper procedures to turn the CAM back to the OPERATE mode after completion of routine maintenance on the CAM. In addition, due to the type of operations performed that day, the reactor control room CAM Readout Monitor levels appeared approximately correct when observed by the Reactor Operator. This was particularly significant as the Startup Checklist was completed earlier that morning, ensuring that all required instruments were operable.

The event was reported to the Reactor Facility Director, who notified the USNRC telephonically, the following morning.

Probable consequences:

Leaving the CAM in the TEST mode during reactor operations would not enable the reactor ventilation system to be automatically secured via closure dampers by a signal from the CAM if the high alarm setpoint had been reached. However, manual closure of the dampers would occur if initiated by operator action. The reactor operations during the day of the incident consisted of a series of short, low power experimental runs and a series of medium power pulses, and thus any release of airborne radioactivity would have been unlikely. In addition, none of the Reactor Room Radiation Area Monitors (R1, R2, R3, R5) nor the Reactor Stack Gas Monitor registered any above normal readings, consistent with the reactor operations throughout the day. The Reactor Stack roughing and HEPA filter systems were functional throughout the day and were capable of performing mitigation had a release occurred. Based on the above, there were no radiation releases due to the incident and no adverse effects on the facility.

Status of corrective action - 1 March 1988:

The event was reported by the Reactor Facility Director to the USNRC, Region I (Curtis Cowgill) by telephone at 1130.

Two sets of corrective actions were taken to prevent recurrence. The first was to instruct the SHD staff to be more attentive to their routine maintenance on the Reactor CAMs. The second was the installation of a flashing visual light on the reactor auxiliary instrumentation console in the control room. This flashing light is illuminated by a signal from the primary Reactor Room CAM when the TEST circuit is activated. This alerts the reactor operator when the CAM is in TEST mode.

Reference:

There have been no previous similar events at this Facility.

Point of Contact: Reactor Facility Director, M. L. Moore (202) 295-1290.

George W. Irving, III
Colonel, USAF, BSC
Director

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

DIR

9 November 1988

Nuclear Regulatory Commission
Region I
Office of Inspection and Enforcement
475 Allendale Road
King of Prussia, PA 19406

SUBJECT: Licensee Event Report

Gentlemen:

In accordance with 10 CFR 50.73, the attached Licensee Event Report is submitted for your consideration.

The point of contact for further information concerning this event is the Reactor Facility Director, M. L. Moore, (202) 295-1290.

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

CF: Herb Williams

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

RSD 8 NOV 88

LICENSE EVENT REPORT (LER)

Abstract: On Oct 11, 1988, the reactor key was placed in the "on" position to review a facility circuit. A short while later the senior reactor operator who had unlocked the console while tracing an electrical circuit left the control room to retrieve an electrical schematic located in an office directly across the hall. Another reactor operator entered the control room, noticed the console was unlocked, locked the console and removed the key. Two corrective actions were taken in response to this event to prevent reoccurrence. One, the reactor operator was counseled by the Reactor Operations Supervisor (ROS) and the Reactor Facility Director (RFD) on his responsibilities as a senior operator. Two, all the reactor console circuit schematics will be available in the control room.

Circumstances surrounding the event - 11 October 1988:

1335 Reactor operator unlocked the console to perform electrical circuit tracing and testing. Schematics used to identify circuitry in the console were obtained from a book shelf in the control room. While tracing a circuit the operator discovered that the other necessary schematics were in a different room across the hall.

1344 The reactor operator left the control room to retrieve the schematics.

1345 Another reactor operator entered the control room, removed the key from the console, and notified the ROS.

The reactor operator who unlocked the console was out of the control room for less than one minute before another reactor operator locked the console. The reactor was shutdown, and there were no operations, core maintenance, or fuel handling, in progress. The only personnel in the reactor were four other reactor operators and one trainee.

The event was reported to the Reactor Operators Supervisor, who then notified the RFD and the USNRC that afternoon, on the same day.

Probable Consequences:

With the reactor in a shutdown condition and no fuel handling, core maintenance, or operations being in progress, there was no reactivity change of the reactor core during this period of time. The reactor power chart recorder indicated no power increases above normal source level during the time when the console was unlocked. This information verifies that there were no consequences to the reactor facility or personnel during this time.

Status of corrective action - 12 October 88

Two corrective actions were taken to prevent a reoccurrence. One, the reactor operator was counseled by the Reactor Operations Supervisor (ROS) and the Reactor Facility Director (RFD) about full compliance with regulations and his responsibilities as a senior operator. Two, all the reactor console circuitry schematics will be available in the control room.

Reference:

There have been no previous similar events at this facility.

Point of Contact: Reactor Facility Director, M. L. Moore (202) 295-1290.

Mark Moore

Mark Moore
Chairman
Radiation Sources Department
Reactor Facility Director

ATTACHMENT E

RRFSC APPROVAL
OF
SPECIAL REACTOR AUTHORIZATION

DISPOSITION FORM

For use of this form, see AR340-15; the proponent agency is TAGO.

REFERENCE OR OFFICE SYMBOL	SUBJECT		
RSD	Special Reactor Authorization # 20		
TO	FROM	DATE	CMT1
Files	Chairman, RSD (RFD)	20 October 1988 FELTY/jrf/51290	

1. On 19 October 1988 the Reactor and Radiation Facility Safety Committee (RRFSC) approved the use of Kodak type LR115 film for irradiation in reactor exposure facilities for radiographic purposes. The attached documentation contains the package that was reviewed by the committee along with the committee's approval voting sheet.

2. This Special Reactor Authorization has been assigned: Special Reactor Authorization # 20.

Mark L Moore

MARK L MOORE
Chairman, RSD
Reactor Facility Director

DA FORM 1 DEC 80 2496

AFRRI Computer Overprint

DISPOSITION FORM

For use of this form, see AR340-15; the proponent agency is TAGO.

REFERENCE OR OFFICE SYMBOL	SUBJECT	
RSD	Request for Special Reactor Authorization	
TO	FROM	DATE CMT1
Reactor & Radiation Facility Safety Committee	Mark Moore Chairman, RSD	19 October 1988 TING/wwt/51290

1. Request a special reactor authorization be issued to place 1.6 grams thin-layer cellulose nitrate film products (Kodak type - LR115 film) in reactor exposure facilities for the purpose of exposing to neutron flux densities of 10-15 x 10EXP12 neutrons per square centimeter. The materials to be irradiated include thin layers of rat tissues with Boron 10 loading on the film. The present user includes an investigator from the National Institute of Health.

2. There has recently been a greater research interest in using these film products which convert each impact by ionizing particle to a separate physical image which could result in a hole in the sensitive film layer. This particular type of film is insensitive to the effect of gamma or beta radiations that are present in neutron beams as well as to light. Therefore, neutron radiography or alpha-particle detection can be carried out in the presence of a high beta or gamma field. This film is being used to study boron neutron capture therapy for brain tumors through a radiation therapy modality, employing the administration of Boron 10 compound, which accumulates in the tumor tissue, followed by irradiation with low-energy neutrons. Detailed information on the efficacy of boronated compounds for neutron capture therapy is described in the attached article from the Brookhaven National Laboratory.

3. Similar experiment was approved by the RRFSC in April 1979, in irradiation of old paintings, for the Smithsonian via the National Bureau of Standards for radiographic verification of age and authenticity. In this new film experiment, there will be no reactivity changes associated with the irradiation of the film and no radiological safety hazard to the personnel. The only safety consideration is the spontaneous ignition of the film at 180 degrees C or greater. At that temperature, over a period of time, the film would incur a slow burning. It is not expected that the 180 degrees C limit be reached at all in this experiment.

4. Radiological safety considerations will be handled through the Safety and Health Department.

Mark Moore

MARK MOORE
Chairman, RSD

Attached: as stated

DA FORM 1 AUG 60 2496

AFRRI Computer Overprint

Request this special reactor authorization be approved and once it is successfully performed it will be converted to routine reactor authorization.

MARK MOORE
Reactor Facility Director

The ~~following~~ is a voting summary of RRFSC members:

APPROVED	DISAPPROVED	R. Walker, CAPT, MSC, USN ~~Chairman~~
APPROVED	DISSAPPROVED	T. O'Brien, Acting Head SHD
APPROVED	DISAPPROVED	M. Moore, Reactor Facility Director, AFRRI
APPROVED	DISAPPROVED	Dr. Marcus H Voth Pennsylvania State University
APPROVED	DISAPPROVED	J. N. Stone Naval Research Lab.
APPROVED	DISAPPROVED	J. Misner, O. R. I
APPROVED	DISAPPROVED	G. Zeman, CDR, MSC, USN. MRA, AFRRI
APPROVED	DISAPPROVED	D. Alberth, MAJ, USN U.S.U.H.S.

Recommend Approval.

R I Walker

RICHARD I. WALKER, CAPT
Chairman, RRFSC

Approved.

George Irving

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

Microanalytical techniques for boron analysis using the $^{10}B(n,\alpha)^{7}Li$ reaction[a), b)]

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(Received 4 January 1985; accepted for publication 9 October 1985)

In order to predict the efficacy of boronated compounds for neutron capture therapy (NCT), it is mandatory that the boron concentration in tissues be known. Various techniques for measurement of trace amounts of boron (1–100 ppm) are available, including chemical and physical procedures. Experience has shown that, with the polyhedral boranes and carboranes in particular, the usual colorimetric and spark emission spectroscopic methods are not reliable. Although these compounds may be traced with additional radiolabels, direct physical detection of boron by nondestructive methods is clearly preferable. Boron analysis via detection of the prompt-γ ray from the $^{10}B(n,\alpha)^{7}Li$ reaction has been shown to be a reliable technique. Two prompt-γ facilities developed at Brookhaven National Laboratory are described. One, at the 60-MW high flux beam reactor, uses sophisticated beam extraction techniques to enhance thermal neutron intensity and reduce fast neutron and γ contamination. The other was constructed at Brookhaven's 5-MW medical research reactor and uses conventional shielding and electronics to provide an "on-line" boron analysis facility adjacent to beams designed for NCT, thus satisfying one of the requisites for clinical application of this procedure. Technical restrictions attendant upon the synthesis and testing of boronated biomolecules often require the measurement of trace amounts of boron in extremely small (mg) samples. A track-etching technique capable of detecting ng amounts of boron in mg liquid or cell samples is described. Thus it is possible to measure the boron content in small amounts (mg samples) of antibodies, or boron uptake in cells grown in tissue culture.

I. INTRODUCTION

Technical difficulties associated with the synthesis of boronated biomolecules make it necessary to measure the boron content of these compounds at various stages in their production. In addition, the many variables involved in evaluating biological effects from the $^{10}B(n,\alpha)^{7}Li$ reaction make it necessary to have accurate information about boron content in tissues during the time course of experimentation. Perhaps most important of all, there is a firm consensus among those who have been or are involved in clinical application of neutron capture therapy (NCT), that knowledge of boron concentration in tissue immediately prior to irradiation of humans is mandatory.[1–4]

It is the common experience of many of those involved in the development of boron compounds for NCT that conventional microanalytical techniques for boron analyses are unsatisfactory.[5] Chemical colorimetric procedures require the conversion of boron from the starting form to boric acid, and are time consuming, taking many hours to days.[6] In particular, we have found that the latter methods do not produce consistent results with the polyhedral borane and carborane cages commonly incorporated in experimental compounds; this inconsistency may be due to incomplete breakdown o[f] the cages.[7] Spark emission spectroscopic techniques we[re] equally unreliable. Conventional radiolabels have been use[d] as tracers in order to circumvent problems encountered [in] boron microanalysis; however, a further synthetic step is re[-]quired to incorporate the radiolabel and in addition, the po[-]tential lability of the tag as well as of boron itself introduc[es] unwarranted complications.

Boron analysis via detection of the prompt-γ ray from the $^{10}B(n,\alpha)^{7}Li$ reaction has been reported by others in the U.S.[8,9] and has been developed in Japan for investigation of NCT.[2,10] The 478-keV prompt-γ ray is emitted from the ex[-]cited state of ^{7}Li in 93.5% of the decays. This method of quantification is independent of the chemical form of boron. Another major advantage of this procedure is that it is non[-]destructive; no observable damage is produced following the relatively low exposures to thermal neutron beams. Conse[-]quently, for example, boronated monoclonal antibodies ma[y] be evaluated for boron content, and then used for biologic[al] experiments. Unfortunately, facilities for prompt-γ-r[ay] analysis often have low sensitivities for μg amounts of boro[n] in tissue samples. In addition, the use of sophisticated bea[m] extraction devices, such as neutron guide tubes to extra[ct] pure thermal neutrons, as well as specialized electronics, such as anticoincidence devices for background suppression, present formidable obstacles to investigators with limited resources.[2,8–10] Two prompt-γ facilities have been developed at BNL: one of these is at the 60-MW high-flux beam reactor (HFBR), and uses sophisticated (and therefore expensive) beam extraction devices to enhance neutron intensity while minimizing fast neutron and γ contamination. The second facility was constructed at the 5-MW medical research reactor (MRR), with conventional (and inexpensive) shielding and electronics. With this apparatus it is possible to obtain rapid "on-line" boron analyses immediately prior to possible patient irradiations (for NCT) at the therapy beam ports of the MRR. Both facilities have similar sensitivities of ~ 200 counts/min per μg ^{10}B, and can detect at least 1 μg ^{10}B per gram tissue in a few minutes.

While prompt-γ measurements are adequate for boron analysis of g amounts of human and animal tissues, they are not sensitive enough to detect ng amounts in μl (mg) samples. Biochemical techniques used in the synthesis and testing of boronated biomolecules typically produce extremely small quantities of natural boron for measurement; monoclonal antibodies are difficult to obtain and expensive even in mg amounts; various column chromatographic analyses produce multiple small samples; and typical cell culture experiments produce $\sim 10^6$ cells (~ 1 mg) per cell culture flask. Such aliquots require boron analysis with a sensitivity in the order of 1 ng boron per sample. Over a period of time track-etching techniques have been developed to satisfy such requirements. Fleischer found that cellulose esters could detect charged particles selectively,[11] thus making it possible to detect alpha particles from the $^{10}B(n,\alpha)^7Li$ reaction in the presence of γ rays and thermal neutrons. Lelental used this technique to measure ng amounts of boron per cm^2 that had been vacuum deposited on tape,[12] while Thellier and colleagues evaluated Li distribution in the brain.[13] More recently, Larsson has used similar techniques to evaluate boron distribution in tissue sections[14] and Gabel *et al.* have reported a method for evaluating ng amounts of boron in 0.5 μl droplets.[15] The latter procedure has been modified as detailed here to evaluate similar amounts of boron in lysed cells. Thus the results of cell culture experiments can be evaluated on the basis of known cellular boron content.

II. METHOD AND RESULTS

A. Prompt-γ analysis

At Brookhaven's HFBR, various beam extraction techniques, such as Ni-plated glass wave guides and single crystal Bi "Brockhouse" filters, are used to produce "pencil" beams of pure thermal neutron (~ 2 cm in diameter) that are free of significant contaminations of fast neutrons and γ rays. A stylized configuration of such a facility is shown in Fig. 1. In this geometry, background radiation was low; shielding around the 2×2 in. solid-state detector [pure Ge or Ge(Li)] was minimal, and the detector could be positioned within a few inches of the beam. Background was further reduced by absorbing thermal neutrons scattered off the sample with an enriched $^6Li_2CO_3$ cylinder positioned coaxially with the beam and around the sample. Currently

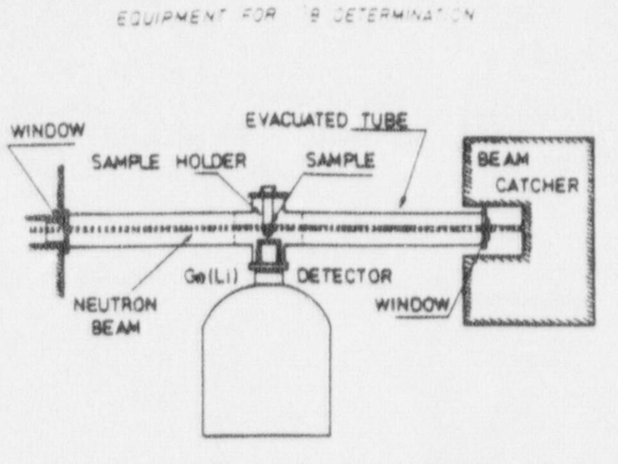

FIG. 1. Diagram of typical geometry used for prompt-γ analysis of ^{10}B at a pure thermal neutron beam from BNL's high-flux beam reactor.

the H-1 beam at the HFBR is being used. The thermal neutron flux density is $\sim 2 \times 10^7$ n/cm^2 s at 60 MW; sensitivity is ~ 700 counts per μg ^{10}B per 200 s, which is $\sim 8\%$ of background over the Doppler-broadened prompt-γ peak at 478 keV. Thus, for the usual 200-s measurement, the error caused by background (1 standard deviation) $\simeq 0.15\,\mu g$ ^{10}B. All measurements are obtained from 1-g samples of tissue and/or water in quartz test tubes. Typical spectra obtained from water and boron standards with a conventional multichannel analyzer are shown in Fig. 2. Background for each run is obtained by summing counts in an equivalent number of channels above and below the prompt-γ peak. Counts from the 478-keV prompt-γ ray (with background subtracted) are then corrected for possible fluctuations in reactor power, counting time, etc., by normalizing to counts from the 2.23-MeV capture γ ray from hydrogen, as indicated by the equation

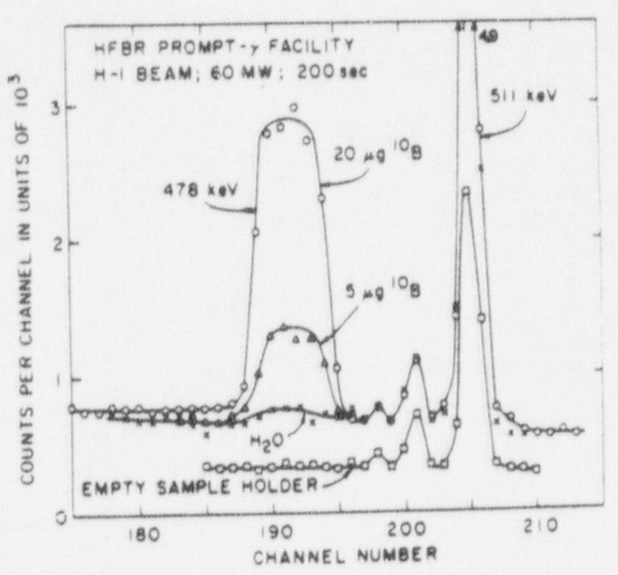

FIG. 2. Spectra obtained from the HFBR, showing background from water samples, and the response from ^{10}B standards in H_2O used for calibration of unknown amounts of boron in various tissues.

$$B_c = (B - L - U)H_{av}/H \quad (1)$$

where B_c = counts from 478-keV Doppler-broadened peak corrected for background and fluctuations in reactor power, sample size, and positioning in beam; B = uncorrected counts in boron "window"; L = lower window (1/2 the width of B), just below the boron peak; U = upper window (1/2 the width of B) just above the boron peak; H = counts from the hydrogen prompt-γ reaction (2.23 MeV); and H_{av} = average of the H prompt-γ reaction for the day's samples.

Absolute calibration for boron was accomplished by constructing standard calibration curves using samples of boric acid ($H_3{}^{10}BO_3$) obtained from the National Bureau of Standards.[7] Five samples varying in ^{10}B contents are used to construct the calibration curve, and the true boron content (B_T) is obtained from

$$B_c = A\,B_T + C. \quad (2)$$

It is assumed that the isotopic concentration of ^{10}B in natural boron is 20% (^{10}B concentration = 18.5% by weight).

Access to the H-1 beam of the HFBR is limited because of requirements of physics experiments. As noted above, problems encountered in clinical application of NCT have demonstrated the necessity of determining boron content of blood, and of tumor also, if possible, immediately prior to the irradiation of patients. Similar measurements during the course of irradiation would, of course, also be desirable. Consequently, an on-line facility for prompt-γ analysis of boron has been constructed at the MRR radial tube (Fig. 3), which is a few meters from the patient treatment facility. In this apparatus, beam extraction, collimation, and shielding have been accomplished by using conventional (low-cost) materials available at any reactor facility. Prompt-γ's from the ^{10}B$(n,\alpha)^7$Li reaction were detected and analyzed as described above. At both the MRR and HFBR, 2×2 in. cylindrical Ge crystals were used so that relative sensitivities were determined primarily by the solid angle subtended by the detector (i.e., by the source–detector distance as given in the text). Specified efficiencies were ~17% (relative to 3×3 in. NaI crystals). Resolution of such detectors can be quite high (<2-keV full width at half maximum). However, crystals used here were selected for economy rather than resolution as the Doppler-broadened 478-keV peak precludes exploitation of the highest resolution available. Thus actual resolution was a few keV (^{60}Co). At a power of 2 MW, the thermal neutron flux density is ~2.7×10^7 n/cm² s, and the sensitivity is ~500 counts per 200 s (16-cm source–detector distance), which is ~4% of background (γ contamination in the beam ≈500 mR min^{-1} or 1.29×10^{-4} C kg^{-1} min^{-1}). Thus at the MRR, background is twice that at the HFBR and the sensitivity is somewhat less. It was found that the signal-to-noise ratio was improved considerably by reducing the angle formed by the incident beam, sample, and detector to slightly less than 90°. Presumably this was a consequence of a reduction in energy of the Compton scattered photons from the sample. Spectra from water and boron standards measured at the MRR are shown in Fig. 4.

B. Boron analysis by track etching

Prompt-γ measurements are ideal for 1-g tissue samples. However, the sensitivity of the method is not adequate for mg samples, such as those obtained from various chemical assays as well as tissue culture experiments. Therefore the technique developed by Gabel *et al.*[15] has been used for these smaller samples. In this procedure, 0.5-μl drops (0.5 mg) of the solution to be analyzed are deposited on cellulose nitrate

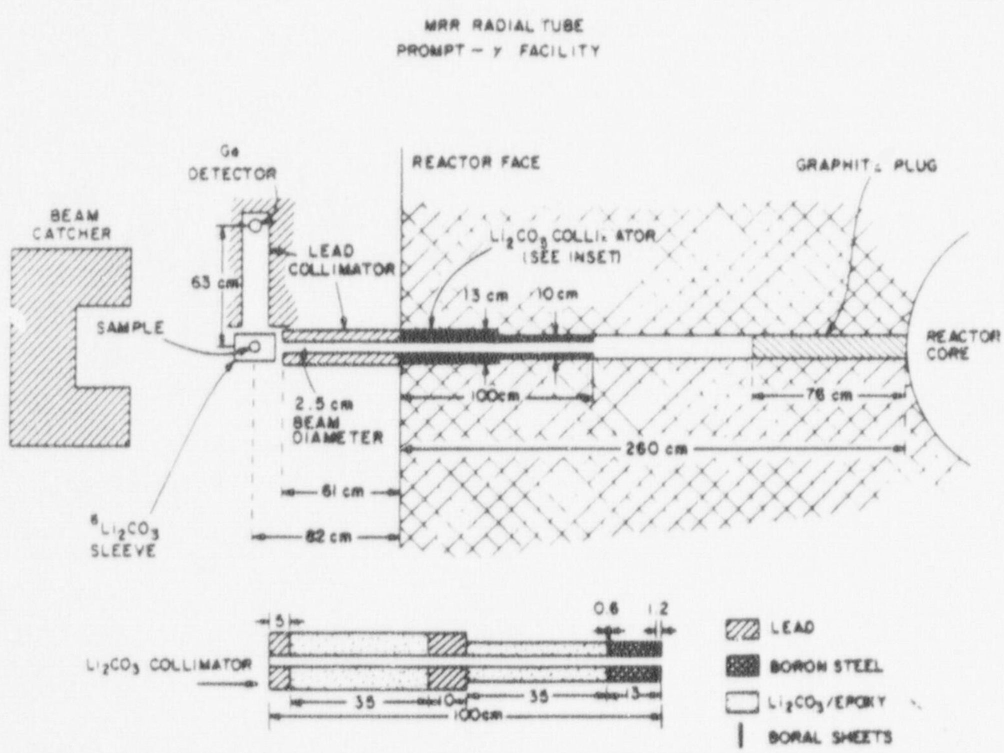

FIG. 3. Geometry of prompt-γ apparatus at the MRR. Source–detector distance has been changed to ~16 cm.

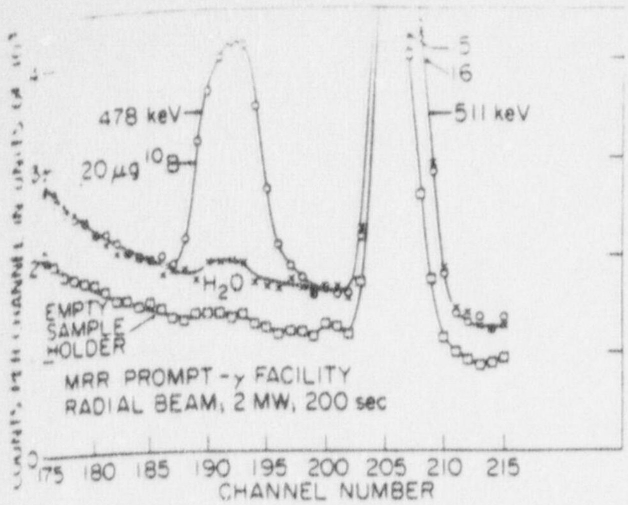

FIG. 4. Spectra obtained at the MRR, showing background from water samples, and the response from ^{10}B standards in water.

track detectors (Kodak Pathé LR115 type 1; 6-μm-thick film on polyester sheets), and irradiated to $\sim 6\times 10^{12}$ n/cm^2 s (1 MW, 300 s), at the patient port of the MRR. This irradiation facility is particularly advantageous as a series of filters between the core and point of irradiation selectively removes fast neutrons and γ rays in an effort to optimize the thermal neutron flux density. For track-etching techniques, proton recoils from the interaction of fast neutrons with hydrogen can cause background interference. The ratio of thermal neutrons to fast neutrons in the above facility is 50. Here, fast neutrons are defined as those with energies above 10 keV. Gamma and fast neutron dose rates are ~15 and 25 rads/min (0.15 and 0.25 Gy/min), respectively, at the point of irradiation, and do not contribute significant background (see Ref. 16 for a detailed discussion of the dosimetry).

Standard solutions of natural boron are deposited on the film to construct a calibration curve, as shown in Fig. 5.

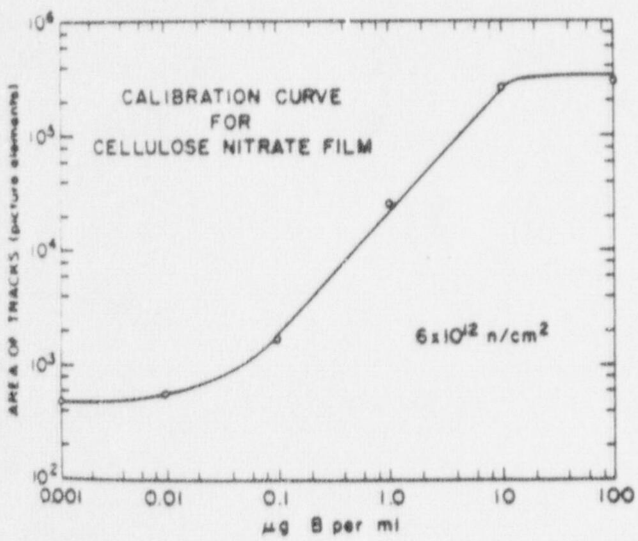

FIG. 5. Calibration curve for cellulose nitrate film obtained with 0.5-μl drops of standard solutions of natural boron and irradiated with 6×10^{12} n/cm^2. Background not subtracted.

Boron is mixed with F-10 cell medium plus fetal bovine serum (FBS), which serves as a matrix to hold the boron while drying. Relatively uniform fields of alpha tracks are obtained in this manner (Fig. 6). Asymmetric deposition of boron is found if water is used as a solvent; the small crystals that were observed when phosphate buffered saline (PBS) + 1 mg protein/ml was used as the solvent[15] were also eliminated.

After irradiation, the film is developed in 10% NaOH at 60 °C for 45 min. Tracks are then counted optoelectronically with a Leitz microscope (4× objective with a green filter) interfaced with a Quantimet image analyzer. Fields obtained from drops with 1 and 10 μg natural boron per ml are shown in Fig. 6 along with background fields. As can be seen from the calibration curve (Fig. 5), the sensitivity extends below 0.1 μg natural boron/ml (0.02 μg ^{10}B/ml). The technique can thus detect ng amounts of natural boron in the 0.5-mg drops used.

Cells are analyzed for boron content following the procedures described for hamster V-79 cells below. It is possible to work with a "slurry" consisting of a maximum of $\sim 5\times 10^4$ cells (~0.05 mg) in 0.5 mg H_2O, so that the lower limit for boron determination in cells is ~1 μg B/g. Due to the residual mass of dried cells in the samples analyzed, correction for self-absorption of the α particle must be made. In order to obtain "standard" curves against which "unknown" cell samples could be compared, known amounts of boron were mixed in with cell slurrys. Boron standard solutions were prepared using unenriched boric acid, water, and 10% fetal bovine serum and diluted to make concentrations of 0.01, 0.1, 1.0, and 10.0 μg B/ml, with a control of 0.0 μg B/ml. V-79 cells were trypsinized, harvested, and counted. Cell pellets consisting of 10^7 cells were obtained. Each cell pellet was frozen for one week, to allow lysing. Then, 100 μl of the boron standard was added to each cell pellet. After vortexing, the standard cell solution was lifted in a ~25-μl capacity Hamilton syringe. Using a 0.5-μl repeat dispenser, droplets were placed on cellulose nitrate film, dried, and irradiated at 1 MW for 60 s (1.2×10^{12} n/cm^2); 0.5-μl droplets of the standard solution (no cells) were also added to film, which was etched as described above.

The calibration curve with cells is shown in Fig. 7 (bottom curve) and compared to a calibration curve obtained from the same film, without cells (upper curve). It is apparent that self-absorption in the cells reduces sensitivity by ~40%. The V-79 cells have a diameter of 12.5 μm,[17] or a volume of ~1000 μm^3. Thus 10^7 cells will have a volume of 10 μl, so that when the 10^7 cells are brought up to a final volume of 100 μl (water + 10% FBS) there will be 5×10^4 cells per 0.5-μl drop. When dried, the residual cellular mass will be ~0.005 mg spread over ~1.3 mm^2 (assuming dried weight ≃ 10 % of wet weight) or ~0.38 mg/cm^2. This corresponds to 3.8 μ of unit density material, which is a sizable fraction of the 10-μ range in tissue of the α particle from the $^{10}B(n,\alpha)^7Li$ reaction, thus producing the self-absorption demonstrated in Fig. 7.

From the above data it is evident that mg amounts of cells can be evaluated for boron content with a sensitivity down to ~1 μg B/g of cells, and that corrections must be made for

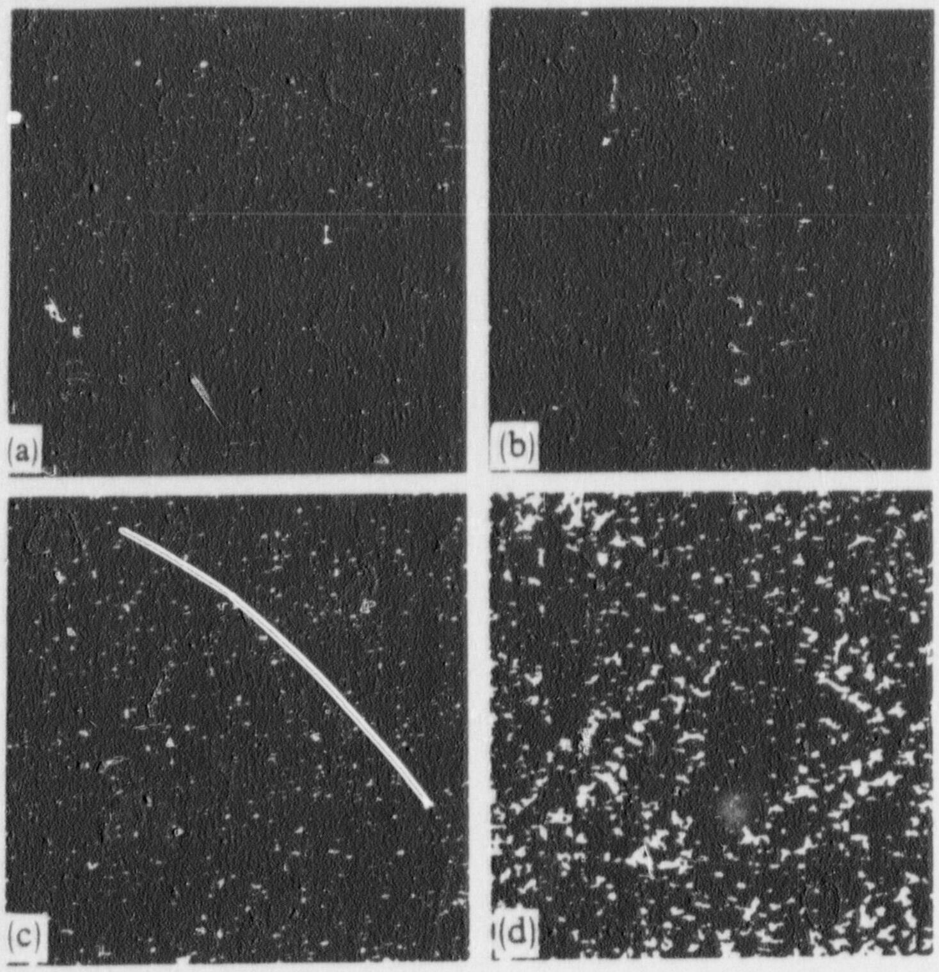

FIG. 6. Cellulose nitrate film irradiated ~6×10^{12} n/cm^2 s at the MRR. (a) Ba ground; (b) background from cell grow media used for suspension of boron sta dards; (c) 1 μg B/ml in 0.5-μl drop; (d) μg B/ml in 0.5-μl drop.

self-absorption in the cells. In view of the experimental variables to be encountered in the lysing of cells and the deposition and drying of the resulting slurry on the cellulose nitrate film, it is clearly desirable to obtain a calibration curve using boron standards mixed with unboronated cells at cell concentrations equivalent to those used with the unknown slurry.

III. DISCUSSION

Endogenous boron levels in humans are ~0.1 μg B/g in blood, for example; (Ref. 18). The lowest level of ^{10}B considered to be useful for therapy is ~1 μg ^{10}B/g tumor.[16] These levels define the limits of sensitivity of interest for prompt-γ analysis of boron in tissue samples, although different requirements may obtain in other fields (i.e., plant physiology, biochemistry, and environmental studies). As described above, the two prompt-γ facilities at BNL provide this capability, with the additional sensitivity at the HFBR used for analysis of low boron concentrations in tissue often obtained with trial compounds synthesized with natural boron.

Since the response from 1 μg of ^{10}B is only a few percent of background, background subtraction is of great importance in the prompt-γ method. In particular, the prompt-γ from Na (472 keV) interferes with B analysis and must be corrected for, to obtain accurate results. Table I summarizes Na and H contents in human tissues. If sample size or exposure tim is monitored by the hydrogen capture-γ ray (at 2.23 meV) care must be taken to account for varying contents of H i different tissues. Most importantly, Na content is seen t vary from 8% to 18%, and thus will be responsible for sign icantly different backgrounds. We have taken multiple read ings of various "blank" tissue samples from our mouse an rat tumor models, and have found that background must b corrected according to tissue type as well as sample size. Fo example, the background produced by Na from gram quan tities of various tissues varies from the equivalent of ~0.5 μ ^{10}B/g in muscle to ~1.1 μg ^{10}B/g for blood or tumor. Th the apparent boron content of each unknown tissue sample i corrected by subtracting the appropriate ^{10}B equivalence o its Na content as derived from both the tissue type and u known sample mass.

Although the relative background obtained at the MRR prompt-γ facility is ~2 times that at the HFBR, the sensitiv ity and background is such that the analysis of 1-μg ^{10}B/ tissue is possible in a single 200-s run. Therapeutic applica tion of the $^{10}B(n,\alpha)^{7}Li$ reaction will require ~15 μg ^{10}B/ in tumor for an epithermal beam, or ~30 μg/g for a therma beam.[16] It would be hoped that boron-10 levels in norma tissues would be a factor of from 5 to 10 less (i.e., 1.5–6 μ ^{10}B/g). Consequently it has been possible to develop an on line capability for boron analysis that will make possible th determination of boron content in blood immediately pri

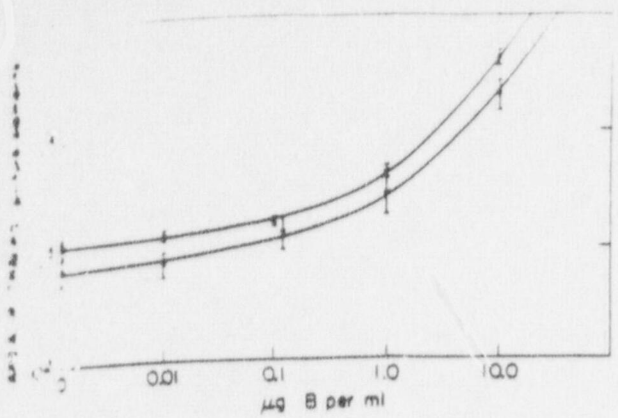

FIG. 7. Calibration curve for cells (bottom curve; 0.45×10^5 cells per drop) compared to calibration curve for standards (no cells; top curve) on cellulose nitrate film.

to, and during, irradiation of humans for NCT. Thus one of the prime requisites for future clinical trials has been met. Furthermore, only conventional shielding materials were used, so that any reactor facility considering NCT should be able to construct a similar facility.

Probably the most difficult way to evaluate a boron compound for potential efficacy in NCT is to try to measure tumor control in animal tumor models following NCT. This necessitates irradiation of multiple animals at multiple dose points followed by prolonged observation, with concomitant experiments on controls in the neutron field as well as in x-ray beams. The development of information about the boron content necessary for successful NCT and about the potential effects due to variation in microdistribution[16,19,20] significantly reduces the necessity for such time-consuming and expensive experiments. The basic information is then available from the boron content of animal tissues measured as described above.

Some basic parameters are more easily obtained from *in vitro* experiments in cell cultures. The biological activity of boronated antibodies and the binding of melanin-affinic molecules as well as that of steroid hormone and nucleoside analogs may all fit into this category. A simple boron analysis on a cell culture of $\sim 10^6$ cells (~ 1 mg) is much simpler

TABLE I. Percent by weight of H and Na in human tissues, from Snyder (Ref. 18).

Tissue	Wt. (g)	H g/tissue	H Wt. %	Na g/tissue	Na Wt. %
Brain	1 400	15[illegible]	[illegible]0.7	[illegible]	0.18
Blood	5 500	550	10.0	10	0.18
Small intestine	640	67	10.5	0.64	0.10
Liver	1 800	180	10	1.8	0.10
Muscle	28 000	2800	10	21	0.08
Pancreas	100	9.7	9.7	0.14	0.14
Lung	1 000	99	9.9	1.8	0.18
Parenchyma	570	55		1.0	
Blood	430	44		0.8	
Spleen	180	18	10	0.22	0.12
Water			11.2		

than the determination of meaningful survival curves in mixed fields of neutrons and γ rays, which would of course also be accompanied by appropriate controls. The track-etching techniques described above provide the capability for this type of analysis, as cells may be lysed and deposited on film for boron analysis.

Additionally, needle biopsies may be obtained from tumor immediately prior to NCT, for boron analysis by track etching. The whole procedure could be completed within 1 h, thus providing a most important parameter for determining the requisite exposure time for tumor control.

ACKNOWLEDGMENTS

We are indebted to Dr. Walter Kane of BNL's Nuclear Energy Department, who served as consultant on the various neutron physics aspects during the course of these experiments. In particular, Dr. Kane designed the Li_2CO_3 collimator used at the MRR, and shown in Fig. 3. We also acknowledge the help given in the early stages of this work by Dr. Kamil Ettinger, University of Aberdeen, Scotland, and the continuing advice and assistance provided by Dr. Lucien Wielopolski and Dr. Daniel Slatkin, Medical Department, BNL.

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[b] Presented in part at the First International Symposium on Neutron Capture Therapy, Oct. 1983; Brookhaven National Laboratory Report No. 51730.

[c] Fullbright fellow, 1984–85; work supported in part by Deutsche Forschungs-gemeinschaft.

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DEPARTMENT OF HEALTH & HUMAN SERVICES

Public Health Service
National Institutes of Health

Memorandum

Date

From

Subject

To

Experimental protocols for irradiation experiments of Oct. 21st and 28th in AFFRI.

Assuming a positive response on the part of the safety committee meeting on the a.m. of the 19th OCT., with regard to the use of cellulose nitrate film, the plan for Friday 21st of Oct. is as follows:

We will undertake 7 runs.
The following are the desired fluences:

1×10^{12} n/cm^2
5×10^{12} n/cm^2
1×10^{13} n/cm^2

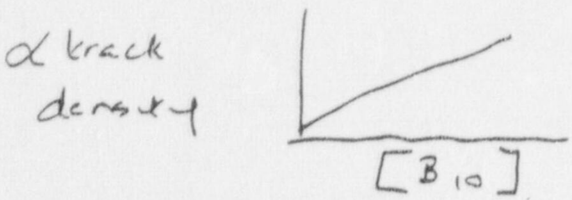
These will be undertaken twice, once in the Carmichael set-up,and once in the Loughlin/ Matson set up of the 8th and 15th of Dec. 1987, using either 9" D_2O and 0" H_2O or 3" D_2O and 3" H_2O
(to be discussed with Captn. Manson)
The seventh run is intended to provide a 'worst case sample' for safety and should therefore be run at the highest fluence, and should probably be run first.

The film will be coated with tissue slices 10 microns thick, containing a uniform quantity of ^{10}B enriched Bo, known to produce appropriate pits or etchings on the film.
The film will be mounted in either cardboard or plastic card mailers, which should hold it upright in the beam, and which will be tailored to suit the appropriate set up.

The experiment is designed to examine the system and to check that the fluence is uniformly distributed in both designs, to identify the optimum fluence for irradiation, and to subsequently develop the film and devise an optimal numerical system for quantification of pit density.

The experiment planned for Friday Oct 28th will utilize the optimum design and fluence of those used the previous week.

We will uniformly irradiate tissues impregnated with varying concentrations of Bo compound, in order to devise a dose-response curve



We would anticipate six-eight runs.
The optimum fluence and design will be selected on the basis of the previous weeks experiments.

November 2nd will be devoted to cell irradiation, assuming the incubator has been installed in the prep. area.
Two tumors will have been excised by surgery, cultured, and will be irradiated in the presence of the Boron compound bound to the appropriate hypothalamic factor.

We will need four runs at 500 watts x 20 mins, and four other runs, 2 at 500 watts for 5 mins and 2 at 2KW for five mins.