TECH SPECS
DOCKET 50-170

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

DIR

26 MAR 1991

SUBJECT: Submission of Annual Report

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

Dear Sir:

Attached please find the 1990 Annual Report for the AFRRI TRIGA reactor facility, submitted as required by license R-84, facility docket 50-170.

Should you need any further information, please contact the Reactor Facility Director, Mr. Mark Moore, at (301) 295-1290.

Attachment
as stated

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

CY Furn:

U. S. Nuclear Regulatory Commission
ATTN: Mr. Alexander Adams, Mail Stop 11B20
Washington, DC 20555

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King of Prussia, Pennsylvania 19406

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1990
ANNUAL REPORT

ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE

REACTOR FACILITY

1990

ANNUAL REPORT

REACTOR FACILITY

ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE

Prepared under direction of:

MARK L. MOORE
Reactor Facility Director

Approved by:

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

1990 ANNUAL REPORT

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Introduction:

As 1990 began, the reactor facility staff continued the process of testing and preparing the new microprocessor-based instrumentation and control console for installation. After Nuclear Regulatory Commission (NRC) approval of the installation through Amendment 19 to the facility operating license (Attachment E), the reactor staff and engineers from General Atomics installed the new control console, control rod drives, and related processing electronics during an extended maintenance shutdown period of October-December 1990. Significant post-installation testing and calibration was performed to verify the validity of the original safety analysis (Attachment A) and the reactor was declared fully operational in December 1990.

The B-ring fuel element that developed a bulge in the cladding during 1989 was placed in a sealed storage container and an amendment to the facility operating license (Attachment F) was obtained to eliminate yearly measurements of that element. After a survey of numerous other TRIGA reactors to determine the extent of cladding degradation experiences associated with pulsing, the Reactor Facility Director authorized a return of the pulse size limit from $1.50 to $2.00 with further increases expected following additional study.

The Reactor Facility was inspected by the Defense Nuclear Agency Inspector General from 25 to 27 September 1990. The inspection found that the AFRRI TRIGA Reactor Facility was satisfactory in organization, operations, material handling, and safeguards. There were no reportable deficiencies found in the maintenance records, operational log books, or training records. During the inspection, seven minor deficiencies were documented. None of the cited deficiencies were considered significant and, neither singly nor in aggregate, impaired the performance or degraded the safety of the TRIGA nuclear reactor operations. The Reactor Facility was also inspected by NRC personnel from Region I 20-22 February and 28-30 November. No violations were identified during either inspection.

Changes were made to the procedures and facilities during 1990. These changes were supported by an extensive safety review process in accordance with the provisions of 10 CFR 50.59. The changes will be discussed fully in sections I and V.

Three trainees were added to the reactor staff during 1990 and four former trainees obtained Senior Reactor Operator licenses. Five Senior Reactor Operators departed during the year. Requests from non-AFRRI investigators continued to supplement the substantial inhouse experimental work load. These experimenters included representatives from the National Institutes of Health (NIH), Smithsonian Institution, Federal Bureau of Investigation (FBI), National Institute of Standards and Technology (NIST), Naval Medical Research Institute (NMRI), and the University of Maryland at Baltimore. The reactor staff was also tasked with providing personnel to assist in conducting inspections of the Fast Burst Reactor facilities at Aberdeen Proving Grounds, Maryland and White Sands Missile Range, New Mexico.

Two Licensee Event Reports were submitted during the year and these are discussed in Section IV.

A revision of the Reactor Physical Security Plan (Attachment C) was approved by

the NRC to coincide with the installation of an upgraded physical security alarm system. This new system exceeds the capabilities of the former system and allows the reactor staff increased flexibility in monitoring and access control.

A major revision of the Safety Analysis Report was completed, approved by the Reactor and Radiation Facility Safety Committee (RRFSC), and submitted to the NRC in April 1990. The facility changes are discussed in Attachment G-3.

A revised Reactor Emergency Plan was approved by the NRC on 27 September 1990 incorporating the implementation guides into a flowchart format covering both radiological and hazardous materials (HAZMAT) scenarios.

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

General Information

1990 ANNUAL REPORT

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 December 1990) are as follows:

Director - Col George W. Irving, III, USAF; BSC.

Scientific Director - Mr. John E. Ainsworth

Chairman, Radiation Sources Department - Capt. C. B. Galley

Chairman, Safety and Health Department - Mr. Douglas Ashby

2. Current key Reactor Operations Personnel:

Reactor Facility Director - Mr. Mark Moore (SRO)

Reactor Operations Supervisor - Mr. Thomas Wright (SRO effective 30 March)

Training Coordinator - 1st Lt Matthew Forsbacka (SRO effective 30 March)

Maintenance/Procurement - Mr. Robert George (SRO effective 30 March)

Administration - MSG Harry Spence (SRO effective 30 March)

3. Senior Reactor Operator Candidates:

Mr. John Nguyen (effective 12 February)
CPT Christopher Owens (effective 2 March)
SFC Michael Laughery (effective 23 March)

4. Departures during CY 1990:

SFC Philip Cartwright (SRO license terminated 23 February)
Ms. Wendy Ting (SRO license terminated 23 April)
SFC Wayne Reed (SRO license terminated 26 June)
MAJ James Felty (SRO license terminated 5 September)
SFC Stephen Holmes (SRO license terminated 29 November)

5. There were several changes to the RRFSC during the 1990 calendar year. Mr. James Caldwell, who had been serving on a provisional appointment as the Special Observer from Montgomery County, was made a permanent Observer on 27 March. Mr. Ron Luerson replaced the retiring Jason Stone as a regular member effective with the September meeting. Also, Mr. John Misner resigned as a Special Member following completion of his special project on 28 March.

The 1990 RRFSC consisted of the following membership to satisfy the Reactor Technical Specifications (as of 31 December):

Chairman - Col. Nicholas Manderfield

Regular Members:

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

Special Member:

Capt. C.B. Galley, CHP (Chairman, Radiation Sources Dept., AFRRI)

Observer:

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

Meetings of the RRFSC were held:

27 March 1990

24 July 1990 (Subcommittee)

11 September 1990

11 December 1990

Section I

Changes to the Facility and Facility Procedures; Surveillance Tests and Inspections

Section I

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

A. DESIGN CHANGES:

1. As previously discussed in the Introduction section, a new reactor control console, control rod drives, and associated electronics were installed during the year as was an upgraded security alarm and access system.

2. A pool water level monitor and readout meter were installed to allow daily measurements of pool water losses. These items are not connected to any reactor systems and have no affect on reactor operations. (Attachment G-1)

3. The primary and secondary air particulate monitors (CAMs) were modified so that alarm of either CAM will now provide various alarm indications that previously were associated with only the primary CAM. Alarm of either CAM will also now result in closure of the reactor room air dampers. (Attachment G-2)

4. An aluminum plate was installed on the floor of Exposure Room #1 near the core projection to minimize damage to the wooden floor from the wheels on various experimental tables. (Attachment G-4)

5. The voltage regulator that previously provided stable current to the old control console was replaced by a larger uninterruptible power supply (UPS) necessary for the new microprocessor-based control circuitry. Also, an UPS was added to the reactor stack gas monitor to ensure continuous air monitoring even in the event of a power outage. (Attachment G-5)

6. Much of the wood (Douglas Fir) on one wall in Exposure Room #1 was replaced to repair damage caused by the effects of age, heat, and radiation. All wood removed was radiologically monitored and disposed of as radioactive waste if required. The design of the exposure room did not change since the wood was replaced with identical materials.

B. PERFORMANCE CHARACTERISTICS:

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

C. ADMINISTRATIVE PROCEDURES:

Two new Reactor Administrative Procedures were approved and implemented during the year.
Administrative Procedure II sets forth the Fitness for Duty program for the AFRRI reactor staff.
Administrative Procedure III clarifies restrictions on personnel passage through the exposure room preparation area to minimize radiation exposure to individuals not directly involved in reactor operations.

1

IMAGE EVALUATION
TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8 3.2 3.6 4.0

150mm

6"

D. OPERATING PROCEDURES:

Numerous changes were made to the operating procedures to improve clarity and to account for the new control console instrumentation. The changes are summarized below and a complete set of current operating procedures is at Attachment B.

1. Procedure I, Conduct of Experiments, was revised to clarify the types of dosimetry required to be worn when entering the exposure rooms or removing a sample from the core experiment tube. A separate change emphasizes that tour pulses may be performed with verbal approval of the RFD or ROS and without a RUR. This is consistent with requirements of the Technical Specifications.

2. Procedure VI, Emergency Procedures, was revised to make requirements and terminology compatible with the new emergency plan approved by the NRC.

3. Procedure VIII, Reactor Operations, was extensively revised as follows:

a. Tabs B, H & I: The Startup, Shutdown and Weekly Checklists were completely revised in conjunction with installation of the new reactor console. Also, the requirement to initial each line on the checklists was removed and one SRO is now indicated as supervising each checklist.

b. Tab B: The air particulate monitor procedure was changed to emphasize the need to close the sampling chamber door after performing the alarm test. This change is in response to the LER discussed in Section IV.

c. Tab F: The square wave procedure was modified to allow both cold-critical and subcritical square waves.

d. Tab G: The pulse procedures were changed to be compatible with the new reactor console instrumentation.

e. Tab K: The stack gas monitor procedure was changed to delete the requirement to maintain historical records of the daily operational check and the 6-hour Ar-41 release printouts. These printouts were formerly used during testing of the stack gas system and they are not used to calculate Ar-41 releases or for any other purpose.

E. RESULTS OF SURVEILLANCE TESTS AND INSPECTIONS:

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

Section II

Generated Energy

Section III

Unscheduled Shutdowns

Section II

Energy generated by the reactor core:

MONTH	KwHr
JAN	3451.2
FEB	6829.2
MAR	3745.1
APR	4219.7
MAY	2784.9
JUN	285.2
JUL	872.0
AUG	2632.5
SEP	2186.3
OCT	2183.0
NOV	4088.5
DEC	3145.1
TOTAL	36442.7

Total energy generated this year: 36442.7 KwHrs

Total energy on core: 702115.1 KwHrs

Total Pulses this year ≥ \$2.00 10 (all equal \$2.00)

Total Pulses on core ≥ \$2.00 4112

Total Pulses on core 9628

SECTION III

Unscheduled Shutdowns:

There were no unscheduled shutdowns during this reporting period.

Section IV

Safety Related Corrective Maintenance

Section IV

Safety-related corrective maintenance:

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

23 Jan 90 Problem: Stack Gas Monitor printer ribbon slipped off printhead resulting in no historical reports printed one weekend,

Solution: Printer ribbon was replaced and backup data recorded from computer memory.

26 Jan 90 Problem: During routine testing of the reactor room air dampers, maintenance personnel determined that air was escaping from the reactor room exhaust duct while the damper was in the closed position.

Solution: Investigation determined that the rod connecting the damper to the air actuating mechanism was hitting the bottom of the duct thus preventing full damper closure. The rod was repaired, the air flow and damper closure circuit were tested, and the system was declared operational. A written Licensee Event Report was submitted to the NRC on 7 February 1990.

21 Feb 90 Problem: While moving a fuel element from the core to storage the fuel handling tool would not disengage from the element.

Solution: The part of the tool above the water level was disassembled to permit repairs. The fuel element remained under water at all times.

19 Mar 90 Problem: A relay control board in the ventilation system burned out causing exhaust fans to shut down and ventilation dampers to close.

Solution: Control board was replaced and system tested. No reactor operations were conducted while the system was not operational.

12 Apr 90 Problem: A backup fuel temperature chart recorder began to fluctuate severely during reactor operations.

Solution: The reactor was secured. An investigation determined that the fuel element thermocouple feeding that recorder had failed. The recorder was connected to a different thermocouple in the same element and the system was tested.

24 Apr 90 Problem: A line voltage spike caused a blown fuse in the primary CAM.

Solution: Fuse was replaced; CAM tested and returned to service.

10 May 90 Problem: Periodic cycling of exhaust fans and closure of reactor room exhaust dampers without signal from CAM discovered during daily startup.

Solution: Switched to manual fan control system and kept dampers closed during operations. Investigation revealed a malfunction in the building-wide control air regulating system which was repaired.

04 Jun 90 Problem: Stack Gas Monitor failed quality assurance test.

Solution: Facility Director and Radiation Safety notified. Electronic and isotopic calibrations completed and system tested.

03 Jul 90 Problem: During startup, determined that Stack Gas Monitor had experienced a power failure over weekend probably due to severe thunderstorms.

Solution: Initialized system and performed successful quality assurance check. Returned unit to service. No operations while system out of service.

17 Jul 90 Problem: Power monitoring channel Safety 1 on reactor console failed during startup testing.

Solution: Burned resistors on circuit board replaced. Channel tested and returned to service.

01 Aug 90 Problem: Console chart recorder would not operate when key inserted into console.

Solution: Loose wire on key switch repaired.

09 Aug 90 Problem: During daily startup procedure the CAM readout meter in the control room would not trigger the audible alarm.

Solution: Investigation revealed that the paper scale on the meter face had separated from its backing and was interfering with free movement of the indicator needle. Scale was reglued to backing.

09 Aug 90 Problem: Stack Gas Monitor printer failed.

Solution: Printer was replaced and tested.

05 Nov 90 Problem: Rod positions observed during startup K-excess measurements were significantly different from the previous day.

Solution: An inspection of the core determined that the screws connecting the piston to the connecting rod on the SAFE rod had come out causing the rod to remain fully in the core even when the drive was raised with magnet power applied. The screws were replaced with self-locking screws and the SAFE rod system tested.

05 Dec 90 Problem: During startup testing the 0.5cps rod withdrawal prevent did not occur when the source was removed from the core.

Solution: The circuit was checked and a loose wire was found between two DAC terminal boards. The wire was repaired and the RWP tested.

In addition to the LER submitted for the 26 January malfunction, a second LER was submitted on 29 March. During a routine daily check of the primary air particulate monitor (CAM), the staff determined that the door to the detection chamber was partially left open during that morning's startup procedure. The secondary CAM operated correctly during the entire time the door on the primary CAM was open and an analysis of the charts from both CAMs as well as pool water samples showed no release of fission fragments. The ventilation system operated correctly at all times. Corrective actions included modifications to the CAM daily test procedure as discussed in Section I.D.3.b. and modification of the secondary CAM as discussed in Section I.A.3.

Section V

Facility Changes and Procedure Changes as Described in the SAR; New Experiments and Tests.

Section V

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

A. As previously discussed in the Introduction section, a new reactor control console, control rod drives, and associated electronics were installed during the year.

B. A pool water level monitor and readout meter were installed to allow daily measurements of pool water losses. These items are not connected to any reactor systems and have no affect on reactor operations. (Attachment G-1)

C. The primary and secondary air particulate monitors (CAMs) were modified so that alarm of either CAM will now provide various alarm indications that previously were associated with only the primary CAM. Alarm of either CAM will also now result in closure of the reactor room air dampers. (Attachment G-2)

D. An aluminum plate was installed on the floor of Exposure Room #1 near the core projection to minimize damage to the wooden floor from the wheels on various experimental tables. (Attachment G-4)

E. The voltage regulator that previously provided stable current to the old control console was replaced by a larger uninterruptible power supply (UPS) necessary for the new microprocessor-based control circuitry. Also, an UPS was added to the reactor stack gas monitor to ensure continuous air monitoring even in the event of a power outage. (Attachment G-5)

F. Much of the wood (Douglas Fir) on one wall in Exposure Room #1 was replaced to repair damage caused by the effects of age, heat, and radiation. All wood removed was radiologically monitored and disposed of as radioactive waste if required. The design of the exposure room did not change since the wood was replaced with identical materials.

G. An upgraded physical security system was installed. This new system exceeds the capabilities of the previous system and allows the reactor staff increased flexibility in monitoring and access control. Installation of the new system did not require any major changes to the Safety Analysis Report.

H. There were no new experiments or tests performed during the reporting period that are not encompassed in the Safety Analysis Report. However, a new Routine Reactor Authorization #102 (Attachment D) was approved by the RRFSC to clarify the authorization to irradiate krypton gas (and other inert gases) in the CET. The consequences of an accident involving a krypton gas experiment would not exceed those already described in the SAR for other accidents.

Attachments G-1 through G-5 are a summary of safety evaluations made for changes not submitted to the NRC pursuant to the provisions of 10 CFR 50.59. Each modification was described and qualified using Administrative Procedure I, Facility Modifications. This procedure utilizes a step-by-step process to document the fact that there were no unreviewed safety questions, no changes in procedures or facilities as described in the SAR, and no changes to the Technical Specifications.

Section VI

Summary of Radioactive Effluent Released

Section VII

Environmental Surveys

Section VI

Summary of radioactive effluents released:

A. Liquid Waste - The reactor produced no liquid waste during CY 1990.

B. Gaseous Waste - There were no particulate discharges in CY 1990. The total Ar-41 discharges in CY 1990 were 6.748 Curies.

On a quarterly basis:	Jan - Mar 1990	1929.1 mCi
	Apr - Jun 1990	1631.5 mCi
	Jul - Sep 1990	1430.3 mCi
	Oct - Dec 1990	1756.6 mCi

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

Section VII

Environmental radiological surveys:

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

B. The environmental monitoring (dosimetry) program reported the following results for CY 1990.

1. The average background of 19 thermoluminescent dosimeters (TLD) located outside a 15 mile radius of the AFRRI site was determined to be 81.40 ± 2.48 millirem.

2. The average reading of approximately 30 environmental stations located on the AFRRI site was determined to be 0.37 ± 0.54 millirem above background.

3. The single highest environmental station reading was 13.20 ± 10.38 millirem above background. This station is approximately 500 meters from the AFRRI.

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 restricted-access areas.

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

Section VIII

Exposures over 25% 10 CFR 20 Limits

Section VIII

Exposures greater then 25% of 10 CFR 20 limits:

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

Attachment A

10 CFR 50.59 Safety Evaluation for
New Reactor Console

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

ATTACHMENT A

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.

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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

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.

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 10^8 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 AFRRI'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.

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

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 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

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

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

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 operat r via the Servoed Rod Selector Switch on the control console's Mode Control Panel.

In a square-wave operation, the reactor s first brought to a critical condition bel KW, leaving the transient rod partially in the core. All of the ste state instrumentation is in operation. The transient rod is ejected . om 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 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

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

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.

- 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

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 (~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 throug 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 (~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.

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.

- 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.

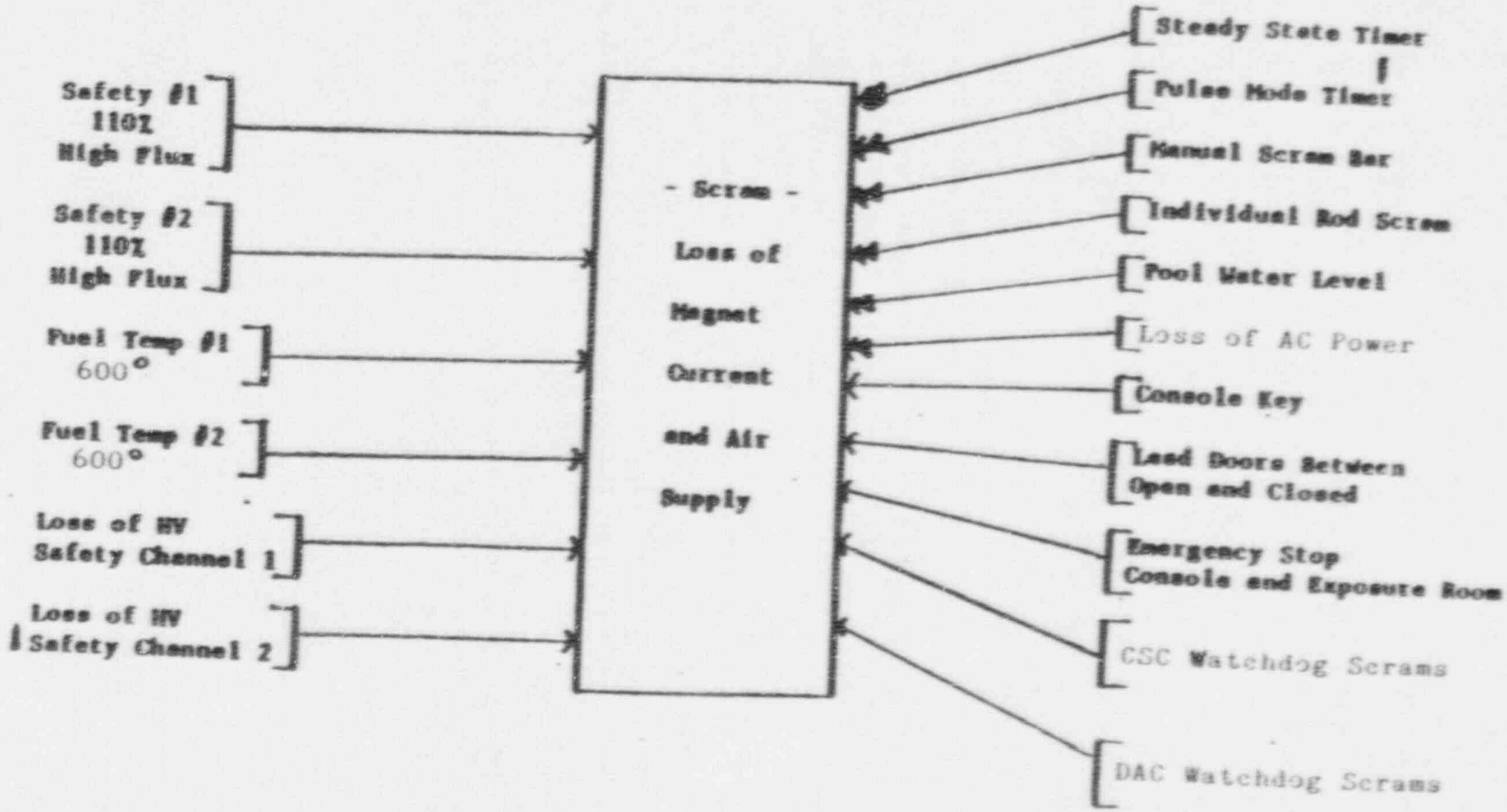
- A loss of reactor pool water which leaves less than or equal to 14 feet of pool water above the core (technical specifications limit) causes a reactor scram. The actual operational limits for the pool water level may be set more conservatively than the technical specifications limit.

- 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

FIGURE #2: AFRRI REACTOR SCRAMS



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).

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.

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

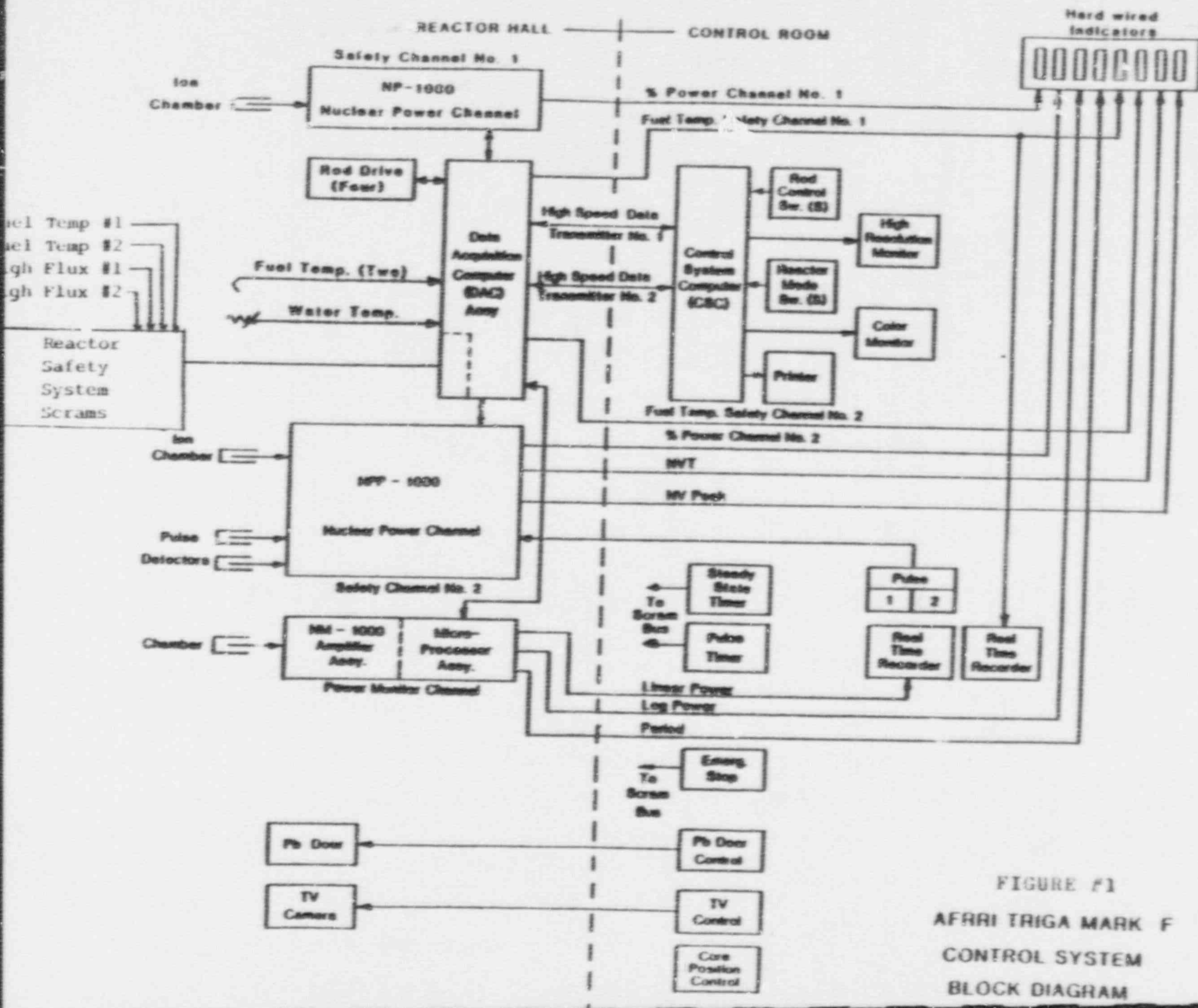


FIGURE #1

AFRRI TRIGA MARK F

CONTROL SYSTEM

BLOCK DIAGRAM

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

and Radiation Facility Safety Committee (RRFSC) for their concurrence tha 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 50.5 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.

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 rodr
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. REACTOR CONTROL COMPONENTS (Section 4.10)

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

PROPOSED SAR WORDING:
"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. STANDARD CONTROL ROD DRIVES (Section 4.10.2)

a. CURRENT SAR FIGURE:
Figure 4-8

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

b. CURRENT SAR WORDING:
"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."

PROPOSED SAR WORDING:
"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. CURRENT SAR WORDING:
"Clockwise rotation of the motor shaft raises the draw tube assembly."

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

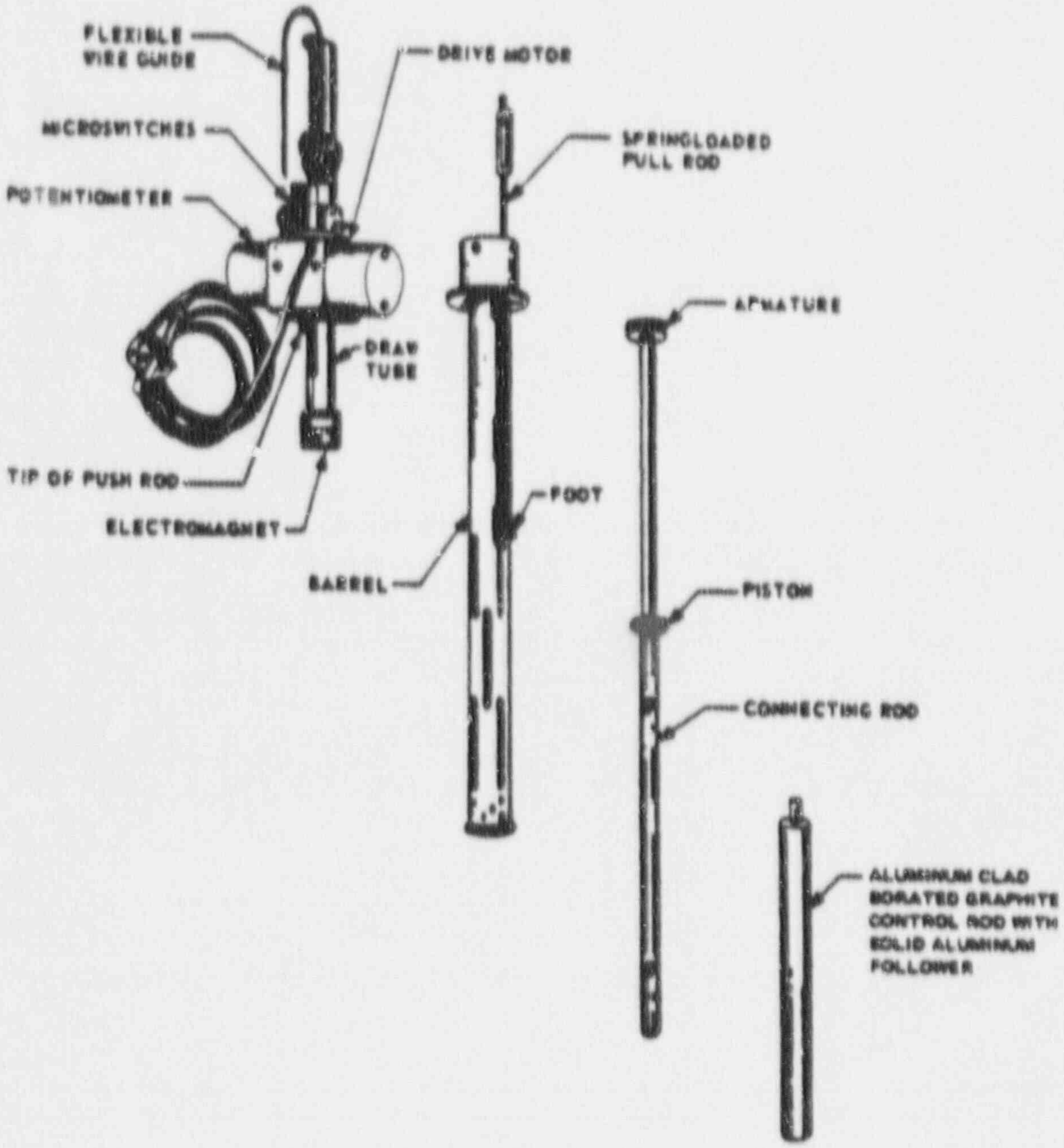


FIGURE 4-8
STANDARD CONTROL ROD DRIVE
FOR SAFETY AND SHIM RODS*

*Regulating Rod Drive Similar Except Drive Motor Contains Tachometer

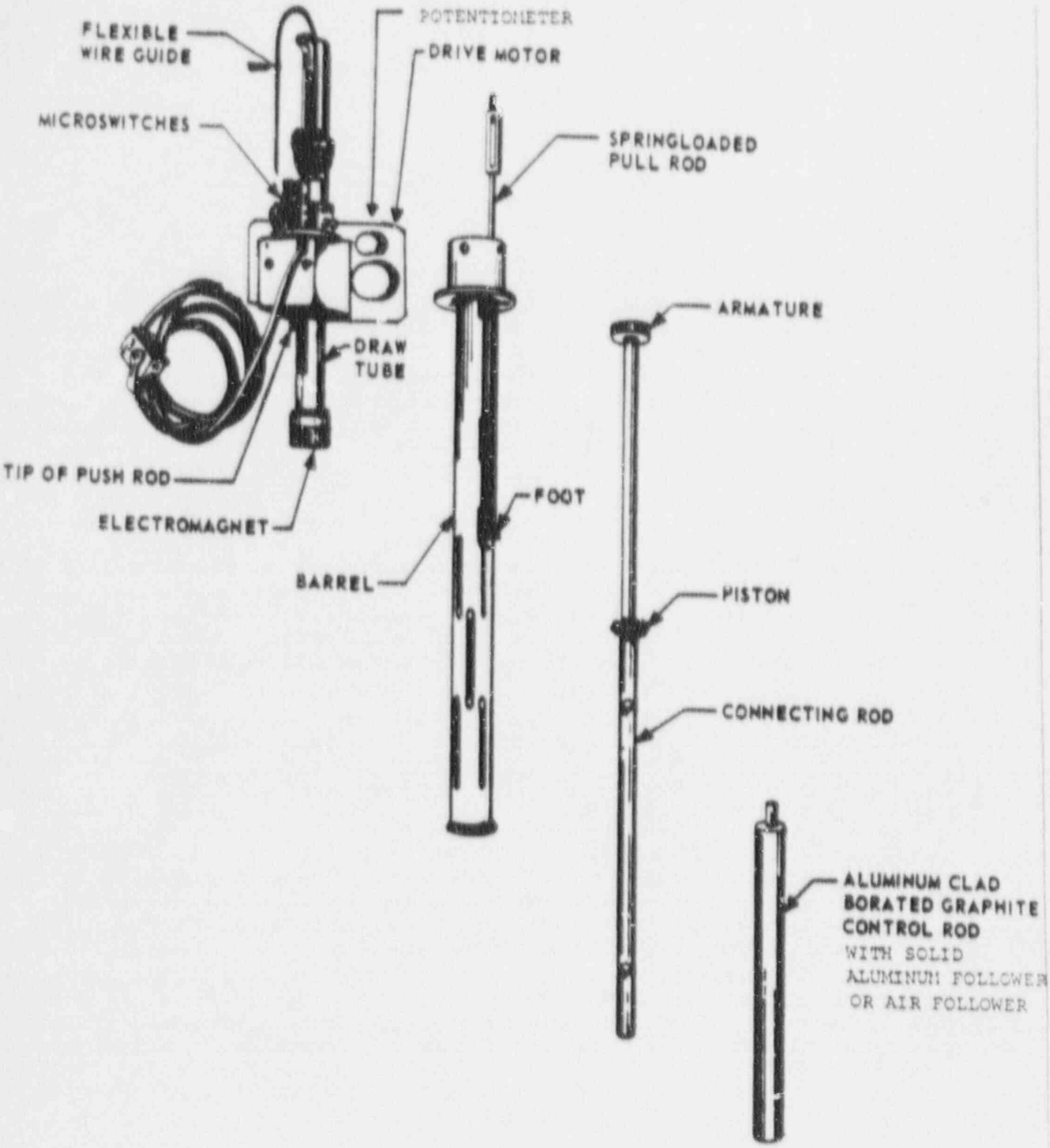


FIGURE 4-8
Standard Control Rod Drives

3. <u>REACTOR INSTRUMENTATION</u> (Section 4.11)

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

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

4. <u>NM-1000</u>

<u>ADD TO THE SAR:</u> (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. <u>MULTIRANGE LINEAR CHANNEL</u> (Section 4.11.1)

<u>CURRENT SAR WORDING:</u>
"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. <u>WIDE-RANGE LOG CHANNEL</u> (Section 4.11.2)

<u>CURRENT SAR WORDING:</u>
"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."

<u>PROPOSED SAR WORDING</u>
"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. <u>HIGH FLUX SAFETY CHANNELS ONE AND TWO</u> (Section 4.11.3)

a. <u>CURRENT SAR WORDING:</u>
"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."

<u>PROPOSED SAR WORDING:</u>
"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. <u>CURRENT SAR WORDING:</u>
"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 bar-graphs 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. <u>SCRAM LOGIC CIRCUITRY</u> (Section 4.14)

<u>CURRENT SAR WORDING:</u>
"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."

<u>PROPOSED SAR WORDING:</u>
-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 it's 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

off Trac
Hood

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

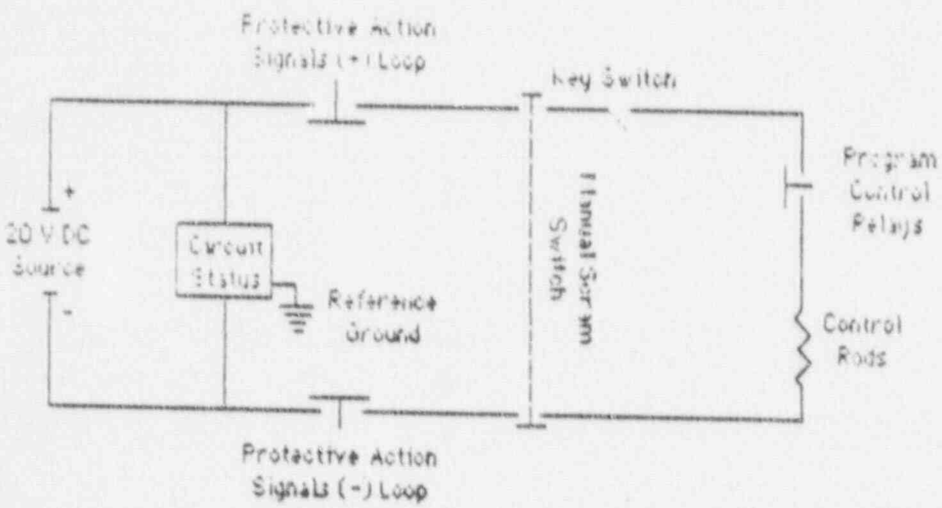
Engineering Science Student

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.

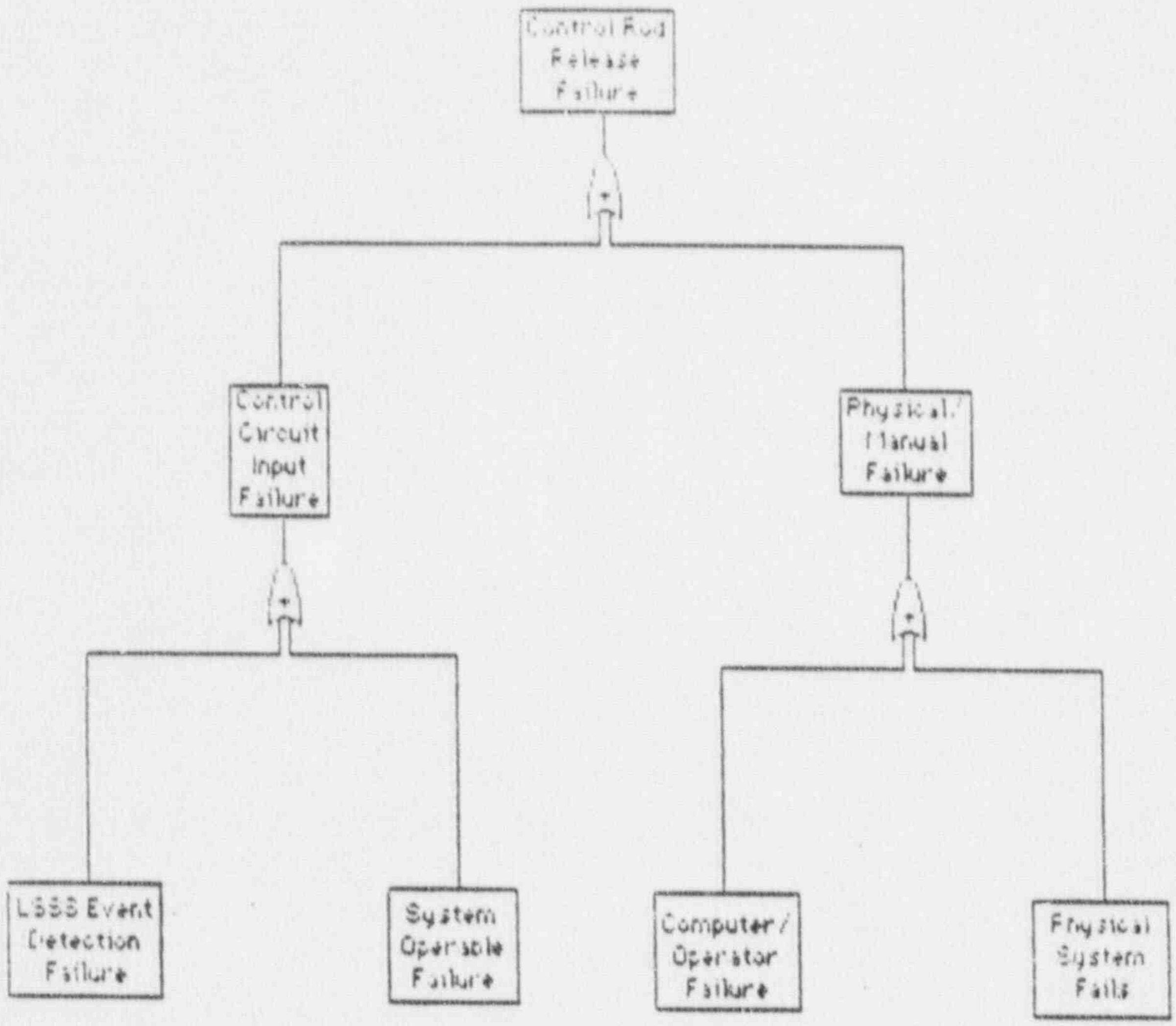
The failure analysis is based on a fault tree approach in which the probability of a particular failure is broken down into component parts which

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

$$P_{Failure} = P_{Signal} + P_{LTTS} + P_{Valve} + P_{[illegible]} \qquad (1)$$

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



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

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.

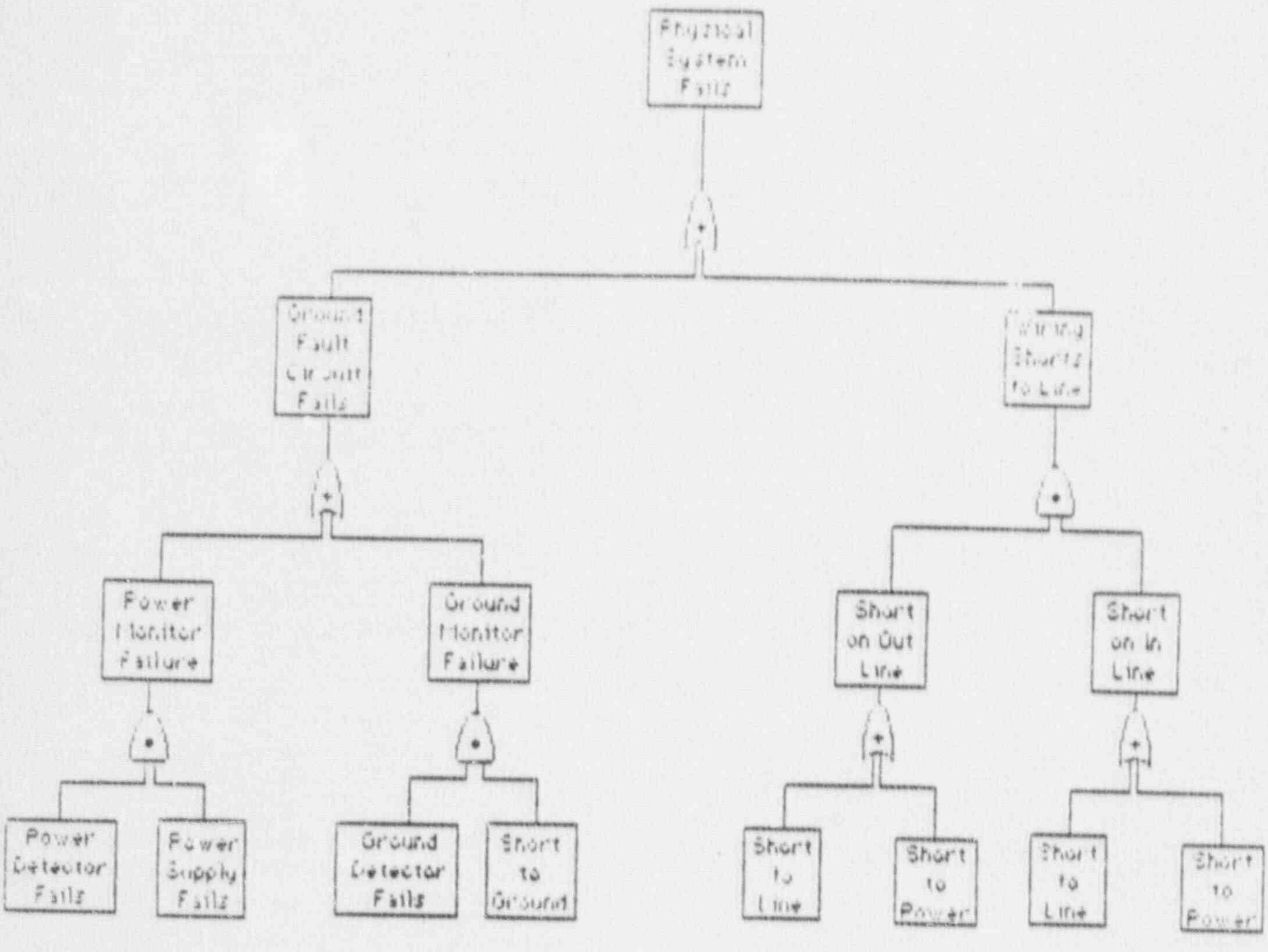
The other branch of the fault tree shows the probabilities associated with faults in the ground detect and voltage detect circuits. For these to cause a potential non-scram situation, however, a short to ground must occur as well as the ground detect failure. Similarly for the voltage detect circuit,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

$$P_{Physical} = P_{Vr\,Fault} * P_{Vr\,Detect} + P_{V\,Fault} * P_{V\,Detect} + (P_{Sh\,Line} + P_{Sh\,Power})^2 \quad (2)$$

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. $P_{Physical}$ 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

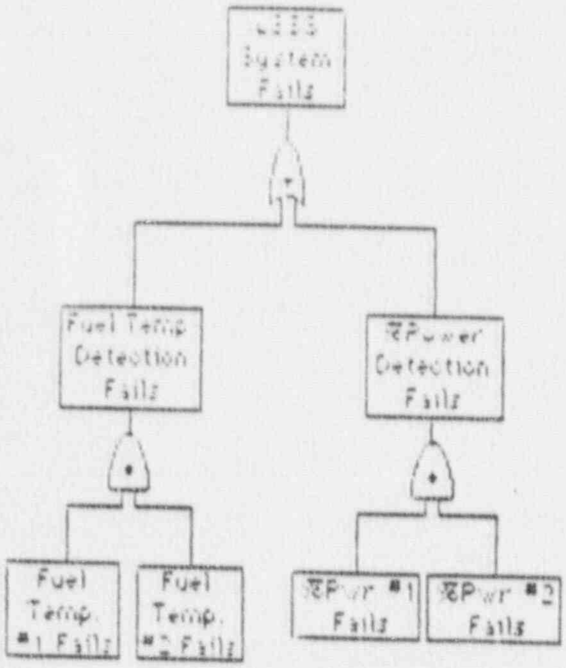
The LSSS consists of the fuel temperature monitors and the percent power monitors. For either high fuel temperature or percent power to cause a non-scram situation, relays on both the supply and return trains must fail. This is because there are two independent fuel temperature monitors, one connected 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:

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

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

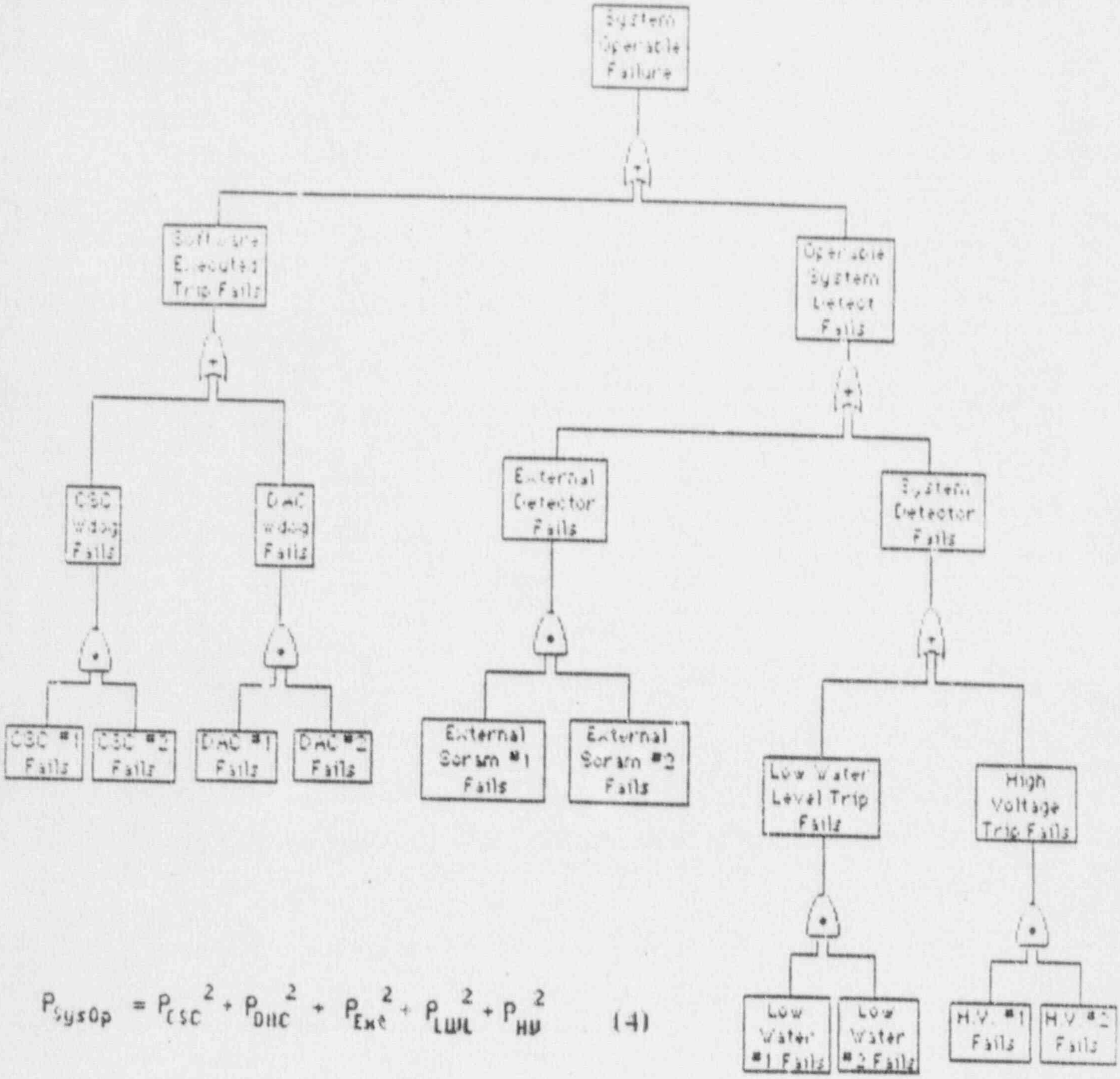
The system operable components are the low-water level, high voltage, watchdog, and external scram relays. Each of these has independent sensors wired into both the supply and return lines and so is a non-single failure mode. The low-water level monitors water level in the tank. High voltage checks the 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.

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

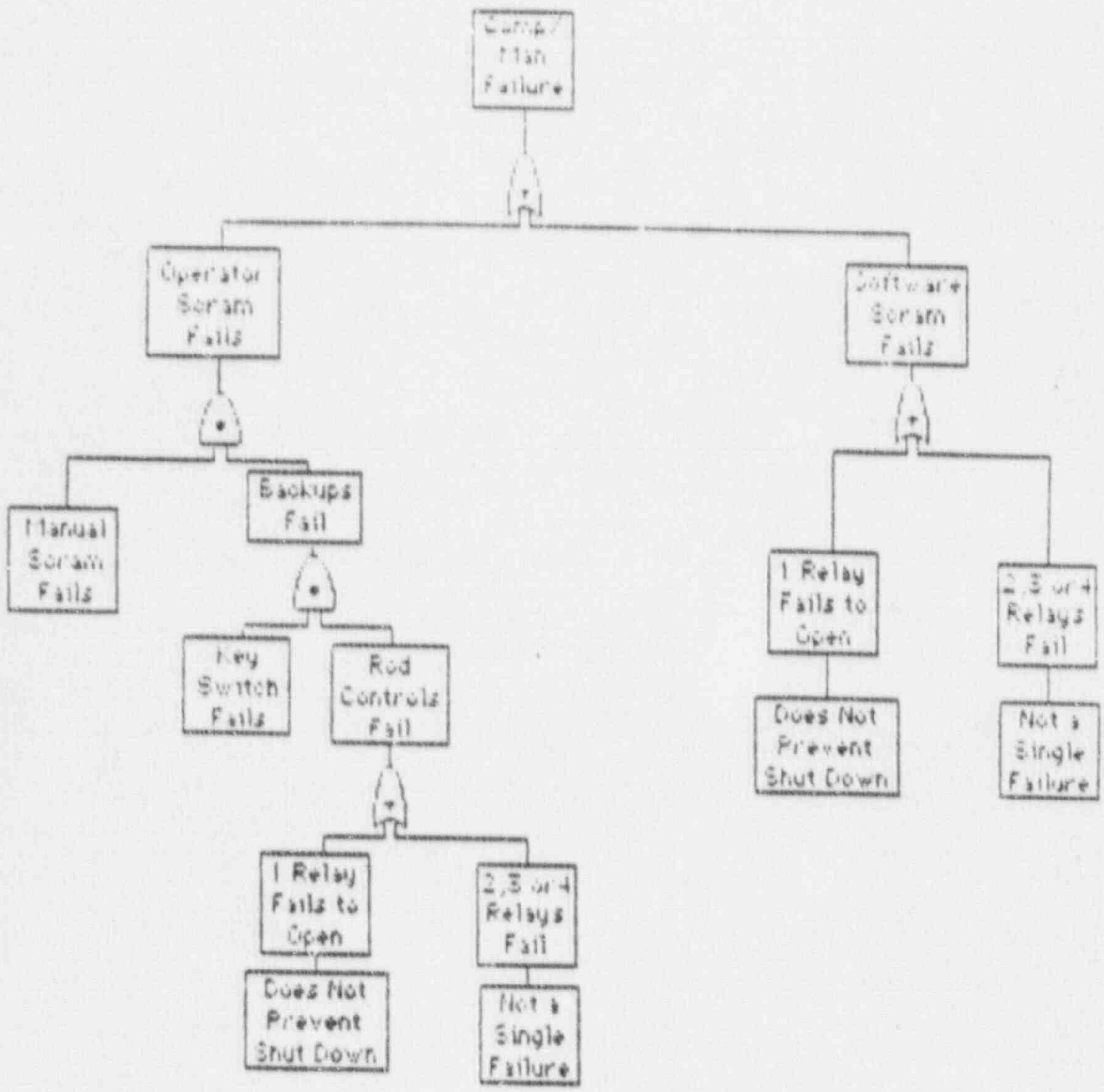
Note that the operator has three independent methods to scram the system, all of which must fail for a non-scram situation to arise. This is highly unlikely as the switches themselves are redundant. The manual scram switch, for example, is wired directly into the rod control circuit at two places. Both 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 CIC. This signal stops the CIC 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 CIC 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



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

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 Military Handbook 217 Revision E. 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

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

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:

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

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 = .034 : Base Switch Failure Rate

Then λ_s = 5 failures/10^6 hrs and P_s = 5×10^{-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 = 1×10^{-7} failures/hour/10 circuit feet. The probability of a short to power is P_{pwr} = 6×10^{-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} = 1×10^{-6} failures/hour.

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

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

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

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

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_b) 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

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

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:

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

$$\lambda_r = .006\,(1.28 * 4.6 * 1.5 * 1.0 * 12 * 1.5)$$

$$\lambda_r = 1 \text{ Failure}/10^6 \text{ hrs}$$

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

$$\lambda_s = .034\,(2.9 * 2.0 * 1.0 * 1.48)$$

$$\lambda_s = .3 \text{ Failures}/10^6 \text{ hrs}$$

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 itself failed and the system did not return from pulse mode. In that event, the system could operate for an extended period without the NP-1000 to provide 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:

$$P_{LSSS} = (P_{f\,Temp})^2 + (P_{\%Pwr})^2 + (P_{\%Pwr} * P_{bypass}) = 3(P_R{}^2) = 3\times10^{-12}$$

Instead of $P_{LSSS} = 2(P_R{}^2) = 2\times10^{-12}$ as before.

This still gives an overall $P_{Failure} = 1\times10^{-11}$ failures/hr, or a mean time

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.

$$P_{\%pwr/hv} = P_{unit1} * P_{unit2} = P_R^2 * P_R^2 = P_R^4 = 1X10^{-24} \text{ Failures/hr}$$

$P_{F.T.} = P_R^3 = 1 \times 10^{-18}$ 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

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

$$P_{BP/NPPFail} = 8 \times 10^{-12} \text{ failures/hr.}$$

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	N	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 affect 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 $6*P_i^2$. 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% dκ/κ ($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 κ = 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% dκ/κ 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% dκ/κ ($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% dκ/κ ($2.21). The total scram activity is, then 1.55% + 2.56% dκ/κ, or 4.11% dκ/κ ($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 dκ/κ 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 .

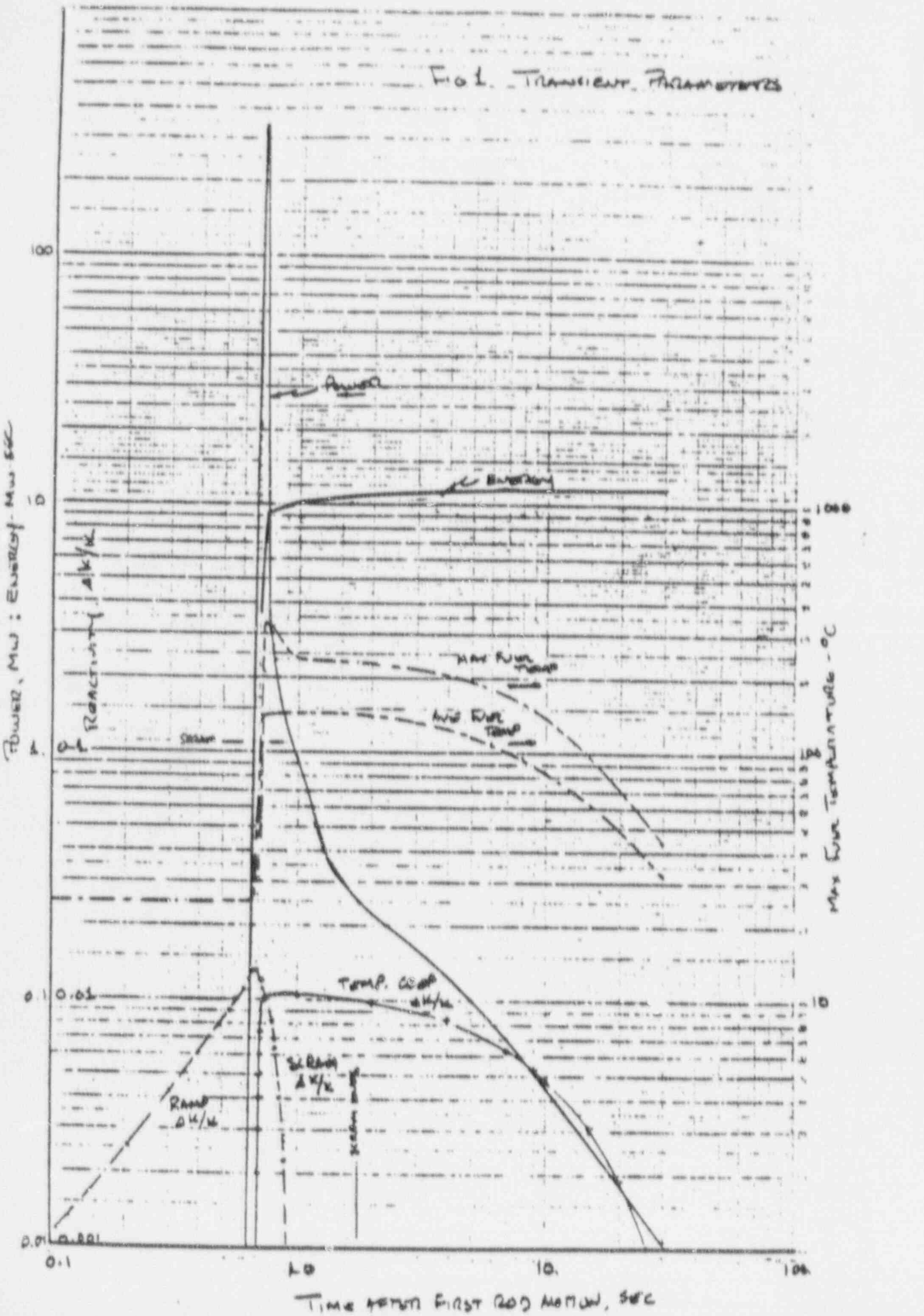


FIG 1. TRANSIENT PARAMETERS
POWER
ENERGY
MAX FUEL TEMP
AVG FUEL TEMP
SCRAM
TEMP. COEF Δk/k
SCRAM Δk/k
RAMP Δk/k
POWER, MW ; ENERGY, MW-SEC
REACTIVITY Δk/k
MAX FUEL TEMPERATURE - °C
TIME AFTER FIRST ROD MOTION, SEC
100
10
1.
0.1
0.01
0.1
0.01
0.001
1000
100
10
0.1
1.0
10.
100.

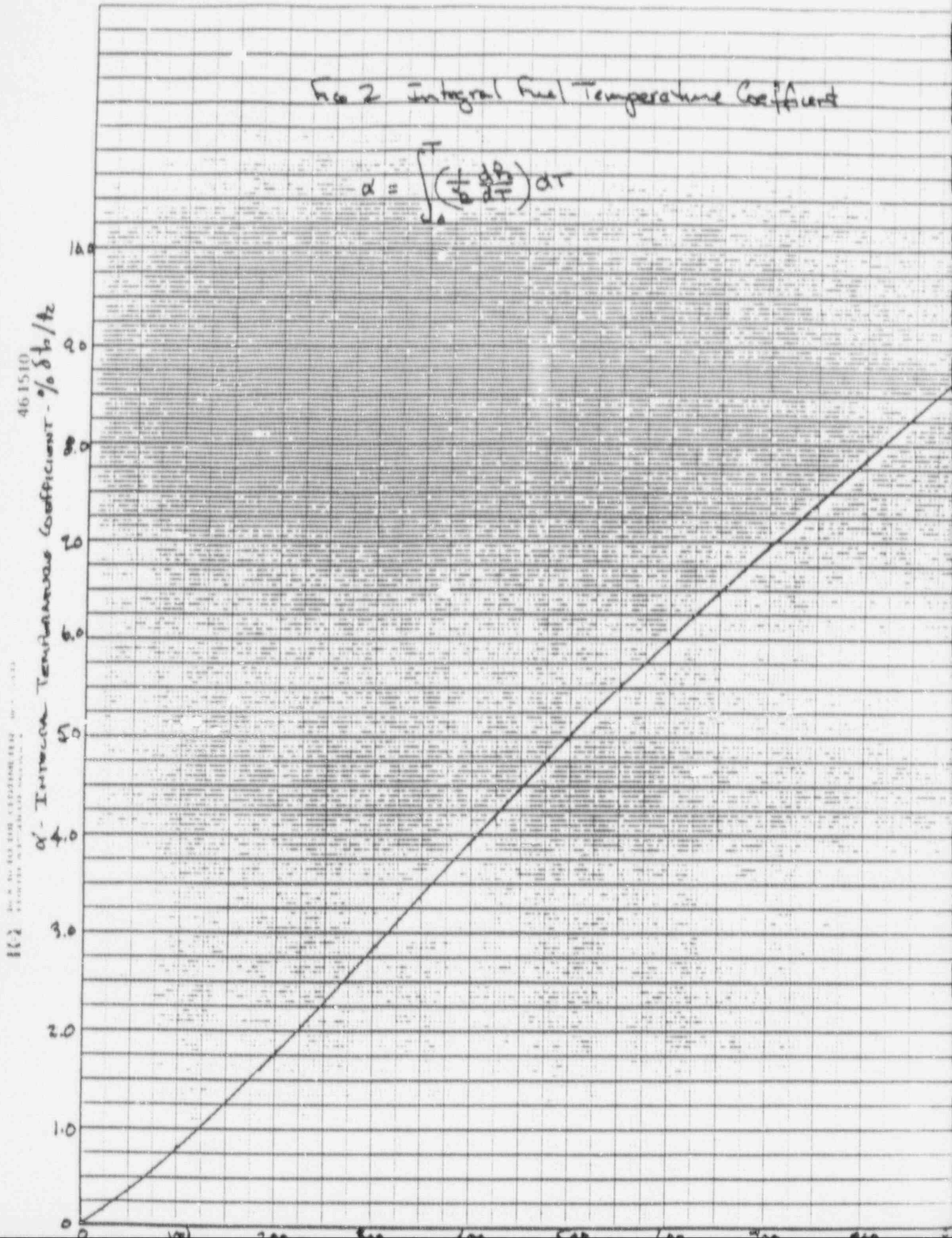


Fig 2 Integral Fuel Temperature Coefficient
$\alpha = \int^{T} \left(\frac{1}{k} \frac{dk}{dT} \right) dT$
α - Integral Temperature Coefficient - %δk/k
10.0
9.0
8.0
7.0
6.0
5.0
4.0
3.0
2.0
1.0
0
461510

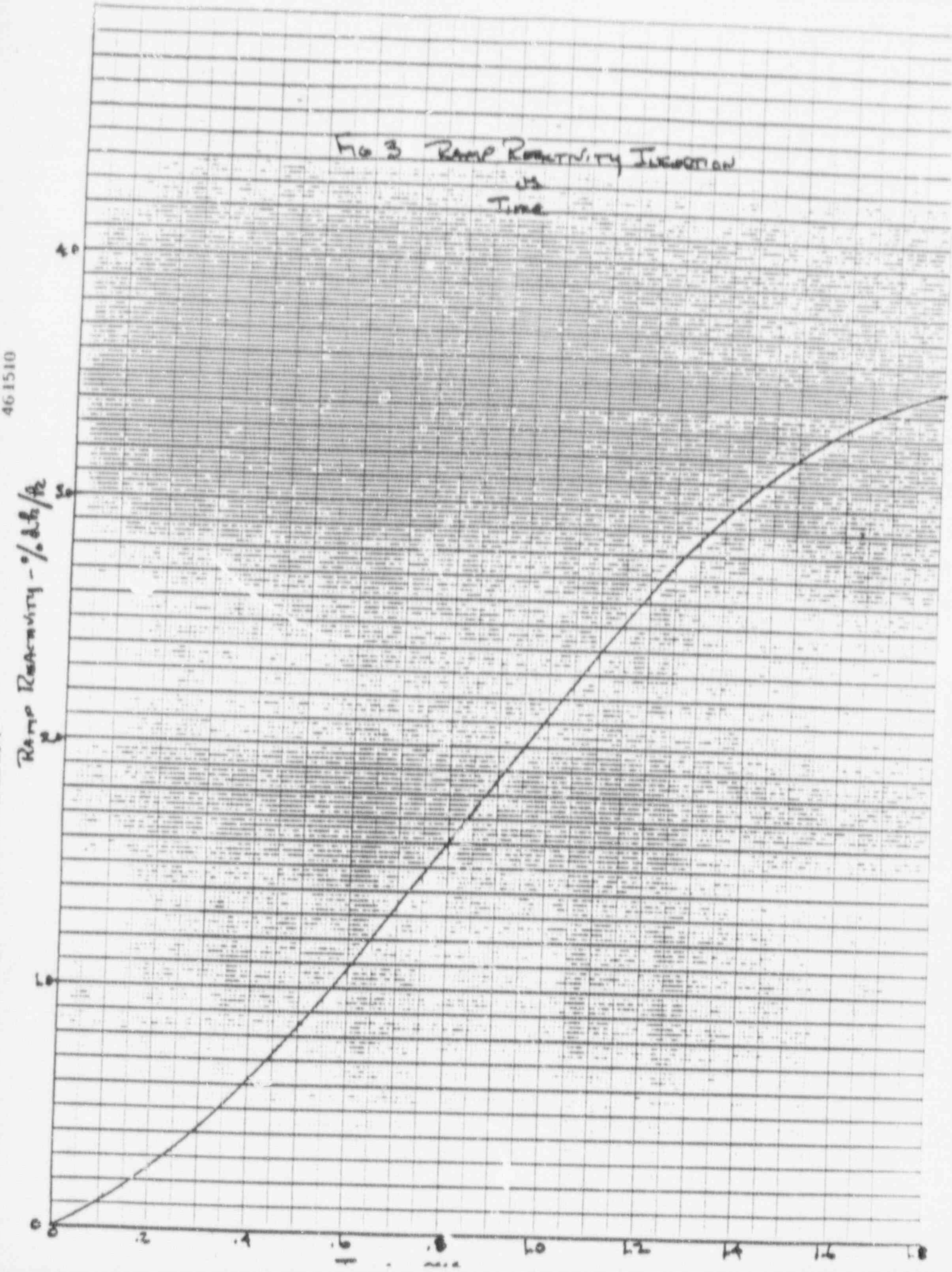


FIG 3 RAMP REACTIVITY INSERTION
VS
TIME
RAMP REACTIVITY - % Δk/k
40
30
20
10
0
0
.2
.4
.6
.8
1.0
1.2
1.4
1.6
1.8

461510

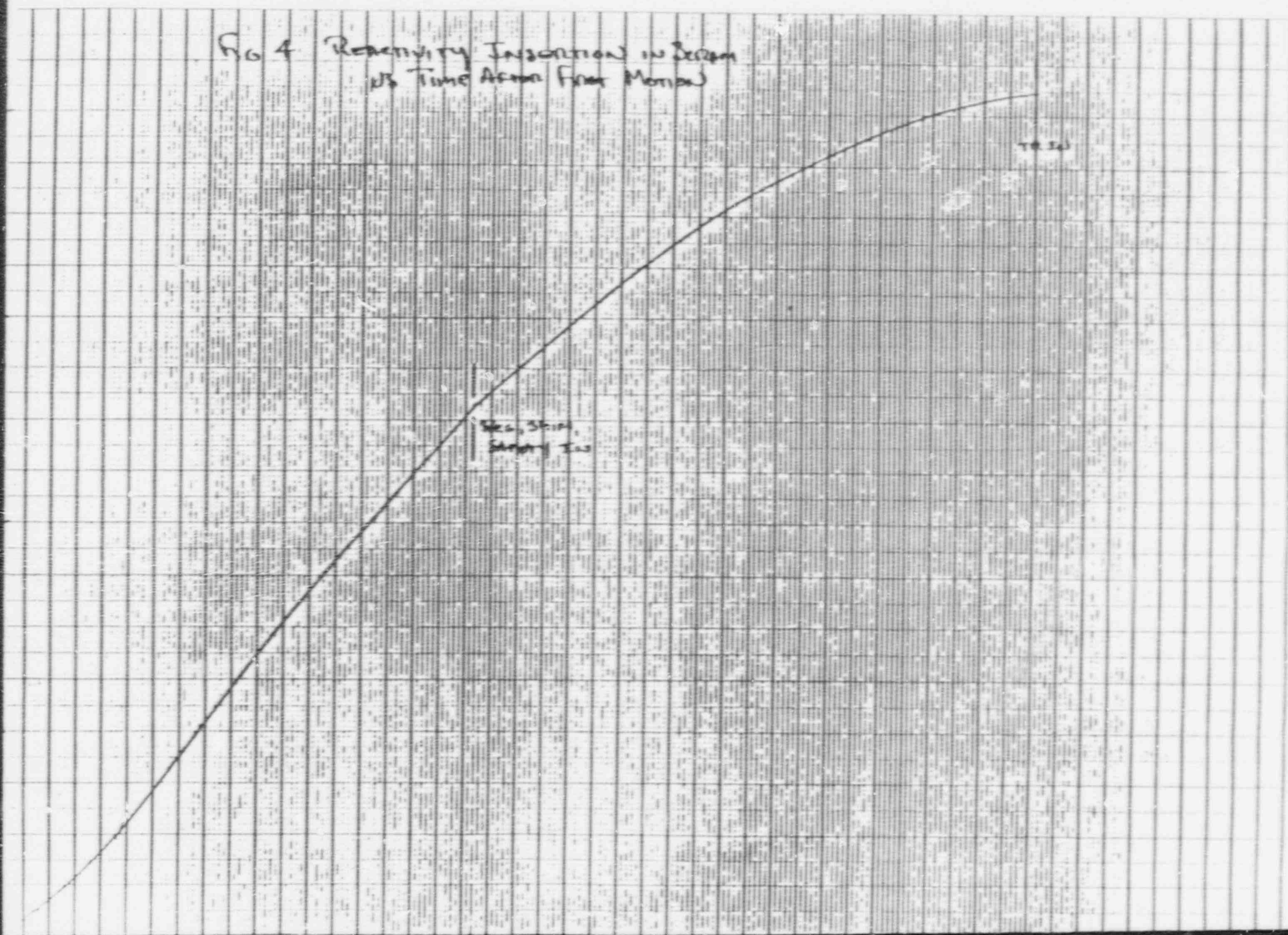


FIG 4 Reactivity Insertion in Scram vs Time After First Motion

Attachment B

Current Reactor Operating Procedures

Revised: 25 July 1990

REACTOR OPERATING PROCEDURES

INDEX

PROCEDURE TITLE	DATE OF LAST REVISION	
0. PROCEDURE CHANGES	~~28 Feb 1989~~	11 SEP 90
I. CONDUCT OF EXPERIMENTS	~~19 Dec 1989~~	4 MAY 90
TAB A: Exposure Room Entry	12 Jun 1990	
TAB B: Core Experiment Tube (CET)	~~13 Jan 1989~~	11 SEP 90
TAB C: Extractor System	Dec 1987	
TAB D: Pneumatic Transfer System (PTS)	Jan 1984	
TAB E: In-Pool/In Core Experiments	15 Dec 1988	
II. REACTOR STAFF TRAINING	Jan 1985	
III. MAINTENANCE PROCEDURES	Jan 1985	
IV. PERSONNEL RADIATION PROTECTION	7 Jun 1989	
V. PHYSICAL SECURITY	15 Dec 1988	
VI. EMERGENCY PROCEDURES	24 Jan 1990	
VII. REACTOR CORE LOADING AND UNLOADING	Dec 1987	
VIII. REACTOR OPERATIONS	27 Mar 1990	
TAB A: Logbook Entry Checklist	6 Feb 1989	
TAB B: Daily Operational Start-up Checklist	~~29 May 1990~~	10 DEC 90
TAB B, Section V, Paragraph 6: APM Procedure	12 Jun 1990	
TAB B1: Daily Safety Checklist	~~29 May 1990~~	10 DEC 90
TAB B1, Section IV, Paragraph 6: APM Procedure	12 Jun 1990	
TAB C: Nuclear Instrumentation Set Points	Apr 1988	
TAB D: K-Excess	Jul 1982	
TAB E: Steady State Operation (Mode I/IA)	~~Jan 1985~~	NOV 90
TAB F1: Square Wave Operation (Mode II - Subcritical)	2 Mar 1990	
TAB F2: Square Wave Operation (Mode II - Cold Critical)	2 Mar 1990	
TAB G1: Pulse Operation (Mode III - Cold Critical)	~~13 Mar 1989~~	11 DEC 90
TAB G2: Pulse Operation (Mode III - Subcritical)	~~13 Mar 1989~~	11 DEC 90
TAB H: Weekly Operational Instrument Checklist	~~23 Apr 1990~~	11 DEC 90
TAB I: Daily Operational Shut-down Checklist	~~20 Jun 1990~~	11 DEC 90
TAB J: Reactor Monthly Usage Summary	~~6 Jul 1982~~	30 NOV 90
TAB K: Stack Gas Monitor Procedure	~~13 Mar 1989~~	29 AUG 90
IX. REACTOR ROOM SAFETY	Dec 1986	

ATTACHMENT B

APPROVED

SWJ

PROCEDURE 0

PROCEDURE CHANGES

Revised 11 Sept 90

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 bock of procedures is reviewed, a single signature block on a title page will substitute for individual review.

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

   a. The change will result in no decrease in the safety of the actions being addressed.
   b. The change will result in no decrease in the efficiency of procedure performance.
   c. The change will not affect the ability of the procedure to perform its intended function.

7. All changes will be staffed to the following:

   a. Chairman, Safety and Health Department (SHD)*
   b. Reactor and Radiation Facility Safety Committee (RRFSC)
   c. AFRRI TRIGA Reactor Facility staff

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

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

APPROVED

Revised: 4 May 1990

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

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

5. ALARA will be practiced during all experiments.

Specific:

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

2. Experiment Review (Processing of RURs):

   a. Check the RUR for completeness (Section I should be filled out).
   b. Forward the RUR to the Military requirements & Applications Department, Operational Dosimetry Division (MRAD) if dosimetry support is required.
   c. Forward it to the Safety & Health Department (SHD) for radiological safety coordination.
   d. Check experiment protocol against reactor authorization.
   e. Fill-in Section II of RUR with special instructions, as appropriate. Assign an RUR sequence number. Write in estimated or measured experiment worth and the core position of the experiment facility to be utilized in the appropriate block (lower left-hand corner of form).
   f. Have the RFD, acting RFD, or ROS review and sign the form.
   g. Ensure the RUR form is placed in the reactor control room prior to the irradiation date.

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

   a. Exposure Room Entry - TAB A.
   b. Core Experiment Tube (CET) - TAB B.
   c. Extractor System - TAB C.
   d. Pneumatic Transfer System (PTS) - TAB D.
   e. In-pool/In-core Experiments - TAB E.

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

5. Attach form to clipboard in the control room.

REVIEWED BY THE REACTOR STAFF

NAME	DATE	INITIALS
WRIGHT	22 OCT 1990	RW
GEORGE	[illegible]	[illegible]
FOARBACKA	23 Oct 90	[illegible]
[illegible]	22 Oct	[illegible]
[illegible]	22 Oct 90	[illegible]
[illegible]	5 Nov 90	[illegible]
NGUYEN	23 oct 90	[illegible]
OWENS	22 Oct 90	[illegible]

Revised: 12 June 1990

PROCEDURE I, 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 separate 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 special access from the RFD will not be given the combination to the prep area.

c. ER ENTRY INSTRUCTIONS - All personnel will:

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

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

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

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

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

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

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

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

requirements.

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

3. SHD EXPOSURE ROOM SURVEY

a. EXPOSURE ROOM CAM: Prior to opening either exposure room, the respective CAM must read 2000 cpm or less, above background. If the CAM reads 2000 cpm or greater above background, change the filter of the CAM. If 10 minutes or more have 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) Readings over 100 mR/hr (closed window) will be reported to the Reactor representative by the SHD monitor. These areas of the exposure room will be identified to the Reactor representative and entry personnel. When appropriate, after consultation with the SHD and Reactor representatives, stay times will be assigned for entry personnel. All personnel entering will be assigned a stay time if they will be working in the high radiation area. AFRRI limits of 100 mR/week and 50 mR/day are to be used as the basis of stay time determinations.

(2) All exposure room entries will be checked by the SHD monitor for compliance with radiation safety aspects of applicable Reactor Use Requests (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) Survey meter readings at the door indicate safe entry conditions (should be less than 1 mR/hr).

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

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

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

5. PERSONNEL PROTECTION PROCEDURES

a. Dosimetry and protective clothing requirements are given in paragraph 2.c, entry instructions.

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

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

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

e. All personnel 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; ensure all obstacles are clear

of the door (including ropes).

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

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

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

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

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

7. ACTIVATED MATERIALS

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

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

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

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

(4) Non AFRRI owned equipment/material: A signed memorandum from the responsible property owner that they understand that byproduct materials generated in excess of their license will be disposed of as 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, rotators, motors, meters, etc., will be stored under the control of the reactor license or the AFRRI byproduct license in warm storage or the prep area. Removal of items from the prep area will only be allowed in accordance with 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 consists 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 activated 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 Principle Investigator
"87" is the calender year
"005" indicates it is the fifth activated item removed from the prep area by Dr. Ledney in 1987

Each item that has been activated must be assigned an activated item number, i.e., if a dog was activated in a plastic 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 all appropriate 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 ensure that no personnel remain in the room. To ensure compliance with the reactor Technical Specifications, the names of these licensed operators present at the exposure room closing shall be entered into the reactor operations logbook and on AFRRI FORM 130. At the completion of the test or experiment, the warning horn shall be reconnected and tested. All actions regarding the warning horn shall in entered in GREEN ink in the reactor operations logbook.

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

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

d. Resecure the prep area on departure.

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Revised: 11 Sep 90

PROCEDURE I, 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 ~~mode~~ made in the operations logbook and the CET logbook.
      (Sp) TPW

3. Rabbit Retrievals:

   a. Ensure that a reactor staff member and a Safety & Health Department (SHD) monitor are present in the reactor room. Any staff member who will be handling the sample following the irradiation may be required to wear a pocket chamber

and appropriate extremity dosimetry depending on the radiation levels of the irradiated sample. If the CET is in the core, a reactor operator must monitor the console during the retrieval.

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

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

d. Drop the extractor head firmly on the rabbit.

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

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

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

h. Stop when 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.

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

4. CET Removal from Core:

a. Complete steps 1a-c above.

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

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

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

e. Secure the CET with cable ties.

f. Secure the CET bracket with the two bolts.

g. Remove the fuel element from the storage rack end 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.

Dec '87 29 SEP 1989 APPROVED

TAB C

EXTRACTOR SYSTEM

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

SPECIFIC:

1. Assembly of the extractor system:

   a. Inside the exposure room:
      (1) Move the inside receiver section into position in front of the core; screw tube supports to the floor 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 i[illegible] final position against the wall (if necessary, add [illegible] 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; remove the tape on the take-up reel.
      (8) Pull back on the drive motor assembly until there is no slack in the cables; tighten the adjustment bolts on the drive assembly.
      (9) Connect the electrical cables to the motor, control unit, and limit switches.

2. Disassembly:

a. Reverse the order of the above with the following changes:
   (1) Before loosening the motor assembly, place tape on the cable drum to keep the cable from moving (ensure the carrier is in the receiver section).
   (2) Before pulling the cable through the tubes, attach a new string to it.
   (3) Leave enough slack for disassembly inside the exposure room.
   (4) Cut the string at the joints in the room and tape the ends to the tubes.

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 p[illegible]r 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 Saf[illegible] ealth Department (SHD) monitor is present.

29 SEP 1989

APPROVED

Jan '84

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.

29 SEP 1989

APPROVED
18 Jan 89 [illegible]

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.

REVIEWED BY THE REACTOR STAFF

NAME	DATE	INITIALS
[illegible]TY	13 Jan 89	JRF
[illegible]GTON	3 Jan 89	GT
[illegible]IGHT	13 Jan 89	PPC
[illegible]	13 Jan 89	ws
[illegible]ND	13 Jan 89	AMM
[illegible]	27 JAN 89	[illegible]
[illegible]S	8 FEB 89	[illegible]
Ting	3 Feb 89	[illegible]
George	3 Feb 89	[illegible]

Stallings

[illegible] 2 Feb 89 [illegible]

Jan '85 29 SEP 1989 APPROVED

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.

Jan '85 29 SEP 1989 APPROVED

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.

29 SEP 1989

APPROVED
3 Aug 89 [illegible]

Revised: 7 JUN 1989

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. 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.

29 SEP 1989

APPROVED
13 Jan 89 [illegible]

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.

VIEWED BY THE REACTOR STAFF

ME	DATE	
	13 Jan 89	JRF
[illegible]TON	3 Jan 89	GT
[illegible]IGHT	13 Jan 89	PPC
	13 Jan 89	[illegible]
	13 Jan 89	[illegible]
	27 JAN 89	[illegible]
	5 FEB 89	SWA
Ting	3 Feb 89	WWT
George	[illegible]	[illegible]
Stallings		
Flossbacher	2 Feb 89	[illegible]

APPROVED

Revised: 24 Jan 1990

PROCEDURE VI EMERGENCY PROCEDURES

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

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

1. Reactor Emergency:

   a. SCRAM reactor.

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

   c. Notify ERT Commander of situation.

   d. Activate emergency organization.

2. AFRRI Complex Emergency Evacuation:

   a. SCRAM reactor.

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

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

   d. Ensure reactor area doors are secured upon departure.

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

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

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

Dec '87

29 SEP 1989

APPROVED

PROCEDURE VII REACTOR CORE LOADING AND UNLOADING

General: Loading and unloading of the reactor core shall be under the supervision of the Reactor Fac ity Director or the Reactor Operations Supervisor. Th e procedures are superseded in the following situations: during CET 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.

APPROVED
mf 3/29/90

Revised: 27 Mar 90

PROCEDURE VIII REACTOR OPERATIONS

General:

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

Specific:

1. The names of the individuals who supervised and performed the daily and weekly checklists will be shown at the top of the checklist. Checkmarks or numbers, as appropriate, will then be entered on each checklist line as that item is performed.
2. Perform reactor Daily Operational Startup Checklist (Tab B), utilizing appropriate nuclear instrumentation set points (Tab C). In the case of no planned operations, a Daily Safety Checklist (Tab B1) may be performed.
3. Record at the 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 calender 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 a Daily Operational Shutdown Checklist (Tab I).
8. Complete the monthly summary (Tab J).

REVIEWED BY THE REACTOR STAFF

NAME	DATE	INITIALS
WRIGHT	28 MAR 1990	[illegible]
TING	29 March 90	WJT
GEORGE	28 Mar 90	RC
FORSBACKA	28 Mar 90	[illegible]
SPENCE	28 Mar 90	[illegible]
HOLMES	30 MAR 90	[illegible]
FELTY	28 Mar 90	JRF
REED	28 Mar 90	[illegible]
~~CARTWRIGHT~~ NGUYEN	28 MAR 90	[illegible]
LAUGHERY	29 March 90	MEL
OWENS	2 Apr 90	[illegible]

29 SEP 1989

Revised: 6 Feb 1989

3 Aug 89

TAB 4

LOGBOOK ENTRY CHECKLIST

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

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

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

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

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

a. BLACK and BLUE-BLACK:

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

(2) Checklist number and completion time.

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

(4) Reactor SCRAM.

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

(a) 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.

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

b. RED

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

(2) Actions which affect reactivity:
(a) Core movement.
(b) Fuel movement.
(c) Control rod physical removal for maintenance.
(d) Experiment loading and removal from the CET, PTS, pool, or core.

c. GREEN

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

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

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

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

PROCEDURE VIII, TAB B, DAILY OPERATIONAL STARTUP CHECKLIST, SECTION V, REACTOR ROOM, LINE 6, AIR PARTICULATE MONITOR PROCEDURE

AND

PROCEDURE VIII, TAB B1, DAILY SAFETY CHECKLIST, SECTION IV, REACTOR ROOM, LINE 6, AIR PARTICULATE MONITOR PROCEDURE.

REVISED: 12 JUNE 1990
APPROVED BY THE REACTOR FACILITY DIRECTOR ______________

CHANGES TO MAKE THIS PROCEDURE APPLY TO BOTH TAB B AND TAB B1 OF PROCEDURE VIII REVIEWED BY THE RRFSC ON 24 JULY 1990.

REVIEWED BY THE REACTOR STAFF:

REVIEWED BY THE REACTOR STAFF

NAME	DATE	INITIALS
WRIGHT	27 July 90	TPW
GEORGE	27 July 90	RG
FORSBACKA	7 Aug 90	[illegible]
SPENCE	6 Aug	[illegible]
HOLMES	1 AUG 90	SWH
LAUGHERY	27 Jul 90	MCL
NGUYEN	27 Jul 90	NJ
OWENS	31 July 90	CGO
Felty	25 July 90	JRF

Revised 10 December 1990

APPROVED

PROCEDURE VIII, TAB B: DAILY OPERATIONAL STARTUP CHECKLIST

Checklist number ____________ Date ____________
Senior SRO Present/On Call ____________ Supervised by ____________
Assisted by ____________

Operators ____________ Time completed ____________

I. EQUIPMENT ROOM (Room 3152)

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

II. LOBBY AREA

Lobby audio alarm turned off .... ________

III. EQUIPMENT ROOM (Room 2158)

1. Prefilter differential pressure .... ________
2. Primary discharge pressure (psi) .... ________
3. Demineralizer flow rates set to 6 gpm .... ________
4. Stack roughing filter (inches of water) .... ________
5. Stack absolute filter (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 rods in tank storage — fuel elements ________ control rods ________
6. Air particulate monitor (CAM)
   (a) Operating and Tracing .... ________
   (b) Alarm test completed, damper closure verified.. ________
7. Door 3162 SECURED .... ________
8. Stack gas monitor quality assurance checked ...... ________

VI. REACTOR CONTROL ROOM (Room 3160)

1. Emergency air system reset . . . . ______
2. Console recorder dated . . . . ______
3. Stack gas and fuel temperature recorder dated . . . . . . ______
4. Logbook dated and reviewed . . . . ______
5. Water monitor box (conductivities must be > 0.5 Mohm-cm)
   (a) Background activity (mA) . . . . ______
   (b) Alarm test completed and alarm reset to 0.5 mA . . . . ______
   (c) Water monitor box conductivity [Mohm-cm]. . . . . . . ______
   (d) DM1 conductivity [Mohm-cm]. . . . . ______
   (e) DM2 conductivity [Mohm-cm]. . . . . ______
6. Stack gas flow rate [kcfm] . . . . ______
7. Stack linear flow rate (ft/min) . . . . ______
8. Gas stack monitor
   (a) Background (cpm) . . . . ______
   (b) Alarm check . . . . ______
   (c) High ala set to 800 MPC Ar-41 . . . . ______
9. Radiation monitors

Monitor	ALARM POINT Functional	READING (mR/hr)	ALARM SETTING (mR/hr)
(a) R-1	______	______	______ 500
(b) R-2	______	______	10
(c) R-3	______	______	10
(d) R-5	______	______	______ 50
(e) E-3	______	______	10
(f) E-6	______	______	10

10. TV monitors on . . . . ______
11. CAM high level audible alarm check . . . . ______
12. Water temperature (inlet) . . . . ______
13. Water level log completed . . . . ______
14. Time delay operative . . . . ______
15. Source level power greater/equal to .5 cps. . . . . ______
16. Prestart operability checks performed . . . . ______
17. Interlock Tests
   (a) Rod raising, SS mode ______
   (b) Rod raising, Pulse mode ______
   (c) Source RWP ______
   (d) Period RWP ______
   (e) 1 kW/Pulse mode ______
   (f) NM-1000 HV ______
   (g) Pool Temp ______
18. SCRAM checks (at least one per rod)
   (a) % Power 1 ______
   (b) % Power 2 ______
   (c) Fuel temp 1 ______
   (d) Fuel temp 2 ______
   (e) HV loss 1 ______
   (f) HV loss 2 ______
   (g) Pool level ______
   (h) Reactor key ______
   (i) Manual ______
   (j) Emergency Stop ______
   (k) Timer ______
   (l) CSC Watchdog ______
   (m) DAC Watchdog ______
19. Zero power pulse . . . . ______

Revised: 12 June 1990

PROCEDURE VIII, TAB B: DAILY OPERATIONAL STARTUP CHECKLIST

SECTION V REACTOR ROOM

6. AIR PARTICULATE MONITOR PROCEDURE

GENERAL: This procedure specifies how to test the CAM to insure proper operation of this monitoring device.

SPECIFIC: This procedure uses a radioactive source to test the alarm set points of the CAM.

A. OPERATING AND TRACING
Observe to see that CAM is operating and tracing.

B. ALARM TEST WITH SOURCE
Open the detector chamber door and slowly bring a radioactive source near the detector. Observe the meter on the front of the CAM. The yellow light will come on at approximately 4000 counts per minute. The red light will come on at approximately 10,000 counts per minute, the alarm will sound and the dampers will close. Reset the alarm, close the chamber door and replace the source in the drawer.

APPROVED
MJ 12/12/90

Revised: 10 December 1990

PROCEDURE VIII, TAB B1: DAILY SAFETY CHECKLIST

Checklist number ______________ Date ______________
Senior SRO Present/On Call ______________ Supervised by ______________
Assisted by ______________

Operators ______________ Time completed ______________

I. EQUIPMENT ROOM (Room 3152)

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

II. EQUIPMENT ROOM (Room 2158)

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

III. PREPARATION AREA

Visual inspection of area .... ______

IV. REACTOR ROOM (Room 3161)

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

V. LOBBY AREA

Lobby audio alarm turned off ______

VI. REACTOR CONTROL ROOM (Room 3160)

1. Emergency air system reset ______
2. Console recorder dated ______
3. Stack gas and fuel temperature recorder dated ______
4. Logbook dated and reviewed ______
5. Water monitor box (conductivities must be > 0.5 Mohm-cm)
   (a) Background activity (mA) ______
   (b) Alarm test completed and alarm reset to 0.5 mA ______
   (c) Water monitor box conductivity [Mohm-cm] ______
   (d) DM1 conductivity [Mohm-cm] ______
   (e) DM2 conductivity [Mohm-cm] ______
6. Stack gas flow rate [kcfm] ______
7. Stack linear flow rate (ft/min) ______
8. Gas stack monitor
   (a) Background (cpm) ______
   (b) Alarm check ______
   (c) High alarm set to 800 MPC Ar-41 ______

9. Radiation monitors

Monitor	ALARM POINT Functional	READING (mR/hr)	ALARM SETTING (mR/hr)
(a) R-1	______	______	______ 500
(b) R-2	______	______	10
(c) R-3	______	______	10
(d) R-5	______	______	______ 50
(e) E-3	______	______	10
(f) E-6	______	______	10

10. TV monitors on ______
11. Source level power greater/equal to .5 cps ______
12. Water temperature (inlet) ______
11. CAM high level audible alarm check ______
13. Water level log completed ______

Revised: 12 June 1990

PROCEDURE VIII, TAB B1: DAILY SAFETY CHECKLIST

SECTION IV REACTOR ROOM

6. AIR PARTICULATE MONITOR PROCEDURE

GENERAL: This procedure specifies how to test the CAM to insure proper operation of this monitoring device.

SPECIFIC: This procedure uses a radioactive source to test the alarm set points of the CAM.

A. OPERATING AND TRACING
Observe to see that CAM is operating and tracing.

B. ALARM TEST WITH SOURCE
Open the detector chamber door and slowly bring a radioactive source near the detector. Observe the meter on the front of the CAM. The yellow light will come on at approximately 4000 counts per minute. The red light will come on at approximately 10,000 counts per minute, the alarm will sound and the dampers will close. Reset the alarm, close the chamber door and replace the source in the drawer.

29 SEP 1989

APPROVED

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~~ 0.5 SEC
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~~ INLET 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

APPROVED 29 SEP 1989

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 ~~10~~ 5 watts. If criticality can not be reached with the REG rod full out, use the TRANS rod to bring to critical.

3. When power is stabilized at ~~10~~ 5 watts, record rod positions in reactor operations logbook, entering all information in red ink.

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

*Note: Use the curves for position ~~567~~ 500 when doing K-excess at ~~200~~ 700

APPROVED
N 6/12/90

Revised: November 1990

PROCEDURE VIII, TAB E: MANUAL AND SERVO MODE OPERATION

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

Specific:

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

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

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

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

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

   c. Select the rods that are to servoed.

   c. Make sure that all rods that will be servoed have been raised approximately 5%.

   d. Enter Servo mode.

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

5. 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.

APPROVED

TAB F-1 SQUARE WAVE OPERATION (MODE II - Subcritical)

OK [illegible]
2 March 90

General:

The square wave mode cannot be used above 500KW.

Specific:

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

2. Given a core position, a final desired transient rod position, and rod curves; determine the transient rod critical position using the following equation (INSERTION ($) cannot exeed 75 cents):

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

3. Apply air to the TRANS rod and raise the anvil to its critical position as determined by the equation in step 2.

4. Bring the reactor cold critical using the remaining three standard control rods; use a rod configuration commensurate with the core position or experimental requirements. Note: A series of repetitive square waves may be run using the same rod positions on the same day without achieving cold critical prior to each square wave.

5. Set flux control dial to desired power level.

~~6. Adjust power range switch to the desired range.~~

7. SCRAM the Transient rod.

8. Raise the Transient rod anvil to the desired FINAL POSITION ($), as required for operation.

9. Allow power to fall to desired power level.

10. Switch into sqware wave mode, making sure the TRANS rod ready light is "on".

    press ready /fire button.

TAB F-1 SQUARE WAVE OPERATION (MODE II - Subcritical) cont.

12. Wait for the power to approach the desired power level, then utilize the REG or TRANS rod to attain desired power level. Turn mode selector switch to manual or automatic if desired.

13. Scram the reactor manually or use the timer, as appropriate; move the core if applicable.

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

APPROVED
2 March 80 [illegible]

TAB F-2 SQUARE WAVE OPERATION (MODE II - Cold Critical)

General:

The square wave mode cannot be used above 500KW.

Specific:

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

2. Bring the reactor cold critical using the three standard control rods; use a rod configuration commensurate with the core position or experimental requirements.

3. Determine TRANS rod anvil setting for desired insertion. Insertion cannot exeed 75 cents.

4. Set flux control dial to desired power level.

~~5. Adjust power range switch to the desired range.~~

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

7. Depress ready /fire button.

8. Wait for the power to approach the desired power level; then utilize the REG or TRANS rod to attain desired power level. Turn mode selector switch to manual or automatic if desired.

9. Scram the reactor manualy or use the timer, as appropriate; move the core if applicable.

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

DISPOSITION FORM

REFERENCE OR OFFICE SYMBOL	SUBJECT		
RSDR	Limitation of Reactor Pulse Value		
TO	FROM	DATE	CMT 1
All Reactor Staff	Reactor Facility Director	27 Mar 1990	

1. Effective this date no pulses greater than $2.00 will be performed without the permission of the Reactor Facility Director. This DF is in effect until recinded by the Reactor Facility Director.

2. This DF superceides DF, DTD 10 OCT 1989, Limitation of Reactor Pulse Value.

M L Moore
Mark L. Moore
Reactor Facility Director

CP: Operation Procedure
JRF VIII, TAB G-1 & G-2

REVIEWED BY THE REACTOR STAFF

NAME	DATE	INITIALS
WRIGHT	28 MAR 1990	RW
TING	29 March 90	WJT
GEORGE	28 mar 90	RG
FORSBACKA	28 Mar 90	[illegible]
SPENCE	28 Mar 90	[illegible]
HOLMES	28 MAR 90	[illegible]
FELTY	28 Mar 90	JRF
REED	29 Mar 90	[illegible]
~~CARTWRIGHT~~ NGUYEN	29 MAR 90	[illegible]
LOUGHERY	29 March 90	MEL
OWENS	2 Apr 90	CWO

AFRRI FORM OCT 80 165 (Formerly DA Form 2496) AFRRI Computer Overprint

APPROVED
MW 12/12/90

Revised: December 11,1990

PROCEDURE VIII, TAB G1: PULSE OPERATION (COLD CRITICAL)

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

Specific:

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

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

3. Stabilize in the manual mode.

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

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

Detector 1 = Pulse Ion
Detector 2 = Cerenkov

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

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

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

9. Reset R-1 and R-5 to their normal alarm points when pulsing operations are complete.

APPROVED

Revised: December 11, 1990

PROCEDURE VIII, TAB G2: PULSE OPERATION (SUBCRITICAL)

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

Specific:

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

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

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

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

4. Stabilize in the manual mode.

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

Detector 1 = Pulse Ion
Detector 2 = Cerenkov

6. Scram the Transient rod.

7. Raise the Transient rod anvil to 100%.

8. Let the power decay to approximately 1 watt or less.

9. Enter Pulse Mode and enter an identifying string at the prompt.

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

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

12. Reset R-1 and R-5 to their normal alarm points when pulsing operations are complete.

APPROVED

Revised: 11 December 1990

PROCEDURE VIII, TAB H: WEEKLY OPERATIONAL INSTRUMENT CHECKLIST

CHECKLIST # ______________________ DATE ______________________

SUPERVISED BY ______________________

ASSISTED BY ______________________ REVIEWED BY ______________________

I. WATER LEVEL INDICATOR

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

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

II. WATER CONDUCTIVITY

List resistivity readings for previous week from daily startup checklists. Determine the average at each point is > 0.5 M-ohm-cm.

	MON	TUES	WEDS	THURS	FRI	AVGE
Monitor Box	____	____	____	____	____	____
DM1	____	____	____	____	____	____
DM2	____	____	____	____	____	____

III. RADIATION ALARMS

A. Test alarm functions for high level and failure

Monitor	Failure alarm functional	HIGH Level alarm functional
R-1	____	____
R-2	____	____
E-3	____	____
E-6	____	____
R-5 (criticality)	____	____
Reactor Room APM	____	____
Gas Stack Monitor	____	____

B. Reset alarms.. ____

IV. TOP LOCK KEY SEALS

Top lock key seals at Security Desk and at LOG verified intact.............. ____

Revissd: 11 December 1990

APPROVED

PROCEDURE VIII, TAB I: DAILY OPERATIONAL SHUTDOWN CHECKLIST

Checklist No. ____________ Date ______________________

Time Completed ____________ Supervised by ______________

Assisted by ______________

____

I. REACTOR ROOM (Room 3161)

1. All rod drives DOWN .... ________
2. Carriage lights OFF .... ________
3. Door 3162 SECURED .... ________
4. Door 3161 locked with key .... ________

II. EQUIPMENT ROOM (Room 3152)

1. Distillation unit discharge valve CLOSED .... ________
2. Air dryer OPERATIONAL .... ________
3. Doors 231, 231A, 3152 and Roof hatch SECURED .... ________

III. EQUIPMENT ROOM (Room 2158)

1. Primary discharge pressure (PSI) .... ________
2. Demineralizer flow rates set to 6 GPM .... ________
3. Visual inspection for lerks .... ________
4. Door 2158 SECURED .... ________

IV. PREPARATION AREA

1. ER 2 plug door CONTROL LOCKED;
   Door closed; and handwheel PADLOCKED .... ________
2. ER 2 lights ON and rheostat at 10% .... ________
3. ER 1 plug door CONTROL LOCKED;
   Door closed; and handwheel PADLOCKED .... ________
4. ER 1 lights ON and rheostat at 10% .... ________
5. Visual inspection of area .... ________

AFRRI FORM 62 (R)

V. LOBBY ALARM

1. Lobby alarm audio ON .... ________

VI. REACTOR CONTROL ROOM (Room 3160)

1. Reactor tank lights OFF .... ________
2. Linear Chart Recorder OFF .... ________
3. TV monitors OFF .... ________
4. Console LOCKED, and console and exposure room keys locked in lock box .... ________
5. Diffuser and secondary pumps OFF .... ________
6. Purification and primary pumps ON .... ________
7. Reactor monthly usage summary completed .... ________
8. Exposure room camera power supply turned OFF .... ________
9. Radiation monitors .... ________

MONITOR	READING	HIGH LEVEL ALARM SETTING (Mr/Hr)	
a. R-1	________	20	________
b. R-2	________	N/A	
c. R-3	________	N/A	
d. R-5	________	20	________
e. E-3	________	N/A	
f. E-6	________	N/A	
g. R-6	________	N/A	

REACTOR MONTHLY USAGE SUMMARY

Month ____________

	K-EXCESS AT INFINITE WATER	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 (200)	INFINITE WATER	ER NO. 2 (700)	OTHER			TOTAL TO DATE	TOTAL
CARRY OVER												
DATE												
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AFRRI Form 7 c(R)
30 Nov 93

APPROVED
TSep

Revised: 29 Aug 1990

PROCEDURE VIII, 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 instrument is used to sample and measure the gaseous effluent in the building exhaust system.

SPECIFIC:

A quality assurance check (QA) is performed daily, prior to reactor operations, as part of the reactor start-up. This check is performed in the following manner:

1. The particulate filter is changed if necessary.

2. The front cover of the detector shield is removed and the check source is inserted all the way in to the face of the detector. The blue alert light should come on as the count rate rises above the alert setpoint. The red high level alert light and bell should come on as the count rate rises above the high level set point. The audible alarm can be silenced by pushing the red button on the front of the SGM cabinet.

3. The detector voltage system set point is checked.

4. The air sampling flow rate (should be greater than 3.5 cubic feet per minute).

5. The counts per minute reading shown on the display or the 1 minute printout should be checked against the plot of counts per minute versus Julian date to determine if it falls within the plus or minus 5% deviation lines for the detector and check source. If it does, the check source should then be removed and the detector cover replaced.

   If the counts per minute consistently fall outside the +/-5% window, it is considered an abnormality and should be reported immediately to the Reactor Facility Director and to the Safety and Health Department.

6. The SGM alarms will be acknowledged by pushing the "ACK" button on the SGM keyboard.

Dec '86 29 SEP 1989 APPROVED

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

AFRRI TRIGA Reactor
Physical Security Plan

The Reactor Physical Security Plan is protected from public disclosure and was previously submitted to the Nuclear Regulatory Commission under separate cover.

Attachment D

Routine Reactor Authorization #102

Routine Reactor Authorization #102

4 September 1990

Introduction: To facilitate the support of Principal Investigators, the Reactor and Radiation Facility Safety Committee is asked to review and approve the routine irradiation of inert gases, with pressurizations up to two atmospheres, in the reactor exposure facilities. This request is similar to Routine Authorization #5 which permits the irradiation of Argon-40 in the exposure facilities and Routine Authorization #100 which authorizes the irradiation of materials with atomic number 1 through 83 for use in construction, support, containment, and shielding. The low pressure of two atmospheres is insufficient to cause damage in the event of container failure.

Irradiated Materials will remain in the physical operations boundaries of the reactor and under control of the reactor staff until released to the Safety and Health Department. Removal and release of any of these materials shall be carried out under the appropriate radiological and safety procedures. The production of byproduct material in a utilization facility is covered under 10 CFR 50 and 10 CFR 30.

Authorization: As permitted by USNRC License, Code of Federal Regulations 10 (10 CFR), Radiation Sources Department Procedures, and appropriate Safety and Health Department procedures, the reactor staff is authorized to irradiate inert gases in the reactor exposure facilities in support of approved experiments.

Prior to irradiation, these gases will be individually reviewed and at least verbally approved by a licensed Senior Reactor Operator to insure compliance with appropriate regulations.

Reactor and Radiation Facility Safety Committee:

4 Sep 90
(D. Ashby)

4 Sep 90
(M. Moore)

R Luersen 11 Sept 90
(R. ~~Larsen)~~
Luersen

11 Sept 90
(M. Voth)

4 Sep 90
(Capt C. Galley, USN, MSC)

for
N.W. Mandefield 4 Sept 1990
COL, USAF, MSC Chairman
Reactor and Radiation Facility
Safety Committee

Approved:

George W. Irving III 4 Sept 1990
Col, USAF, BSC
Director

ATTACHMENT D

Attachment E

Amendment No. 19 to Facility Operating License

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

RECEIVED 8/1/90 cc

July 23, 1990

FILE

Docket No. 50-170

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

Dear Colonel Irving:

SUBJECT: ISSUANCE OF AMENDMENT NO. 19 TO FACILITY OPERATING LICENSE NO. R-84 - ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE (AFRRI)

The Commission has issued the enclosed Amendment No. 19 to Facility Operating License No. R-84 for the AFRRI TRIGA Research Reactor. The amendment consists of changes to the Technical Specifications in response to your submittal dated April 30, 1990 as supplemented on June 19, 1990, and July 13, 1990.

The amendment approves the installation of a microprocessor based instrumentation and control system on the AFRRI research reactor. The Technical Specifications are amended to reflect the new system.

A copy of the related Safety Evaluation supporting Amendment No. 19 is enclosed.

Sincerely,

Alexander Adams

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

Enclosures:
1. Amendment No. 19
2. Safety Evaluation

cc w/enclosures:
See next page

ATTACHMENT E

Armed Forces Radiobiology Research Institute

Docket No. 50-170

cc:

Director, Maryland Office of Planning
301 West Preston Street
Baltimore, Maryland 21201

County Executive
Montgomery County Government
Rockville, Maryland 20850

Reactor Facility Director
Armed Forces Radiobiology Research Institute
National Naval Medical Center
Bethesda, Maryland 20814

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

ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE

DOCKET NO. 50-170

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 19
License No. R-84

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to Facility Operating License No. R-84 filed by the Armed Forces Radiobiology Research Institute (the licensee), dated April 30, 1990 as supplemented on June 19, 1990, and July 13, 1990 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

   F. Prior notice of this amendment was not required by 10 CFR 2.105(a)(4) and publication of notice for this amendment is not required by 10 CFR 2.106(a)(2).

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment, and paragraph 2.C.(2) of License No. R-84 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 19, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Seymour H. Weiss

Seymour H. Weiss, Director
Non-Power Reactor, Decommissioning and
Environmental Project Directorate
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A Technical
Specifications Changes

Date of Issuance: July 23, 1990

ENCLOSURE TO LICENSE AMENDMENT NO. 19

FACILITY OPERATING LICENSE NO. R-84

DOCKET NO. 50-170

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
10	10
11	11

tions on reactor power level indication are included in this Section, since the power level is related to the fuel temperature.

3.2.2 REACTOR SAFETY SYSTEM

Applicability

This specification applies to the reactor safety system.

Objective

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

Specification

The reactor shall not be operated unless the safety systems described in Tables 2 and 3 are operable.

TABLE 2. MINIMUM REACTOR SAFETY SYSTEM SCRAMS

Channel	Maximum Set Point	Minimum Number in Mode Steady State	Minimum Number in Mode Pulse
Fuel Temperature	600°C	2	2
Percent Power, High Flux	1.1 MW	2	0
Console Manual Scram Bar	Closure switches	1	1
High Voltage Loss to Safety Channels	20% loss	2	1
Pulse Time	15 seconds	0	1
Emergency Stop (1 each exposure room, 1 on console)	Closure switch	1	1
Pool Water Level	14 feet from top of core	1	1
Watchdog (DAC to CSC)	On digital console	1	1

Basis

The fuel temperature and power level scrams provide protection to assure that the reactor can be shut down before the safety limit on the fuel element temperature will be exceeded. The manual scram allows the operator to shut down the system at any time if an unsafe or abnormal condition occurs. In the event of failure of the power supply for the safety channels, operation of the reactor without adequate instrumentation is prevented. The preset timer insures that the reactor power level will reduce to a low level after pulsing. The emergency stop allows personnel trapped in a potentially hazardous exposure

room or the reactor operator to stop actions through the interlock system. The pool water level insures that a loss of biological shielding would result in a reactor shutdown. The watchdog scram will insure adequate communication between the Data Acquisition Computer (DAC) and the Control System Computer (CSC) units.

TABLE 5. MINIMUM REACTOR SAFETY SYSTEM INTERLOCKS

Action Prevented	Effective Mode Steady State	 Pulse
Pulse initiation at power levels greater than 1 kilowatt		X
Withdrawal of any control rod expect transient		X
Any rod withdrawal with count rate in operational channel below 0.5 cps	X	X
Simultaneous manual withdrawal of two standard rods	X	

Basis

The interlock preventing the initiation of a pulse at a critical level above 1 kilowatt assures that the pulse magnitude will not allow the fuel element temperature to approach the safety limit. The interlock that prevents movement of standard control rods in pulse mode will prevent the inadvertent placing of the reactor on a positive period while in pulse mode. Requiring a count rate to be seen by the operational channels insures sufficient source neutrons to bring the reactor critical under controlled conditions. The interlock that prevents the simultaneous manual withdrawal of two standard control rods limits the amount of reactivity added per unit time.

3.2.3 FACILITY INTERLOCK SYSTEM

Applicability

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

Objective

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 19 TO

FACILITY OPERATING LICENSE NO. R-84

ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE

DOCKET NO. 50-170

1.0 INTRODUCTION

AFRRI has determined that due to the progressive obsolescence of their control console, a new reactor instrumentation and control system is needed to maintain reliable operations. On May 11, 1988 AFRRI published their safety analysis of the new reactor instrumentation and control system. In this report AFRRI concluded that the new system has equal or greater safety built-in than the existing system and therefore is an allowable change under 10 CFR 50.59. 10 CFR 50.59 permits licensees to make changes in the facility as described in the safety analysis report without prior Commission approval unless the proposed change, test, or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question. A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (1) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (2) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (3) if the margin of safety as defined in the basis for any technical specification is reduced.

The staff concluded from its review of the AFRRI safety analysis report that since (1) the installation of the new reactor instrumentation and control system did present an unreviewed safety question because of the possibility of an accident or malfunction of a different type than any evaluated previously and (2) additional technical specifications were required, NRC review and approval were required of the replacement computerized control system.

Pursuant to 10 CFR 50.90, the licensee submitted by letter dated April 30, 1990, as supplemented on June 19, 1990 and July 13, 1990, a request to amend Appendix A of Facility Operating License No. R-84, "Technical Specifications for the AFRRI Reactor Facility." The licensee submittal of June 19, 1990 resubmitted the May 11, 1988 safety analyses. The requested amendment would allow installation of the microprocessor based instrument and control system and add the watchdog (DAC to CSC) scram to Table 2 of the Technical Specifications, "Minimum Reactor Safety System Scrams."

The licensee has temporarily installed, in parallel to their existing control console, the new digital microprocessor based instrumentation and control system provided by General Atomics. The transfer of control from the old to

the new system (including scram) is via a series of gradual steps accompanied by tests which are expected by AFRRI to demonstrate the reliability of the new equipment while maintaining the proven performance of the existing control system. Upon completion of all testing (described later in this SER), the new console will be used to control (except for the hardwired trip functions) both the safety and nonsafety aspects of operation of the TRIGA reactor and the old analog console will be disconnected. The new console will replace the old analog console in the control room. Included in this change is the installation of three new stepping-motor control rod drives.

The primary functions of the new system will remain the same as the old system; to monitor critical parameters and provide a scram signal when needed, to provide information to the operator and to provide control for the pulse and steady-state modes of operation.

2.0 HARDWARE AND SYSTEMS ASSESSMENT

This portion of the review focused on the areas of potential vulnerability or susceptibility of the new control console which might compromise its ability to present accurate information to the operator and to provide scram signals when required. No assessment was made of the reliability of the nonsafety-related operation controls. Issues investigated included single failure, environmental qualification, seismic qualification, surge withstand capability (SWC), electromagnetic interference (EMI), failure modes and effects, reliability, error detection, and independence.

The primary review criteria for instrument and control systems for research reactors are presented in ANSI/ANS 15.15 (1978) "Criteria for the Reactor Safety Systems of Research Reactors." The staff performed this evaluation also using criteria which apply to current vintage nuclear power plants. However, due to the inherent reactivity insertion safety feature of the TRIGA reactor design and minimal decay heat generation that cannot cause fuel damage, the staff has concluded that these power plant criteria may serve as guidelines and that strict adherence to the power plant criteria is generally not warranted. The exceptions are noted in the appropriate sections below.

During the review and audit, the licensee described the new system including licensing, engineering, testing and training aspects. The vendor also participated and provided additional information. The staff also had benefit of material from the U.S. Air Force, the University of Texas at Austin and the console owners group. The licensee also had an independent safety review performed by ORI, Inc. which concluded that the system was acceptable. This is the first system of this type provided by General Atomics which the staff has reviewed, therefore, there is no direct comparison that can be made to a previously licensed configuration.

At AFRRI, the Safety System Scram Circuit consists of two analog nuclear power monitor channels (NP-1000, NPP-1000) and two fuel temperature channels which are hardwired. Also wired into the scram circuit are contacts for manual scram, pulse timer, low water level, key switch and watchdog timers. The NM-1000 microprocessor based nuclear power channel monitors reactor power, but is not wired to the scram circuit at AFRRI.

2.1 Environmental and Seismic Qualification

The new control system will be installed in the control room and the reactor hall. The staff considers the reactor hall (excluding within the pool itself) to be a mild environment when compared to power plant requirements and therefore the entire system can be considered to be in a mild environment. The system has been constructed in standard commercial enclosures suitable for a mild environment. The testing that has been done to date has not revealed any problems related to temperature or humidity. The new system should not be unduly susceptible to temperature or humidity problems and is therefore acceptable to the staff.

Though there have been no requirements promulgated for seismic qualification testing of research reactor control equipment, the staff reviewed the equipment to determine general ruggedness. The equipment appears to be mounted in a good commerci.) quality fashion which should prevent any significant movement of components within the console and racks. In this TRIGA reactor, an inadvertent scram does not present a challenge to reactor safety systems because a scram consists of the removal of current to the control rod magnets allowing the control rods to drop into the core by gravity. No other equipment is required to maintain the reactor in a safe shutdown condition. The primary concern remaining would be relay contact chatter which could prevent a scram when required. The safety system scram circuits for this system are designed to scram on failure (which includes contact chatter) and therefore the staff concludes that any further testing is not warranted and the system is acceptable.

2.2 Electromagnetic Interference (EMI)

The staff reviewed the susceptibility of the new equipment to EMI due to the potential for common mode interference which could disable more than one system at a time. As discussed earlier, due to the design characteristics of the TRIGA reactor, an inadvertent scram does not present a similar challenge to safety systems that it would on a power reactor, though it might cause operational difficulties such as disrupting an experiment.

At AFRRI, optical isolators are used which will prevent conducted EMI from being transmitted between the control and safety channels. The neutron flux signal cabling is shielded to reduce the impact of radiated EMI. Previous experience with similar equipment provided by several different vendors at other facilities has indicated that if EMI causes any perturbance in the system it will most likely cause a scram, which is acceptable to the staff for a TRIGA reactor. Based on the above, the staff concludes that EMI should not prevent a scram when required and the design is therefore acceptable.

2.3 Power Supplies

The power supplies for the system are buffered to reduce the possible impact of minor power line fluctuations. The scram circuits for the new system are designed to scram when power is lost to them. The NP-1000 and NPP-1000 are analog devices and will respond to power fluctuations similar to the existing analog equipment. The digita: NM 1000 nuclear power channel uses a battery

backed-up random access memory (RAM) to store constant data during loss of power. In addition to self-diagnostics, the NM-1000 has a watchdog timer circuit which puts the NM-1000 in a tripped condition and scrams the reactor if power fluctuations prevent proper software operation. As described in the NM-1000 Software Functional Specification and Software Verification Program (March 1989), the NM-1000 is also tested to verify that the system returns to proper operation following restoration of power. The staff finds this acceptable.

2.4 Failure Modes and Effects

The May 11, 1988 safety analysis for AFRRI included an April 22, 1988 Scram Circuit Safety Analysis performed by the University of Texas at Austin. This study identified the various ways in which the reactor safety system could fail. These include:

1) Physical System Failure (wire breaks, shorts, ground fault circuits)
2) Limiting Safety System Setting Failure (failure to detect)
3) System Operable Failure (loss of monitoring)
4) Computer/Manual Control Failure (automatic and manual scram)

This study was based on a fault tree approach which predicted failure to scram for various failure modes. The study concluded that a failure of all safety systems and therefore failure to scram was extremely unlikely. Failures attributable to the unique failure modes of the software of the NM-1000 were adequately considered and in addition, at AFRRI, the NM-1000 is not directly wired into the scram circuit. The staff concludes that the failure modes and effects of the new system were adequately considered and the design is therefore acceptable.

2.5 Independence, Redundancy and Diversity

The staff reviewed the data link between the safety channels and the nonsafety systems. The safety channels provide direct hard wired scram inputs and are also hardwired directly to independent indicators on the control console. In addition, the safety channels provide inputs to the Non-Class 1E Data Acquisition Computer (DAC) through optical isolators. The optical isolators used have not been tested for maximum credible faults which the staff requires for power plant use, but have been tested by the manufacturer to standard commercial criteria. The DAC is then connected via redundant high speed serial data trunks to the Non-Class 1E Control System Computer (CSC) which interfaces with the operator by controls, a keyboard and CRT displays. Since the CSC does communicate with the safety channels, this aspect of the system would not meet the independence requirements of a power plant. However, the staff has concluded that the level of independence which has been maintained is appropriate for the AFRRI TRIGA reactor and is acceptable.

For the AFRRI facility, redundant fuel temperature (Temp 1, Temp 2) inputs are provided to the scram circuit. Redundant power level inputs (NP-1000, NPP-1000) to the scram circuit are also provided. The staff finds this redundancy acceptable. Several additional scram signals are provided at the control

console (manual scram, system watchdog timers). At AFRRI, the NM-1000 is not wired to the scram circuit but does provide inputs to the rod withdrawal prevent interlock system. The system as installed at AFRRI meets most of the requirements of IEEE-279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations" and IEEE 379-1977 "Application of the Single-Failure Criteria to Nuclear Power Generating Station Class 1E Systems," and is therefore acceptable to the staff.

The operators are provided with information from both the analog NP monitors and the digital NM monitor. The information is displayed on both direct wired bar graphs and on a graphic CRT. The scram is provided with automatic and manual contacts and, with the exception of the computer watchdog scram contacts, is similar to the old system. The staff considers this system sufficiently diverse and therefore is acceptable.

2.6 Testing

Extensive testing of the new system has been done by both the vendor and the licensee. A significant number of design changes took place during the testing that AFRRI performed during the phase-in of the new system. General Atomics has also reported no significant safety problems with their installation. The staff has reviewed the problems discovered during testing of the system and has concluded that the re olutions appear appropriate. The staff also agrees with the assessment by the licensee that long-term operability and safety is enhanced due to installation of equipment which has spare parts available and is capable of bein properly maintained. An additional improvement is the self diagnostics feature which allows continuous on-line testing and reduces the possibility of undetected failures.

3.0 Software Assessment

3.1 Criteria

The staff requires an approved verification and validation (V&V) plan for software which performs a safety function or provides information to the operator. At AFRRI, the NM-1000 provide inputs to the rod withdrawal prevent interlock system block function. The NM-1000 software development was reviewed by the staff to determine the acceptability of the V&V plan. The staff compared the General Atomics V&V plan to Regulatory Guide 1.152 "Criteria for Programmable Digital Computer Software in Safety-Related Systems at Nuclear Power Plants" which endorses ANSI.IEEE 7-4.3.2 - 1982 "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations." The staff has concluded that this standard is appropriate for use in reviewing research reactor software.

3.2 Verification and Validation Plan

The staff reviewed the verification and validation documentation provided by General Atomics. The staff also reviewed the additional validation which was performed by the AFRRI staff. Since the safety scram circuits at AFRRI are hardwired and do not require software to function the emphasis of the review was to ensure that potential software problems could not prevent a scram if required.

The hardwired scram circuit is wired so that a scram will occur even if the control software is requesting rod withdrawal. An additional important feature is included to prevent software errors from interfering with safety function. The Control System Computer (CSC) and Data Acquisition Computer (DAC) include watchdog timers which must be reset every 10 seconds by the software or they will trip and provide a scram signal to the rod magnet power. The watchdog timers provide a continuous check of proper software operation. The staff finds them acceptable. Though the software was not shown to be in full compliance with Reg. Guide 1.152, the software will not impede the safety systems and is therefore acceptable.

4.0 Technical Specifications

The scram circuit at AFRRI will include watchdog timer contacts which will provide a scram upon software failure. The staff has concluded that the presentation of correct, timely information to the reactor operator contributes to the safe operation of the reactor. Therefore, the watchdog scram inputs are added to Table 2, Minimum Reactor Safety System Scrams of the technical specifications. The operability of the watchdog scram will be verified by Technical Specification 4.2.2 which requires a channel test weekly. The basis of Table 2 is also amended to add the watchdog scrams and safety chambers is changed to safety channels to more accurately describe the high voltage loss scram.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no Environmental Impact Statement or Environmental Assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The staff concludes that the hardware design of the new General Atomics console is acceptable for use in the AFRRI TRIGA reactor. The Software design in the CSC, DAC and NM1000 will not prevent the safety functions of the hardwired scram circuit from performing and is therefoe acceptable. The technical specifications are amended to include the watchdog scram inputs and surveillance requirements.

The staff has also concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated, or create the possibility of a new or different kind of accident from any accident previously evaluated, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities, and (3) such activities will be conducted

in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

Principal Contributor: James C. Stewart

Dated: July 23, 1990

Attachment F

Ammendment No. 20 to Facility Operating License

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

October 4, 1990

F

Docket No. 50-170

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

Dear Colonel Irving:

SUBJECT: ISSUANCE OF AMENDMENT NO. 20 TO FACILITY OPERATING LICENSE NO. R-84 - ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE (AFRRI)

The Commission has issued the enclosed Amendment No. 20 to Facility Operating License No. R-84 for the AFRRI TRIGA Research Reactor. The amendment consists of changes to the Technical Specifications in response to your submittal dated April 30, 1990.

The amendment continues to require inspection of fuel elements that are in the core or returned to the core, but allows deletion of the inspection of fuel elements that are in storage until they are returned to the reactor core and have the requisite operating history. The Technical Specifications are amended to reflect these new fuel element inspection requirements.

A copy of the related Safety Evaluation supporting Amendment No. 20 is enclosed.

Sincerely,

Marvin M. Mendonca

Marvin M. Mendonca, Senior Project Manager
Non-Power Reactor, Decommissioning and
Environmental Project Directorate
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 20
2. Safety Evaluation

cc w/enclosures:
See next page

ATTACHMENT F

Armed Forces Radiobiology Research
Institute

Docket No. 50-170

cc: Director, Maryland Office of
Planning
301 West Preston Street
Baltimore, Maryland 21201

County Executive
Montgomery County Government
Rockville, Maryland 20850

Reactor Facility Director
Armed Forces Radiobiology
Research Institute
National Naval Medical Center
Bethesda, Maryland 20814

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

ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE

DOCKET NO. 50-170

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 20
License No. R-84

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to Facility Operating License No. R-84 filed by the Armed Forces Radiobiology Research Institute (the licensee), dated April 30, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

   F. Prior notice of this amendment was not required by 10 CFR 2.105(a)(4) and publication of notice for this amendment is not required by 10 CFR 2.106(a)(2).

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment, and paragraph 2.C.(2) of License No. R-84 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 20, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Seymour H. Weiss

Seymour H. Weiss, Director
Non-Power Reactor, Decommissioning and
Environmental Project Directorate
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A Technical
Specifications Changes

Date of Issuance: October 4, 1990

ENCLOSURE TO LICENSE AMENDMENT NO. 20

FACILITY OPERATING LICENSE NO. R-84

DOCKET NO. 50-170

Replace the following page of the Appendix A Technical Specifications with the attached page. The revised page is identified by amendment number and contains vertical lines indicating the areas of change.

Remove	Insert
22	22

Specification

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

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

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

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

BASIS

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

4.2.5 REACTOR FUEL ELEMENTS

Applicability

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

Objective

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

Specifications

All the fuel elements present in the reactor core, shall be inspected for damage or deterioration, and measured for length and bow at intervals separated by not more than 500 pulses of insertion greater than $2.00 or annually (not to exceed 15 months), whichever occurs first. Fuel elements in storage need not be inspected and measured until returned to the reactor core.

Basis

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

The limit of transverse bend has been shown to result in no difficulty in disassembling the core. Analysis of the removal of heat from touching fuel elements shows that there will be no hot spots that result in damage to the fuel (caused by this touching).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 19 TO

FACILITY OPERATING LICENSE NO. R-84

ARMED FORCES RADIOBIOLOGY RESEARCH INSTITUTE

DOCKET NO. 50-170

1.0 INTRODUCTION

AFRRI has determined that due to the progressive obsolescence of their control console, a new reactor instrumentation and control system is needed to maintain reliable operations. On May 11, 1988 AFRRI published their safety analysis of the new reactor instrumentation and control system. In this report AFRRI concluded that the new system has equal or greater safety built-in than the existing system and therefore is an allowable change under 10 CFR 50.59. 10 CFR 50.59 permits licensees to make changes in the facility as described in the safety analysis report without prior Commission approval unless the proposed change, test, or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question. A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (1) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (2) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (3) if the margin of safety as defined in the basis for any technical specification is reduced.

The staff concluded from its review of the AFRRI safety analysis report that since (1) the installation of the new reactor instrumentation and control system did present an unreviewed safety question because of the possibility of an accident or malfunction of a different type than any evaluated previously and (2) additional technical specifications were required, NRC review and approval were required of the replacement computerized control system.

Pursuant to 10 CFR 50.90, the licensee submitted by letter dated April 30, 1990, as supplemented on June 19, 1990 and July 13, 1990, a request to amend Appendix A of Facility Operating License No. R-84, "Technical Specifications for the AFRRI Reactor Facility." The licensee submittal of June 19, 1990 resubmitted the May 11, 1988 safety analyses. The requested amendment would allow installation of the microprocessor based instrument and control system and add the watchdog (DAC to CSC) scram to Table 2 of the Technical Specifications, "Minimum Reactor Safety System Scrams."

The licensee has temporarily installed, in parallel to their existing control console, the new digital microprocessor based instrumentation and control system provided by General Atomics. The transfer of control from the old to

the new system (including scram) is via a series of gradual steps accompanied by tests which are expected by AFRRI to demonstrate the reliability of the new equipment while maintaining the proven performance of the existing control system. Upon completion of all testing (described later in this SER), the new console will be used to control (except for the hardwired trip functions) both the safety and nonsafety aspects of operation of the TRIGA reactor and the old analog console will be disconnected. The new console will replace the old analog console in the control room. Included in this change is the installation of three new stepping-motor control rod drives.

The primary functions of the new system will remain the same as the old system; to monitor critical parameters and provide a scram signal when needed, to provide information to the operator and to provide control for the pulse and steady-state modes of operation.

2.0 HARDWARE AND SYSTEMS ASSESSMENT

This portion of the review focused on the areas of potential vulnerability or susceptibility of the new control console which might compromise its ability to present accurate information to the operator and to provide scram signals when required. No assessment was made of the reliability of the nonsafety-related operation controls. Issues investigated included single failure, environmental qualification, seismic qualification, surge withstand capability (SWC), electromagnetic interference (EMI), failure modes and effects, reliability, error detection, and independence.

The primary review criteria for instrument and control systems for research reactors are presented in ANSI/ANS 15.15 (1978) "Criteria for the Reactor Safety Systems of Research Reactors." The staff performed this evaluation also using criteria which apply to current vintage nuclear power plants. However, due to the inherent reactivity insertion safety feature of the TRIGA reactor design and minimal decay heat generation that cannot cause fuel damage, the staff has concluded that these power plant criteria may serve as guidelines and that strict adherence to the power plant criteria is generally not warranted. The exceptions are noted in the appropriate sections below.

During the review and audit, the licensee described the new system including licensing, engineering, testing and training aspects. The vendor also participated and provided additional information. The staff also had benefit of material from the U.S. Air Force, the University of Texas at Austin and the console owners group. The licensee also had an independent safety review performed by ORI, Inc. which concluded that the system was acceptable. This is the first system of this type provided by General Atomics which the staff has reviewed, therefore, there is no direct comparison that can be made to a previously licensed configuration.

At AFRRI, the Safety System Scram Circuit consists of two analog nuclear power monitor channels (NP-1000, NPP-1000) and two fuel temperature channels which are hardwired. Also wired into the scram circuit are contacts for manual scram, pulse timer, low water level, key switch and watchdog timers. The NM-1000 microprocessor based nuclear power channel monitors reactor power, but is not wired to the scram circuit at AFRRI.

2.1 Environmental and Seismic Qualification

The new control system will be installed in the control room and the reactor hall. The staff considers the reactor hall (excluding within the pool itself) to be a mild environment when compared to power plant requirements and therefore the entire system can be considered to be in a mild environment. The system has been constructed in standard commercial enclosures suitable for a mild environment. The testing that has been done to date has not revealed any problems related to temperature or humidity. The new system should not be unduly susceptible to temperature or humidity problems and is therefore acceptable to the staff.

Though there have been no requirements promulgated for seismic qualification testing of research reactor control equipment, the staff reviewed the equipment to determine general ruggedness. The equipment appears to be mounted in a good commercial quality fashion which should prevent any significant movement of components within the console and racks. In this TRIGA reactor, an inadvertent scram does not present a challenge to reactor safety systems because a scram consists of the removal of current to the control rod magnets allowing the control rods to drop into the core by gravity. No other equipment is required to maintain the reactor in a safe shutdown condition. The primary concern remaining would be relay contact chatter which could prevent a scram when required. The safety system scram circuits for this system are designed to scram on failure (which includes contact chatter) and therefore the staff concludes that any further testing is not warranted and the system is acceptable.

2.2 Electromagnetic Interference (EMI)

The staff reviewed the susceptibility of the new equipment to EMI due to the potential for common mode interference which could disable more than one system at a time. As discussed earlier, due to the design characteristics of the TRIGA reactor, an inadvertent scram does not present a similar challenge to safety systems that it would on a power reactor, though it might cause operational difficulties such as dis[illegible]ng an experiment.

At AFRRI, optical isolators are used which will prevent conducted EMI from being transmitted between the control and safety channels. The neutron flux signal cabling is shielded to reduce the impact of radiated EMI. Previous experience with similar equipment provided by several different vendors at other facilities has indicated that if EMI causes any perturbance in the system it will most likely cause a scram, which is acceptable to the staff for a TRIGA reactor. Based on the above, the staff concludes that EMI should not prevent a scram when required and the design is therefore acceptable.

2.3 Power Supplies

The power supplies for the system are buffered to reduce the possible impact of minor power line fluctuations. The scram circuits for the new system are designed to scram when power is lost to them. The NP-1000 and NPP-1000 are analog devices and will respond to power fluctuations similar to the existing analog equipment. The digital NM-1000 nuclear power channel uses a battery

backed-up random access memory (RAM) to store constant data during loss of power. In addition to self-diagnostics, the NM-1000 has a watchdog timer circuit which puts the NM-1000 in a tripped condition and scrams the reactor if power fluctuations prevent proper software operation. As described in the NM-1000 Software Functional Specification and Software Verification Program (March 1989), the NM-1000 is also tested to verify that the system returns to proper operation following restoration of power. The staff finds this acceptable.

2.4 Failure Modes and Effects

The May 11, 1988 safety analysis for AFRRI included an April 22, 1988 Scram Circuit Safety Analysis performed by the University of Texas at Austin. This study identified the various ways in which the reactor safety system could fail. These include:

1) Physical System Failure (wire breaks, shorts, ground fault circuits)
2) Limiting Safety System Setting Failure (failure to detect)
3) System Operable Failure (loss of monitoring)
4) Computer/Manual Control Failure (automatic and manual scram)

This study was based on a fault tree approach which predicted failure to scram for various failure modes. The study concluded that a failure of all safety systems and therefore failure to scram was extremely unlikely. Failures attributable to the unique failure modes of the software of the NM-1000 were adequately considered and in addition, at AFRRI, the NM-1000 is not directly wired into the scram circuit. The staff concludes that the failure modes and effects of the new system were adequately considered and the design is therefore acceptable.

2.5 Independence, Redundancy and Diversity

The staff reviewed the data link between the safety channels and the nonsafety systems. The safety channels provide direct hard wired scram inputs and are also hardwired directly to independent indicators on the control console. In addition, the safety channels provide inputs to the Non-Class 1E Data Acquisition Computer (DAC) through optical isolators. The optical isolators used have not been tested for maximum credible faults which the staff requires for power plant use, but have been tested by the manufacturer to standard commercial criteria. The DAC is then connected via redundant high speed serial data trunks to the Non-Class 1E Control System Computer (CSC) which interfaces with the operator by controls, a keyboard and CRT displays. Since the CSC does communicate with the safety channels, this aspect of the system would not meet the independence requirements of a power plant. However, the staff has concluded that the level of independence which has been maintained is appropriate for the AFRRI TRIGA reactor and is acceptable.

For the AFRRI facility, redundant fuel temperature (Temp 1, Temp 2) inputs are provided to the scram circuit. Redundant power level inputs (NP-1000, NPP-1000) to the scram circuit are also provided. The staff finds this redundancy acceptable. Several additional scram signals are provided at the control

console (manual scram, system watchdog timers). At AFRRI, the NM-1000 is not wired to the scram circuit but does provide inputs to the rod withdrawal prevent interlock system. The system as installed at AFRRI meets most of the requirements of IEEE-279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations" and IEEE 379-1977 "Application of the Single-Failure Criteria to Nuclear Power Generating Station Class 1E Systems," and is therefore acceptable to the staff.

The operators are provided with information from both the analog NP monitors and the digital NM monitor. The information is displayed on both direct wired bar graphs and on a graphic CRT. The scram is provided with automatic and manual contacts and, with the exception of the computer watchdog scram contacts, is similar to the old system. The staff considers this system sufficiently diverse and therefore is acceptable.

2.6 Testing

Extensive testing of the new system has been done by both the vendor and the licensee. A significant number of design changes took place during the testing that AFRRI performed during the phase-in of the new system. General Atomics has also reported no significant safety problems with their installation. The staff has reviewed the problems discovered during testing of the system and has concluded that the resolutions appear appropriate. The staff also agrees with the assessment by the licensee that long-term operability and safety is enhanced due to installation of equipment which has spare parts available and is capable of being properly maintained. An additional improvement is the self diagnostics feature which allows continuous on-line testing and reduces the possibility of undetected failures.

3.0 Software Assessment

3.1 Criteria

The staff requires an approved verification and validation (V&V) plan for software which performs a safety function or provides information to the operator. At AFRRI, the NM-1000 provide inputs to the rod withdrawal prevent interlock system block function. The NM-1000 software development was reviewed by the staff to determine the acceptability of the V&V plan. The staff compared the General Atomics V&V plan to Regulatory Guide 1.152 "Criteria for Programmable Digital Computer Software in Safety-Related Systems at Nuclear Power Plants" which endorses ANSI.IEEE 7-4.3.2 - 1982 "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations." The staff has concluded that this standard is appropriate for use in reviewing research reactor software.

3.2 Verification and Validation Plan

The staff reviewed the verification and validation documentation provided by General Atomics. The staff also reviewed the additional validation which was performed by the AFRRI staff. Since the safety scram circuits at AFRRI are hardwired and do not require software to function the emphasis of the review was to ensure that potential software problems could not prevent a scram if required.

The hardwired scram circuit is wired so that a scram will occur even if the control software is requesting rod withdrawal. An additional important feature is included to prevent software errors from interfering with safety function. The Control System Computer (CSC) and Data Acquisition Computer (DAC) include watchdog timers which must be reset every 10 seconds by the software or they will trip and provide a scram signal to the rod magnet power. The watchdog timers provide a continuous check of proper software operation. The staff finds them acceptable. Though the software was not shown to be in full compliance with Reg. Guide 1.152, the software will not impede the safety systems and is therefore acceptable.

4.0 Technical Specifications

The scram circuit at AFRRI will include watchdog timer contacts which will provide a scram upon software failure. The staff has concluded that the presentation of correct, timely information to the reactor operator contributes to the safe operation of the reactor. Therefore, the watchdog scram inputs are added to Table 2, Minimum Reactor Safety System Scrams of the technical specifications. The operability of the watchdog scram will be verified by Technical Specification 4.2.2 which requires a channel test weekly. The basis of Table 2 is also amended to add the watchdog scrams and safety chambers is changed to safety channels to more accurately describe the high voltage loss scram.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no Environmental Impact Statement or Environmental Assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The staff concludes that the hardware design of the new General Atomics console is acceptable for use in the AFRRI TRIGA reactor. The Software design in the CSC, DAC and NM1000 will not prevent the safety functions of the hardwired scram circuit from performing and is therefoe acceptable. The technical specifications are amended to include the watchdog scram inputs and surveillance requirements.

The staff has also concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated, or create the possibility of a new or different kind of accident from any accident previously evaluated, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities, and (3) such activities will be conducted

in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

Principal Contributor: James C. Stewart

Dated: July 23, 1990

Attachment G

10 CFR 50.59 Safety Evaluations Other than New Reactor Console

Attachment 1

REFERENCE: ADMINISTRATIVE PROCEDURE I, FACILITY MODIFICATIONS

ANALYSIS OF PROPOSED MODIFICATION

Modification Nomenclature Water level meter installation

Analysis by 1LT M. Forsbacka, Mr. B. Stallings Date 16 Feb 1990
Supervised By Reed.

SECTION A

1. Document an analysis to determine if a change to the Technical Specifications is required. Include 10CFR and/or Technical Specifications references as applicable:

No change in the technical specifications is necessary.

2. If your analysis determines that a Technical Specifications change is required, go to SECTION B.

3. If a Technical Specifications change is not required, document an analysis to determine if the proposed modification would constitute a change to the facility as described in the SAR. Include 10 CFR and/or SAR references as applicable.

There is no change in the facility as described in the SAR. The meter is a means of accurately determining change in pool level for purposes of tracking water loss due to evaporation. The meter is not connected to any reactor systems, nor is it required for any operations.

G-1

4. Document an analysis to determine if the proposed modification would constitute a change in a procedure as described in the SAR. Include license and/or SAR references as applicable

There is no change in procedure as described in the SAR. The change in water level is noted on a checklist and evaporation is estimated.

5. Document an analysis to determine if the proposed modification would constitute a change in the tests or experiments described in the SAR. Include license and/or SAR references as applicable.

There is no change in the tests or experiments as described in the SAR. The meter would be used in no tests or experiments.

6. If the proposed modification does not constitute a change to the facility, procedures, tests or experiments as described in the SAR (i.e. your answer to the analyses in SECTIONS A.3, A.4 and A.5 is "NO" in all cases), Go to SECTION C.

7. If the proposed modification does require a change in the facility, or a change in a procedure, or a change in a test or experiment, as described in the SAR (i.e. your answer to the analyses in SECTIONS A.3, A.4 or A.5 is "YES" in one or more cases), document an analysis to determine if the probability of occurrence or the consequences of an accident or malfunction

SECTION C

1. No 10 CFR 50.59 is required. The analyses in SECTIONS A.3, A.4 and A.5 provide the bases for this determination that:

   a. no change in the Technical Specifications is required;

   b. no change in the facility as described in the current SAR is proposed;

   c. no change in the procedures as described in the current SAR is proposed, and

   d. the proposed test or experiment

      (1) concides with those described in the current SAR;
      (2) is permitted by the Technical Specifications, and
      (3) has been previously reviewed and conducted.

2. This modification has been reviewed in accordance with ALARA principles. Comments (if necessary) are as follows: N/A JRF

3. Reviewed and approved by ROS [signature] Date 8 March 90

4. Reviewed and approved by RFD [signature] Date 8 March 90

5. RRFSC Concurrence [signature] Date 27 March 90

   a. Provide specific rationale leading to the conclusions derived from your analyses in SECTIONS A.3, A.4 and A.5:

The meter provides information not covered by the SAR, and does not effect reactor operations.

   b. Update drawings or blueprints where applicable.

      N/A

   c. Description of the installation/maintenance:

      The probe is attatched by a bracket to the side of the tank. No maintenance is necessary.

6. File this completed procedure in the Facility Modifications file.

Attachment 1

REFERENCE: ADMINISTRATIVE PROCEDURE I, FACILITY MODIFICATIONS

ANALYSIS OF PROPOSED MODIFICATION

Modification Nomenclature Continuous Air Monitor damper closure system

Analysis by George Date 21 March 83

SECTION A

1. Document an analysis to determine if a change to the Technical Specifications is required. Include 10CFR and/or Technical Specifications references as applicable:

   NO Change necessary

2. If your analysis determines that a Technical Specifications change is required, go to SECTION B.

3. If a Technical Specifications change is not required, document an analysis to determine if the proposed modification would constitute a change to the facility as described in the SAR. Include 10 CFR and/or SAR references as applicable.

   No change necessary
   However this modification will allow the backup cam to supply a closure signal for the Reactor Room Air Dampers as well as the existing closure system from the Primary Cam. This change in no way interferes with the primary cams ability to perform its intended function. It simply provides an equivalent back up system for the primary Cam.

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4. Document an analysis to determine if the proposed modification would constitute a change in a procedure as described in the SAR. Include license and/or SAR references as applicable.

NO Change in procedures

5. Document an analysis to determine if the proposed modification would constitute a change in the tests or experiments described in the SAR. Include license and/or SAR references as applicable.

NO Change Necessary

6. If the proposed modification does not constitute a change to the facility, procedures, tests or experiments as described in the SAR (i.e. your answer to the analyses in SECTIONS A.3, A.4 and A.5 is "NO" in all cases), Go to SECTION C.

7. If the proposed modification does require a change in the facility, or a change in a procedure, or a change in a test or experiment, as described in the SAR (i.e. your answer to the analyses in SECTIONS A.3, A.4 or A.5 is "YES" in one or more cases), document an analysis to determine if the probability of occurrence or the consequences of an accident or malfunction

of equipment important to safety previously evaluated in the SAR is increased. Include SAR references as applicable.

8. Document an analysis to determine if a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is created. Include SAR references as applicable.

9. Document an analysis to determine if the margin of safety as defined in the basis for any Technical Specifications is reduced. Include Technical Specifications and/or SAR references as applicable.

10. If the probability of occurrence is increased, or the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR exists, or the margin of safety, as defined in the basis for any Technical Specifications, is reduced (i.e. your answer to the analyses in SECTION A.7, A.8 or A.9 is "YES" in one or more cases), go to SECTION B.

11. If the probability of occurrence is not increased, and the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR does not exist, and the margin of safety, as defined in the basis for any Technical Specifications, is not reduced (i.e. the answer to your analyses in Sections A.7, A.8 and A.9 is "NO" in all cases), go to SECTION D.

SECTION B

The 10 CFR 50.59 change is not applicable. You must submit an application for amendment to the license pursuant to 10 CFR 50.90 to the Reactor Facility Director for approval and through the RRFSC for its concurrence and then forward it to the USNRC for approval before a modification to the reactor facility can be made.

SECTION C

1. No 10 CFR 50.59 is required. The analyses in SECTIONS A.3, A.4 and A.5 provide the bases for this determination that:

   a. no change in the Technical Specifications is required;

   b. no change in the facility as described in the current SAR is proposed;

   c. no change in the procedures as described in the current SAR is proposed, and

   d. the proposed test or experiment

      (1) concides with those described in the current SAR;
      (2) is permitted by the Technical Specifications, and
      (3) has been previously reviewed and conducted.

2. This modification has been reviewed in accordance with ALARA principles. Comments (if necessary) are as follows: None

3. Reviewed and approved by ROS [signature] Date 5 March 90

4. Reviewed and approved by RFD [signature] Date 6 March 90

5. RRFSC Concurrence [signature] Date 24 July 90

a. Provide specific rationale leading to the conclusions derived from your analyses in SECTIONS A.3, A.4 and A.5:

This will allow backup cam to be fully operational in closing Reactor Room dampers

b. Update drawings or blueprints where applicable.

New schematic Attached, and Filed

c. Description of the installation/maintenance:

New black box attached to wall outside Control Room window

6. File this completed procedure in the Facility Modifications file.

SECTION D

1. A written 10 CFR 50.59 safety evaluation is needed. Attach a separate 10 CFR 50.59 Safety Analysis Report to include:

   a. a detailed description of the change,

   b. the reason the change is considered desirable,

   c. the specific rationale leading to the conclusions derived from your analyses in SECTIONS A.3, A.4 and A.5,

   d. any ALARA considerations (if applicable),

   e. description of the installation or implementation of the change,

   f. description of functional testing program to assure that the change meets the design intent,

   g. identification of current and proposed changes in the SAR wording,

   h. updated drawings or blueprints where applicable, and

   i. RFD's signature for approval and RRFSC's concurrence.

2. The 10 CFR 50.59 Safety Analysis Report has been:

   a. Reviewed and approved by the Reactor Facility Director:

   RFD Signature ______________________________ Date ______________

   b. Concurred by the Chairman, RRFSC:

   Chairman, RRFSC Signature ____________________ Date ______________

3. DOCUMENT UPDATE AND TRAINING (to be reviewed and approved by ROS, where applicable, after the completion of SECTIONS D.1.a. thru D.1.h. and SECTION D.2., the following must be accomplished.)

   a. Maintenance log update reviewed by ____________ Date______________

   b. Procedure update reviewed by ________________ Date______________

   c. SRO/RO training given on (date)______________

   d. Facility drawing/blueprint update reviewed by ___________ Date______

4. All the items in Section D above are to be completed to an extent necessary for normal facility operations to resume.

   RFD Signature ______________________ Completed Date_______________

5. All documentation and requirements specified by this procedure are now completed. ROS Signature _____________________ Date______________

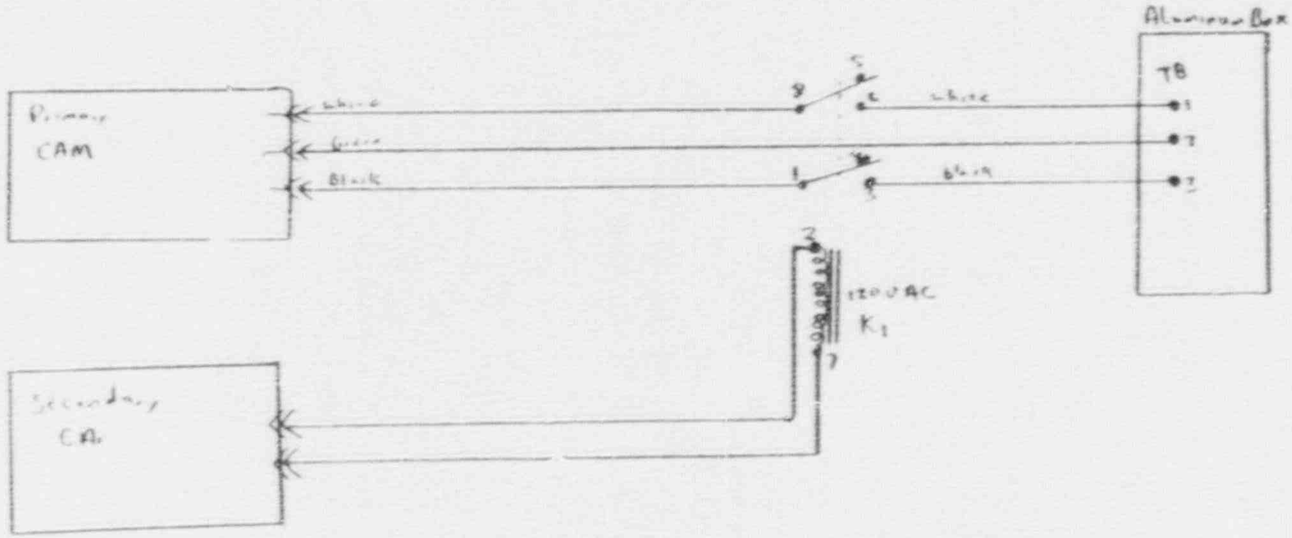
6. Submit a summary of the approved changes with the Annual Report to the USNRC as required by the Technical Specifications, Section 6.6.1.b.

7. File this completed procedure in the Facility Modifications file.

CAM ACTUATED DAMPER CLOSURE SYSTEM

PRE Feb 90

MARCH 90 CHANGE



Attachment 1 (as of Dec 89)

REFERENCE: ADMINISTRATIVE PROCEDURE I, FACILITY MODIFICATIONS

ANALYSIS OF PROPOSED MODIFICATION

Modification Nomenclature Various facility changes and updates to SAR

Analysis by MSG Harry Spence Date 8 Mar 90

Supervised by JRFelty

SECTION A

1. Document an analysis to determine if a change to the Technical Specifications is required. Include 10CFR and/or Technical Specifications references as applicable:

No technical specification changes are required since all changes (#1-17 on Attachment 1) meet requirements of current TS.

2. If your analysis determines that a Technical Specifications change is required, go to SECTION B.

3. If a Technical Specifications change is not required, document an analysis to determine if the proposed modification would constitute a change to the facility as described in the SAR. Include 10 CFR and/or SAR references as applicable.

Changes #1,6,8,9,10,11,12,13,14,16, and 17 constitute changes to the facility.

Changes #2,3,4,5,7, and 15 are updating facility data or making corrections to base building lists only.

attach 63

4. Document an analysis to determine if the proposed modification would constitute a change in a procedure as described in the SAR. Include license and/or SAR references as applicable.

No procedures shown in the SAR are modified by any of these changes.

5. Document an analysis to determine if the proposed modification would constitute a change in the tests or experiments described in the SAR. Include license and/or SAR references as applicable.

No changes are made to any tests or experiments described in the SAR.

6. If the proposed modification does not constitute a change to the facility, procedures, tests or experiments as described in the SAR (i.e. your answer to the analyses in SECTIONS A.3, A.4 and A.5 is "NO" in all cases), Go to SECTION C.

7. If the proposed modification does require a change in the facility, or a change in a procedure, or a change in a test or experiment, as described in the SAR (i.e. your answer to the analyses in SECTIONS A.3, A.4 or A.5 is "YES" in one or more cases), document an analysis to determine if the probability of occurrence or the consequences of an accident or malfunction

of equipment important to safety previously evaluated in the SAR is increased. Include SAR references as applicable.

The probability of occurrence or consequences of an accident or malfunction are not increased and, in many cases, decreased as detailed in the attached Section D Analysis (Attachment 2). See also #8 below.

8. Document an analysis to determine if a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is created. Include SAR references as applicable.

The possibility for an accident or malfunction of a different type than any evaluated previously is not created.

#1,8,16, and 17 are removal of obsolete non-safety related items.

#6,9, and 13 are replacement of obsolete equipment with new equipment performing the same function.

#10,11,12, and 14 are upgrading of radiation detection capabilities that decrease, not increase, the possibility for an accident.

9. Document an analysis to determine if the margin of safety as defined in the basis for any Technical Specifications is reduced. Include Technical Specifications and/or SAR references as applicable

The margin of safety described in any TS basis is not decreased.

#6 conforms to TS Section 5.1

#9 conforms to TS Section 3.3

10. If the probability of occurrence is increased, or the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR exists, or the margin of safety, as defined in the basis for any Technical Specifications, is reduced (i.e. your answer to the analyses in SECTION A.7, A.8 or A.9 is "YES" in one or more cases), go to SECTION B.

11. If the probability of occurrence is not increased, and the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR does not exist, and the margin of safety, as defined in the basis for any Technical Specifications, is not reduced (i.e. the answer to your analyses in Sections A.7, A.8 and A.9 is "NO" in all cases), go to SECTION D.

SECTION B

The 10 CFR 50.59 change is not applicable. You must submit an application for amendment to the license pursuant to 10 CFR 50.90 to the Reactor Facility Director for approval and through the RRFSC for its concurrence and then forward it to the USNRC for approval before a modification to the reactor facility can be made.

SECTION C

1. No 10 CFR 50.59 is required. The analyses in SECTIONS A.3, A.4 and A.5 provide the bases for this determination that:

   a. no change in the Technical Specifications is required;

   b. no change in the facility as described in the current SAR is proposed;

   c. no change in the procedures as described in the current SAR is proposed, and

   d. the proposed test or experiment

      (1) concides with those described in the current SAR;
      (2) is permitted by the Technical Specifications, and
      (3) has been previously reviewed and conducted.

2. This modification has been reviewed in accordance with ALARA principles. JF
Comments (if necessary) are as follows:

Review of #2,3,4,5,7, and 15 are incorporated into approvals in Section D.

3. Reviewed and approved by ROS [signature] Date 8 March 90

4. Reviewed and approved by RFD [signature] Date 8 March 90

5. RRFSC Concurrence [signature] Date 27 March 90

a. Provide specific rationale leading to the conclusions derived from your analyses in SECTIONS A.3, A.4 and A.5:

b. Update drawings or blueprints where applicable.

c. Description of the installation/maintenance:

6. File this completed procedure in the Facility Modifications file.

SECTION D

1. A written 10 CFR 50.59 safety evaluation is needed. Attach a separate 10 CFR 50.59 Safety Analysis Report to include:

   a. a detailed description of the change,

   b. the reason the change is considered desirable,

   c. the specific rationale leading to the conclusions derived from your analyses in SECTIONS A.3, A.4 and A.5,

   d. any ALARA considerations (if applicable),

   e. description of the installation or implementation of the change,

   f. description of functional testing program to assure that the change meets the design intent,

   g. identification of current and proposed changes in the SAR wording,

   h. updated drawings or blueprints where applicable, and

   i. RFD's signature for approval and RRFSC's concurrence.

2. The 10 CFR 50.59 Safety Analysis Report has been:

   a. Reviewed and approved by the Reactor Facility Director:

   RFD Signature M L Moore Date 8 March 90

   b. Concurred by the Chairman, RRFSC:

   Chairman, RRFSC Signature [signature] Date 27 March 90

3. DOCUMENT UPDATE AND TRAINING (to be reviewed and approved by ROS, where applicable, after the completion of SECTIONS D.1.a. thru D.1.h. and SECTION D.2., the following must be accomplished.)

   a. Maintenance log update reviewed by ____________ Date____________

   b. Procedure update reviewed by ________________ Date______________

   c. SRO/RO training given on (date)________________

   d. Facility drawing/blueprint update reviewed by ___________ Date______

4. All the items in Section D above are to be completed to an extent necessary for normal facility operations to resume.

   RFD Signature ______________________ Completed Date_______________

5. All documentation and requirements specified by this procedure are now completed. ROS Signature _____________________ Date_____________

6. Submit a summary of the approved changes with the Annual Report to the USNRC as required by the Technical Specifications, Section 6.6.1.b.

7. File this completed procedure in the Facility Modifications file.

SAFETY ANALYSIS REPORT CHANGES

	PAGE	DESCRIPTION OF CHANGE
1.	X, 5-10	Remove references to old Figure 5-4, Patch Panel system.
2.	2-1	Update base population figures.
3.	2-7/8/9	Update list of buildings on base to remove several demolished storage buildings.
4.	2-10/11	Update building occupancy data and building numbers.
5.	3-1	Revise list of buildings to delete demolished buildings.
6.	3-8/9	Update description of ventilation system to conform to AFRRI TR 83-1.
7.	3-11/13	Remove all technical information about physical security system. Material now contained in Physical Security Plan (restricted disclosure).
8.	3-16	Delete references to chemical treatment of secondary water.
9.	3-21	Change description of conductivity cells to reflect new titanium electrodes in microprocessor-based circuitry.
10.	3-31	Update to indicate that SNM may be in the form of fission chambers.
11.	3-32	Update to show battery backup for RAM R-1.
12.	3-47	Indicate new positions of several criticality dosimeters.
13.	3-48	Change old heated air drying tower to new refrigerated air dryer.
14.	4-26	Increase the number and types of radiation detectors allowed above the reactor core.
15.	4-38/39, 6-5	Update control rod worth data.
16.	5-3	Remove references to epoxy coating on wood in exposure rooms.
17.	5-14	Remove references to Pneumatic Tube System remote control unit.

Attachment 1

SECTION D SAFETY ANALYSIS

Change #	Analysis
1	References are removed to the old dosimetry patch panel system that used to connect ER 1, ER 2, the reactor control room, and the old dosimetry readout room in the LINAC area. Patch panels have been removed as unnecessary. There were no ALARA considerations as the system did not become activated.
6	The ventilation system was upgraded and tested during MILCON in accordance with AFRRI TR 83-1 previously approved by the RRFSC. This current work serves only to update SAR wording to conform to the already approved and implemented system upgrade.
8.	During building mechanical systems upgrading, the reactor secondary water system was shifted to a different cooling tower of the same design and capacity as the original except that the new tower has heating coils to prevent freezing during the winter. This is most desirable since it eliminated the need to add chemicals to the water, chemicals which might have become activated in case of a leak in the heat exchanger. ALARA is improved. The heat exchanger was tested to insure that the new setup was capable of performing its intended function in both winter and summer.
9.	The old platinum conductivity electrodes (parallel plate) in a bridge circuit were replaced with new titanium electrodes (concentric circle design) in modern microprocessor circuitry. This system provides a more representative flow-thru pattern, is easier to calibrate, and has a longer electrode life. There are no ALARA considerations. Testing was performed to insure consistancy of measurements between the old and new systems.
10.	Update SAR wording to indicate that fission chambers may be used as radiation detectors above the ~~care~~ core. Fission detectors are allowed by the TS and contain only milligram quantities of U-235. They are desirable as detectors particularly in the low power ranges. No ALARA considerations are necessary. The operational channel is frequently tested to insure the fission chambers function as required by TS. Any accident involving a fission chamber would not release radioactive material in excess of 10 CFR 20 limits or be more severe than the fuel element accidents described in the SAR.
11.	A backup battery has been added to RAM R-1 to provide for after-hours criticality monitoring in case of power failure. RAM R-5 was always on backup battery and R-1 is added to provide additional protection. ALARA protection is increased. Battery operation is tested monthly, and security guards are instructed in meaning of alarms.
12.	Several criticality accident dosimeters have been moved to new, more radiologically significant locations. This provides better determination of dose levels in case of accident. No functional testing required except ensuring that dosimeters are now more realistically located in areas likely to receive doses in accidents. ALARA documentation improved.

Change # Analysis

13. The old heated compressed air drying tower in room 2158 was replaced by a new refrigerated air dryer in room 3152 nearer to the compressor. The new system provides a level of drying consistent with the old system, uses less power, and requires less maintenance. No ALARA considerations exist. The operation of the dryer is tested as part of the daily startup and filters downstream of the dryer are checked periodically to ensure no moisture passes the dryer.

14. The number and types of radiation detectors allowed above the core are increased to ensure optimal compliance with the TS. All such installed detectors are calibrated by the approved console calibration procedure and compared against originally installed detectors for consistency. No ALARA considerations exist.

16. The one-foot thick wood linings in the exposure rooms were originally given an epoxy coating to aid in decontamination since the wood was the exposed surface in the room. The wood is now covered by masonite panels to protect the wood from contamination (spills,etc). The masonite panels were subjected to test irradiation to ensure that they did not increase activation or modify the radiation spectrum in the room, and as wood is replaced, the new wood is not epoxy coated. The elimination of the epoxy removed a source of possible activation and contamination. With the masonite panels, the epoxy is no longer needed.

17. The pneumatic tube system has been partially disassembled, but it could be returned to operation if needed. The original backup remote control unit has been removed and any PTS operations could be done from the primar, control panel in the same manner as they were done from the remote panel. The change was desirable to all for more efficient use of space in the Radiochemistry Lab. If the PTS were ever returned to service, steps would be included in the operating procedure to ensure that very radioactive samples were not returned to the primary control unit while an individual was standing at the unit. Such samples would either be run in the CET , be allowed to decay at the pool tube terminus, or additional shielding would be added at the control unit during diversion.

Proposed wording changes for all SAR changes are indicated by vertical lines in the right margin of the new SAR.

Attachment 1

REFERENCE: ADMINISTRATIVE PROCEDURE I. FACILITY MODIFICATIONS

ANALYSIS OF PROPOSED MODIFICATION

Modification Nomenclature ALUMINUM PLATE IN EXPOSURE ROOM #1

Analysis MAJ FELTY, SFC LAUGHERY Date 5 July 90

SECTION A

1. Document analysis to determine if a change to the Technical Specification is required. Include 10 CFR and/or Technical Specification references as applicable.

   No change in the Technical Specifications is required.

2. If your analysis determines that a Technical Specification change is required, go to SECTION B.

3. If a Technical Specification change is not required, document an analysis to determine if the proposed modification would constitute a change to the facility as described in the SAR. Include 10 CFR and/or SAR references as applicable.

   Yes. The modification would constitute a change to the facility as described in the SAR.

   Reference: Safety Analysis Report, Section 5.2.1, Page 5-3, Exposure Room Construction:

   "All six surfaces of both exposure rooms are covered with a 1 foot thick wood lining".

4. Document an analysis to determine if the proposed modification would constitute a change in a procedure as described in the SAR. Include license and/or SAR references as applicable.

No. The modification would not constitute a change in a procedure as described in the SAR.

5. Document an analysis to determine if the proposed modification would constitute a change in the tests or experiments as described in the SAR. Include license and/or SAR references as applicable.

No. The modification would not constitute a change in the tests or experiments described in the SAR.

6. If the proposed modification does not constitute a change to the facility, procedures, tests or experiments as described in the SAR (Answer to SECTIONS A.3, A.4 and A.5 is "NO" in all cases), Go to SECTION C.

7. If the proposed modification does require a change in the facility, procedures, tests or experiments, as described in the SAR, document an analysis to determine if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is increased. Include SAR references as applicable.

No. The probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

8. Document an analysis to determine if a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is created. Include SAR references as applicable.

No. The possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created.

9. Document an analysis to determine if the margin of safety as defined in the basis for any Technical Specification is reduced. Include Technical Specifications and/or SAR reference as applicable.

This change does not reduce the margin of safety as defined in the basis for any of the Technical Specifications.

10. If the probability of occurrence is increased, or the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR exists, or the margin of safety, as defined in the basis for any Technical Specifications, is reduced (ie. "YES" answer in SECTION A.7, A.8 or A.9), go to SECTION B.

11. If the probability of occurrence is not increased, and the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR does not exist, and the margin of safety, as defined in the basis for any Technical Specifications, is not reduced (ie. "NO" answer to SECTIONS A.7, A.8 and A.9 in any case), go to SECTION D.

SECTION B

The 10 CFR 50.59 change is not applicable. You must submit an application for amendment to the license pursuant to 10 CFR 50.59 to the Reactor Facility Director for approval and through the RRFSC for its concurrence and then forward it to the USNRC for approval before a modification to the reactor facility can be made.

SECTION C

1. No 10 CFR 50.59 is required. The analysis in SECTIONS A.3, A.4, A.5 provide the bases for this determination that:

   a. no change in the Technical Specification is required.

   b. no change in the facility as described in the current SAR is proposed.

   c. no change in the procedures as described in the current SAR is proposed, and

   d. the proposed test or experiment

      (1) coincides with those described in the current SAR;
      (2) is permitted by the Technical Specifications, and
      (3) has been previously reviewed and conducted.

2. This modification has been reviewed in accordance with ALARA principles. Comments (if necessary) are as follows:

3. Reviewed and approved by ROS ______________________ Date ____

4. Reviewed and approved by RFD M. Hord ______________________ Date 16 July 90

5 RRFSC Concurrence ______________________ Date ____

SECTION D

1. A written 10 CFR 50.59 safety evaluation is needed. Attach a separate 10 CFR 50.59 Safety Analysis Report to include:

   a. a detailed description of the change,

   b. the reason the change is considered desirable,

   c. the specific rationale leading to the conclusions derived from your analysis in SECTIONS A.3, A.4 and A.5,

   d. any ALARA considerations (if applicable)

   e. description of the installation or implementation of the change,

   f. description of functional testing program to assure that the change meets the design intent,

   g. identification of current and proposed changes in the SAR wording,

   h. updated drawings or blueprints where applicable, and

   i. RFD's signature for approval and RRFSC's concurrence.

2. The 10 CFR 50.59 Safety Analysis Report has been:

   a. Reviewed and approved by the Reactor Facility Director:

   RFD Signature ____[signature]____ Date 5 July 90

   b. Concurred by the Chairman, RRFSC:

   Chairman, RRFSC Signature ____[signature]____ Date 24 July 90

3. DOCUMENT UPDATE AND TRAINING (to be reviewed and approved by ROS, where applicable, after the completion of SECTIONS D.1.a. thur D.1.h and SECTION D.2., the following must be accomplished.)

   a. Maintenance log update reviewed by [signature] ROS Date 25 July 90

   b. Procedure update reviewed by [signature] ROS Date 25 July 90

   c. SRO/RO training given on (date) N/R JRF

   d. Facility drawing/update reviewed by [signature] ROS Date 25 July 90

4. All the items in SECTION D above are to be completed to an extent necessary for normal facility operations to resume.

5. All documentation and requirements specified by this procedure are now completed. ROS Signature [illegible] Date 25 July 90

6. Submit a summary of the approved changes with the Annual Report to the USNRC as required by the Technical Specifications, Section 6.6.1.b. ✓

7. File this completed procedure in the Facility Modification file.

SECTION D 10 CFR 50.59 Safety Analysis Report

a. The modification will be the placement of an aluminum plate approximate size 3/16" x 48" x 72" on the floor of Exposure Room #1. The plate will be centered on the core projection, in front of the tracks for the lead shield carriage. The plate will be secured to the wood floor with aluminum screws.

b. The reason this change is considered desireable is that the continuous movement of wheeled experiment holding tables wears grooves in the wooden floor. This makes the exact placement of tables difficult and critical experiments cannot be duplicated.

c. The specific rationale leading to the conclusion stated in SECTION A.3. The SAR currently states that all six surfaces of both Exposure Rooms will be covered with 1 foot thick wood lining; the aluminum plate would be in addition to the wood lining.
As for SECTION A.4, A.5 the modification does not constitute any changes to a procedure, the tests or experiments described in the SAR.

d. ALARA considerations: Activation of the plate will be minimal and poses no additional threat to personnel working in the exposure room.

e. The installation will require drilling approximately 10 holes (1/4" x 3") in the wood floor, aligning the plate and tightening the screws.

f. A functional test was performed using a sheet of plywood similar to the floor in the exposure room, the aluminum plate was placed on the plywood and a table of similar weight was rolled onto the plate. The aluminum plate was stable with no flexing.

g. The SAR wording would require the following addition at the end of Section 5.2.1:

"Exposure Room #1 has a removable 3/16" x 48" x 72" aluminum plate which may be placed on the floor in front of the core projection to minimize wear to the wood floor by the wheels of the experiment holding tables."

h. See attached updated drawing.

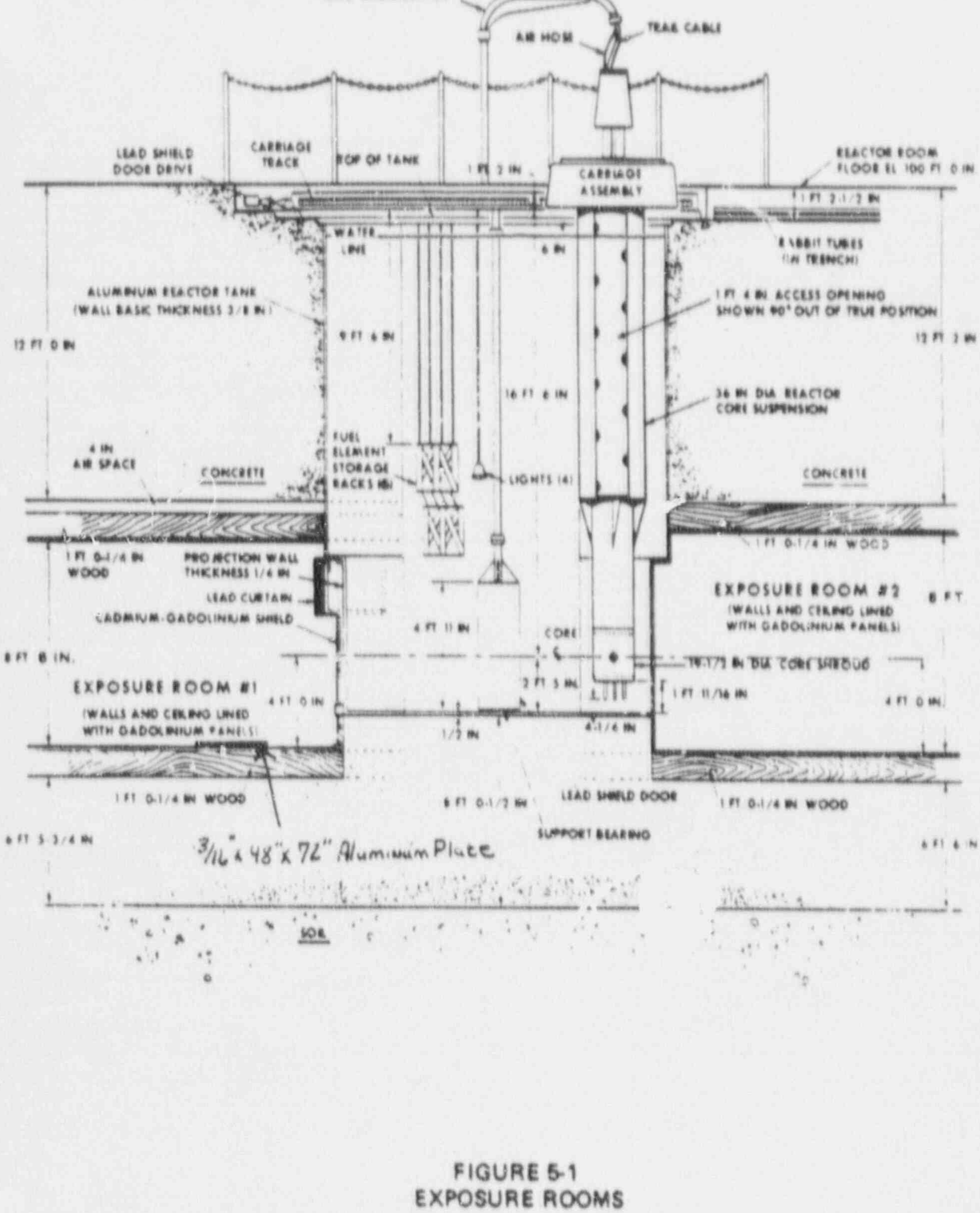


FIGURE 5-1
EXPOSURE ROOMS

Attachment 1

REFERENCE: ADMINISTRATIVE PROCEDURE I, FACILITY MODIFICATIONS

ANALYSIS OF PROPOSED MODIFICATION

Modification Nomenclature Soletron replaced by UPS

Analysis by George Date 07 DEC 1990

SECTION A

1. Document an analysis to determine if a change to the Technical Specifications is required. Include 10CFR and/or Technical Specifications references as applicable:

   None

2. If your analysis determines that a Technical Specifications change is required, go to SECTION B.

3. If a Technical Specifications change is not required, document an analysis to determine if the proposed modification would constitute a change to the facility as described in the SAR. Include 10 CFR and/or SAR references as applicable.

   NO Change

   The SAR states that there is a power regulator between the power distribution transformer and the console. The UPS is a power regulator and therefore, no change has been introduced.

G5

4. Document an analysis to determine if the proposed modification would constitute a change in a procedure as described in the SAR. Include license and/or SAR references as applicable.

NO change necessary

The SAR states that there is a power regulator between the transformer and the console. The UPS is a power regulator. Therefore no change has been introduced.

5. Document an analysis to determine if the proposed modification would constitute a change in the tests or experiments described in the SAR. Include license and/or SAR references as applicable.

NO change necessary

6. If the proposed modification does not constitute a change to the facility, procedures, tests or experiments as described in the SAR (i.e. your answer to the analyses in SECTIONS A.3, A.4 and A.5 is "NO" in all cases), Go to SECTION C.

7. If the proposed modification does require a change in the facility, or a change in a procedure, or a change in a test or experiment, as described in the SAR (i.e. your answer to the analyses in SECTIONS A.3, A.4 or A.5 is "YES" in one or more cases), document an analysis to determine if the probability of occurrence or the consequences of an accident or malfunction

of equipment important to safety previously evaluated in the SAR is increased. Include SAR references as applicable.

8. Document an analysis to determine if a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is created. Include SAR references as applicable.

9. Document an analysis to determine if the margin of safety as defined in the basis for any Technical Specifications is reduced. Include Technical Specifications and/or SAR references as applicable.

10. If the probability of occurrence is increased, or the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR exists, or the margin of safety, as defined in the basis for any Technical Specifications, is reduced (i.e. your answer to the analyses in SECTION A.7, A.8 or A.9 is "YES" in one or more cases), go to SECTION B.

11. If the probability of occurrence is not increased, and the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR does not exist, and the margin of safety, as defined in the basis for any Technical Specifications, is not reduced (i.e. the answer to your analyses in Sections A.7, A.8 and A.9 is "NO" in all cases), go to SECTION D.

SECTION B

The 10 CFR 50.59 change is not applicable. You must submit an application for amendment to the license pursuant to 10 CFR 50.90 to the Reactor Facility Director for approval and through the RRFSC for its concurrence and then forward it to the USNRC for approval before a modification to the reactor facility can be made.

SECTION C

1. No 10 CFR 50.59 is required. The analyses in SECTIONS A.3, A.4 and A.5 provide the bases for this determination that:

   ✓ a. no change in the Technical Specifications is required;

   ✓ b. no change in the facility as described in the current SAR is proposed;

   ✓ c. no change in the procedures as described in the current SAR is proposed, and

   d. the proposed test or experiment

      (1) concides with those described in the current SAR;
      (2) is permitted by the Technical Specifications, and
      (3) has been previously reviewed and conducted.

2. This modification has been reviewed in accordance with ALARA principles. Comments (if necessary) are as follows:

   The Soletron line feeder was Replaced by an Uninterruptable power supply. Both units will Regulate line voltage but the UPS can Maintain Reactor power and allow a safe shutdown of the Reactor electronics. - No ALARA Impact

3. Reviewed and approved by Acting ROS Thomas Wright Date 7 Dec 90

4. Reviewed and approved by RFD [signature] Date 10 Dec 90

5. RRFSC Concurrence [signature] Date 11 Dec 90

a. Provide specific rationale leading to the conclusions derived from your analyses in SECTIONS A.3, A.4 and A.5:

b. Update drawings or blueprints where applicable.

c. Description of the installation/maintenance:

6. File this completed procedure in the Facility Modifications file.

SECTION D

1. A written 10 CFR 50.59 safety evaluation is needed. Attach a separate 10 CFR 50.59 Safety Analysis Report to include:

   a. a detailed description of the change,

   b. the reason the change is considered desirable,

   c. the specific rationale leading to the conclusions derived from your analyses in SECTIONS A.3, A.4 and A.5,

   d. any ALARA considerations (if applicable),

   e. description of the installation or implementation of the change,

   f. description of functional testing program to assure that the change meets the design intent,

   g. identification of current and proposed changes in the SAR wording,

   h. updated drawings or blueprints where applicable, and

   i. RFD's signature for approval and RRFSC's concurrence.

2. The 10 CFR 50.59 Safety Analysis Report has been:

   a. Reviewed and approved by the Reactor Facility Director:

   RFD Signature ______________________________ Date ______________

   b. Concurred by the Chairman, RRFSC:

   Chairman, RRFSC Signature ____________________ Date ______________

3. DOCUMENT UPDATE AND TRAINING (to be reviewed and approved by ROS, where applicable, after the completion of SECTIONS D.1.a. thru D.1.h. and SECTION D.2., the following must be accomplished.)

   a. Maintenance log update reviewed by ____________ Date______________

   b. Procedure update reviewed by __________________ Date______________

   c. SRO/RO training given on (date)________________

   d. Facility drawing/blueprint update reviewed by ____________ Date______

4. All the items in Section D above are to be completed to an extent necessary for normal facility operations to resume.

   RFD Signature ____________________ ____ Completed Date________________

5. All documentation and requirements specified by this procedure are now completed. ROS Signature ________________ _______ Date______________

6. Submit a summary of the approved changes with the Annual Report to the USNRC as required by the Technical Specifications, Section 6.6.1.b.

7. File this completed procedure in the Facility Modifications file.