ANNUAL REPORT

for

VIRGINIA POLYTECHNIC INSTITUTE
and STATE UNIVERSITY
RESEARCH REACTOR

January 1, 1982 to December 31, 1982

Prepared by

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T. S. Smithwick, Reactor Safety Officer

Approved by

T F Parkinson

T. F. Parkinson, Director
Nuclear Reactor Laboratory

Table of Contents

Table of Contents (continued)

LIST OF TABLES

I. Reactor Operations

A. Summary

Reactor Operations time was down significantly during the year 1982 as opposed to previous years due to several reasons. The primary shield tank leak, the failure (and subsequent replacement, installation, and recalibration) of the Neutron Activation Analysis System, reactor staff reduction, and the temporary shortage of qualified operators contributed to the reasons for limited operating time. A summary of operating parameters by quarters is as follows:

Summary of Operations, 1982

Parameter	1st Quarter	2nd Quarter	3rd Quarter	4th Quarter
Kilowatt Hours	0	2806	1063	767
Argon-41 (mCi)	0	4503.4	1707.6	1233
Startups	7	45	43	13
Shutdowns	7	45	43	13
Hours Critical	7.9	57	39.15	20.45
Unscheduled Shutdowns	0	2	1	1

B. Unscheduled Shutdowns

There were four unscheduled shutdowns during 1982. They are categorized as follows:

1) Operator Error - 2
2) Equipment Failure - 2

One shutdown resulted from tripping of the console input power circuit breakers. This has been a recurring problem due to the undersirable locale of the breakers. This problem will be corrected upon completion of installation of an uninterruptible power supply (see II.D). The other operator error shutdown was a result of initial maintenance on the recently installed radiation monitoring system.

The two equipment failure shutdowns which were not related have been corrected with no recurrences. A shorted regulated A/C power supply and a loss of rod control were the causes.

The loss of rod control equipment failure resulted in a procedure change; to address this (see II.C).

I. Reactor Operations (continued)

C. Control Rod/Fuel Inspections

The annual in-core maintenance activities were conducted November 22-27, 1982. A significant amount of information was obtained relevant to the procedure implementation, equipment, and reactor improvements desired by the Virginia Polytechnic Institute and State University reactor staff.

Observation of the fuel elements and control rods yielded satisfactory results. Reactivity measurements (which are not yet completed) have yielded no significant changes. The fuel elements and control rods showed no evidence of degraded integrity.

The desired changes resulting from the core inspection are discussed in their respective subject areas (Part II). Those not being acted on are currently under study and review.

D. Quarterly Scram Time Tests

A summary of resultant scram times by quarter is given. An item to note is the reduction in time for Safety Rod one due to design changes. This is addressed in detail in section II.D.

Summary of Scram Time Tests, 1982

Rod	First Quarter	Second Quarter	Third Quarter	Fourth Quarter		Average
Safety 1	.48	.46	.42	.53	.33	.45
Safety 2	.51	.48	.44	.48	.40	.47
Shim-Safety	.51	.57	.52	.45	.48	.51

Two tests were performed in the fourth quarter to accomodate a scheduling change following core maintenance.

II. Changes

A. Reactor Operator

Mr. D. R. Prater Left the University in April and his license was terminated at that time. Mr. H. G. Knight resigned his staff position in May. Mr. Knight maintained his license until August, (at which time his license was terminated) acting as a consultant for the facility. Mr. William J. Bryant graduated and left the facility in June. Mr. E. R. Ellis, Mr. D. R. Krause, and Mr. P. D. Holian successfully completed the examination for Senior Reactor Operator in July. Mr. E. R. Ellis had been qualified as Reactor Operator at the facility since June, 1981.

II. Change (continued)

B. Reactor Staff

The Virginia Polytechnic Institute and State University Reactor staff underwent a number of changes over the year. Mr. P. D. Holian was hired as Staff Senior Reactor Operator in January. Two staff positions were eliminated entirely and one position was reduced to a half-time status. The Laboratory Supervisor and Computer Programmer positions were ended in June. The position of Staff Senior Operator reverted to half-time in June also. Mr. Prater resigned as Reactor Supervisor in April and Dr. T. F. Parkinson, Nuclear Reactor Laboratory Director, served as Reactor Supervisor for the interim period from April to July. In July, Mr. Holian accepted the position of Reactor Supervisor and served in an Acting capacity until September at which time the appointment was confirmed. Mr. Krause was offered and accepted the position of half-time Staff Senior Reactor Operator in September also.

Although there were significant reductions in operating staff a number of improvements were made in various areas and an even greater number is expected for the coming year.

C. Procedural

The procedures which changed were few in number but primarily dealt with Emergency Procedures. The most significant procedural accomplishment was the development and submission of the Virginia Tech Reactor Facility Emergency Plan. The plan was completed in October and submitted to the Nuclear Regulatory Commission for approval in November. Many future improvements to the plan are expected. All previously approved changes for the year were of an interim nature and are being incorporated into a Revision to be implemented in January or February of 1983. Additional procedural changes to fuel inspection, transfers, and control rod maintenance as a result of the annual in-core maintenance will also be included in this revision.

As a result of experience gained by the reactor staff a number of Technical Specifications changes were written in the last quarter of the year and will also be submitted for approval in January or February of 1983. A brief summary of these proposed changes are given in Table 1.

Procedure changes related to installation and usage of newly installed and utilized equipment were made as interim changes and will also be part of the forthcoming revision.

An entire procedure re-write occurred for control rod casualties due to difficulties encountered with Safety Rod One and the regulating rod malfunctions - in addition to a design change.

TABLE 1

Proposed Technical Specification Changes

Item	Description
Primary Flow Detector	Changed from specific type detector to general type detector.
Portable Neutron Instruments	Performance specifications to address new equipment. Incorporated fast and slow neutron requirements into one range.
Low Power Fuel Elements	Deleted reference to Low Power Elements (not at facility).
Installed Radiation Monitoring Instruments	Performance specifications to address new equipment Range change from 0.01-10.0 mR/hr. to 0.1-10000 mR/hr.
Control Rod Reactivity Insertion Rates	Deleted insertion rate with core devoid of water. Response to an NRC violation; will address rod maintenance.
Vice-President Administration	Changed wording to Vice-President of Administration and Operations.
Radiation Safety Committee	Changed wording to Reactor Safety Committee.
Secondary Cooling System	Performance specifications to allow installation of the new system.

II. Change (continued)

D. Equipment

Most safety related equipment which had been ordered in 1981 was received, inspected, calibrated (if necessary), and utilized during the year. The portable radiacs for both beta-gamma and neutron monitoring were received, inspected, and underwent immediate use.

A new permanent radiation monitoring system was installed in June. The new Victoreen Monitoring system ensures a wider range monitoring capability and is a vast improvement over the previous system. Operation of the system has proved to be highly satisfactory.

The source range cabling was replaced eliminating intermittent noise problems previously experienced. In addition, a high voltage switch was installed in December. This will result in longer detector life by minimizing gas depletion, reduce accumulation of radioactive material, and reduce expenditures. Actual utilization has proved to be satisfactory.

The Uninterruptible Power Supply was received in October and work is currently underway on installation of cable runs, breakers, and associated equipment. When completed, this system will greatly enhance reactor operations and safety. In the event of a power loss while operating, vital instrument indications will be maintained on the line. If the reactor is shutdown, the power supply will ensure no false building evacuation alarms will result from a power loss. In addition, the power supply will greatly aid in simplifying the power distribution system.

Difficulty was experienced in Safety Rod One's operation (see III.A. and VII) and a change was necessitated. The limit switch and rod position indicators were relocated to allow greater flexibility in adjustment and decrease the possibility of interference with rod operation. Since installation, the new system has functioned extremely well and also resulted in a significant reduction in rod insertion time (see I.D). A greater amount of operating time is desired to ensure reliability and sustained performance. If, when re-evaluated, the current characteristics are maintained this new system shall be used for safety rod two and the shim-safety rod.

The original demineralizer used for purifying city water prior to addition to the primary and primary shield experiment tank was replace. The original demineralizer was a Barnstead Model TR-1 demineralizer and the flow rate was insufficient to meet demands ($\sim$ 1/3 gpm). The unit was replaced by a wall mounted monobed demineralizer with replaceable resin, 15-18 Megohm/cm^3 resistivity water production, pressure capability of 125 lbs., and a flow rate of approximately 1-2 gpm. The Reactor Staff is still determining the optimum flow rate.

II. D. Equipment (continued)

In conjunction with installation of the make-up water demineralizer an identical unit was installed with a corrosion-free, submersible, 1/150 HP pump for a continous circulation system to maintain primary shield experiment tank water purity. This system was desired in an effort to prevent a recurrence of the leak which occurred earlier (see III.A) in the tank.

Other equipments, changes and additions included addition of a new Nuclear Data 66 computer system for activation sample analysis, a Tektronix Model 2213 Oscilloscope, a John Fluke 8600A digital multimeter with a thermocouple probe, and a digital rod position indicator. All test equipment is calibrated with National Bureau of Standards accountability.

A summary of the equipment changes, specifications, and drawings is presented in Appendices B through E.

III. Equipment Failures

A. Primary

The foremost difficulty encountered with a primary equipment failure was a leak which developed from the shield tank through pinholes in the primary shield tank window and followed a path through the duct liner and drained out to the core tank drain. The apparent pinhole leak(s) precluded operation at power levels $>$ 10 watts from 10/22/81 until 4/6/82. A report of the final status is included in Appendix A, including an analysis of the effects on nuclear detectors.

Other primary failures included rod control problems which resulted in procedural and design changes. A summary of other primary failures and disposition is included in Table 2.

B. Secondary

No equipment failures occurred in the secondary system.

IV. Research/Services

A. Neutron Acti ation Analysis

As mentioned previously a new Nuclear Data 66 minicomputer interfaced to a Nuclear Data 680 system, two 100 MHZ ADC's, and a high resolution Ge-Li detector were obtained. The services performed in activation analyses were minimal in 1982 due to installation and calibration of the new system. The reactor staff is confident that by the end of next year many previous customers will once again be utilizing the facilities services.

TABLE 2. Chronological Summary of Primary Equipment Failures

Failure Date	Component	Correction Date	Disposition/Comments
10/22/81	Primary shield experiment tank (graphite duct window)	4/6/82	Repaired - see Appendix A
2/8/82	Source Range fission chamber	2/8/82	Replaced - detector tube gas depletion. See II.D Equipment Changes
2/8/82	Power Range Uncompensated Ion Chamber	2/8/82	Replaced - normal detector lifetime obtained
3/6/82	Console mounted digital rod position indication	4/6/82	Temporary meter that was installed was replaced with digital panel meter
4/8/82	Primary shield experiment tank level detector	4/14/82	Replaced
5/11/82	Regulating Rod drive gear train	5/11/82	Repaired - loose set screw
6/12/82	Reactor Room Ventilation control relay	6/13/82	Replaced - coil insulation breakdown
6/16/82	Regulating Rod drive gear train	6/17/82	Repaired - base plate loose.
6/18/82	Source Range signal cable	6/18/82	Replaced - noise problem eliminated
7/24/82	Safety Rod One limit switch	7/24/82	Replaced - resulted in stuck rod, emergency procedure change, and design change
12/15/82	Reactor Room Negative pressure sensor lead	12/16/82	Replaced

IV. Research/Services (continued)

B. Reactor Operator Training

Last year a two phase operator training program was conducted for a utility - for the first time in the facilities history - and was extremely successful. This program is tailored to meet the needs of the individual utility and efforts are underway to continue this program next year. Included in the course are such features as hands-on experience in startups, shutdowns, and experiments in such areas as flux distribution, G-M and neutron detectors, and control rod calibrations — all performed on the University's 100kw Argonaut reactor. Additional facets of this course include lectures by many distinguished faculty members and an optional lodging/meal package.

Beginning in 1983, (for the first time in the facility's history) a Reactor Operator training course will be offered on a quarterly basis to a limited number of individuals. This will include close supervision of student startups, shutdowns, and other experiments - in addition to lectures.

The reactor staff feels that these type of training courses can only serve to further currently qualified operators proficiency, knowledge, and can also be employed to meet the needs of system, theory, and other required lecture areas.

C. New Experiments

No new experiments were performed in 1982. A proposal for a heat transfer performance experiment in the primary system has been delayed until a core tank outlet enlargement can be accomplished to yield the desired flow rate.

Actual performance of this experiment will require a temporary amendment to facility license R-62, an upgraded safety analysis report, and primary system design changes.

This program will be accomplished in a step fashion with the required system design changes to be accomplished following Reactor Safety Committee and Nuclear Regulatory Commission approval.

V. License Renewal Status/FSAR

The original application for a license renewal R-62 was submitted in 1979 and the license has been interimly extended pending Nuclear Regulatory Commission review and approval since then. The actual licensing review process will begin in 1983 - May at the latest.

The Reactor Staff has decided that this renewal will be for 100kw (this had changed several times previously) and power upgrade requests will be made as the necessary equipment, safety analysis reports, and Technical Specification - changes are made (on a case-by-case basis) in order to perform experiments for investigating 500kw feasibility of operations.

V. License Renewal Status/FSAR (continued)

In 1982 a study was performed which was ded[illegible]d solely to heat transfer and flow characteristics of the core. [illegible]tional studies are currently underway for an array of theoretica[illegible]ational scenarios and these will be included in the final safet[illegible]ysis report. It is expected that these will affect the desig[illegible] accident, the emergency plan, the proposed Technical Specificati[illegible] other major aspects of facility operation.

VI. Inspections/Reportable Incidents

A facility inspection was con[illegible] the Nuclear Regulatory Commission Region II Office in Nov[illegible] A copy of their findings and our reply is included in Appendix [illegible]e that final resolution of the items cited has been delaye[illegible] to backlog in the necessary documentation. In any event the required changes will be completed and implemented in the first quarter of 1983. The items requiring a Technical Specification review will be completed following approval by the Nuclear Regulatory Commission.

A stuck rod (Safety One) occurred July 12, which resulted in the previously mentioned design changes. A copy of our report is included as Appendix F.

VII. Health Physics

A. Waste

Radioactive airborne effluents released to the environment through the ventilation exhaust system were limited to Ar-41. The maximum release rate on August 9, 1982 was 4.8×10^{-5} Ci/sec, with a maximum allowable release rate of 1×10^{-4} Ci/sec. The total Ar-41 released to the environment per quarter is shown in the operations summary.

On February 24, 1982, 0.014 mCi of solid waste were transferred to the Campus Waste Storage Building and subsequently shipped by a commercial disposal service to the Washington State burial site on September 8, 1982.

Liquid wastes were analyzed for radioactivity and released to the sanitary sewer system if the levels of radioactivity were below the levels cited in 10 CFR 20.

B. Receipts

No radioactive shipments were received in 1982.

C. Routine Surveys and Swipes

Area surveys and swipes were performed on a quarterly basis. Radiation surveys revealed no significant changes in observed radiation levels during the year. Area swipes in both restricted and unrestricted areas were less than 220 dpm/100cm^2. Source swipes were performed on a quarterly and semi-annual basis. Results were less than 0.005 uCi/swipe.

D. Violations

No violations were issued by the Radiation Safety Office, however, a letter of concern was issued to the Reactor Facility for irradiating a sample without first obtaining the approval of the Health Physicist on the run sheet.

In response to an Nuclear Regulatory Commission violation, a motion was passed that non-operable radiation monitoring equipment be tagged out of order and removed from the control room.

E. Procedure Changes

Procedure VI.2, Pocket Dosimeter Issue Instructions, was changed to limit the maximum exposure to a visitor without health physics approval, from 100 mrem to 20 mrem/quarter.

VII. Health Physics (continued)

F. New Equipment

The following radiation safety equipment was purchased in 1982:

1) 2 - Keithley Beta-Gamma survey instruments
2) 2 - Rem-Pug Neutron Survey instruments
3) Permanent area radiation monitoring system

G. Environmental Monitors

Results from environmental TLD monitors revealed no significant increase in radiation levels over the year. Comparison with previous years showed no significant changes. The highest radiation level obtained during a year of minimal reactor activity, was 550 mrem for the year, on the outside of the double oak door on the west side of the reactor. Efforts are currently underway to reduce this level to below 500 mrem.

H. Personnel Dosimetry

The highest radiation doses recieved during the year are listed below:

	Whole body gamma	Whole body beta	Extremity
Quarter	50 mrem	120 mrem	180 mrem
Year	90 mrem	120 mrem	200 mrem

These doses are well within the maximum allowable occupational doses cited in 10 CFR 20.

I. Building Evacuation Alarm and Drills

In response to a complaint issued on May 3, 1982 concerning Campus Security's response to a building evacuation alarm the following has been put into effect:

1) Security has received instructions in the proper response to the building evacuation alarm.

2) A list of personnel to notify during a reactor emergency has been posted in Security and is updated monthly.

3) A member of the reactor staff and a member of the radiation safety staff will carry a pager after normal working hours.

In order to reduce complaints from professors concerning the timing of building evacuation drills, drills are now being conducted during the first weeks of classes at the beginning of a class period.

VII. Health Physics (continued)

J. In-Core Maintenence

Annual in-core maintenance went relatively well considering the number of new personnel involved. Some difficulties were encountered due to the lack of detailed written procedure; new detailed procedures are currently being written. The existing fuel inspection procedure was found to be inadequate and will be completely revised pending the receipt of a remote optical viewing device. Radiation Safety awareness and the attitude of new employees concerning ALARA principles was commendable.

K. Discussion Items

The following items relating directly to health physics were discussed at the Reactor Radiation Safety Meetings:

1) monthy analyses of primary coolant for iodine

2) monthly, instead of quarterly, TLD checks

3) Reactor Radiation Safety Officer being assigned to the reactor on a half-time basis instead of full-time

4) The sample limit of 2 R/hr on the sample irradiation request form refers to the total number of samples listed on the request, when added together, and not individual samples

5) Radiation Safety Officer would contact Campus Security regarding the possibility of training officers to evaluate the existence of a radiation hazard

6) methods to ensure that new professors in Robeson Hall are aware of the meaning of the building evacuation alarm

STATUS OF REACTOR SHIELD TANK

The shield tank leak has been repaired by covering the original aluminum window with a piece of 3/16 inch 6061 aluminum alloy plate. The aluminum plate was bolted to the shield tank wall over a bead of silicon sealant around the original aluminum window. Epoxy sealant was applied to the outer edge of the aluminum plate and to the mounting bolts as the plate was being secured to the shield tank wall. Following sufficient drying time for the epoxy, the outer edge of the plate was covered with Rockite waterproof patching compound. After adequate drying time for the Rockite, three coats of epoxy paint were applied to the area allowing sufficient drying time between coats with a five-day drying time allowed before the tank was filled with water. On Tuesday, March 30, the tank was filled to the overflow line with water and left for two days with no sign of leakage. The tank was drained and refilled with demineralized water and on April 6, reached the minimum operating level.

The effect of the addition of the aluminum sheet between the two compensated ion chambers in the shield tank was approximated by both calculating the absorption cross-section of the additional elements in the aluminum sheet and by performing transmission tests on a sample of the aluminum sheet. In addition, prior to resuming full power operation, a cobalt flux calibration will be performed and detector output readings will be cross-plotted to ensure that the detectors are not being affected.

A Report On

Neutron Transmission

Through the New Shield Tank

Faceplate

4/13/82

For: The Reactor Safety Committee

By: The Reactor Staff

Approved By: T F Parkinson

T. F. Parkinson, Director

INTRODUCTION

The shield tank leak, which s   to have been due to small holes in the aluminum facepla_e over the graphite duct, has been stopped via the installation of a new aluminum faceplate over the old one. The purpose of this report is to determine the neutron transmission properties through the newly added faceplate to discover whether attenuation of the neutron flux will significantly affect reactor operations.

DESCRIPTION OF TRANSMISSION SAMPLE

The sample used for the transmission analysis was a 3/16" slab of 6061 Aluminum Alloy Plate of the same variety as that placed in the shield tank. The following concentrations of elements (in percent) apply to the sample:

6061 Alloy Consists of:

Element	Concentration
Al	0.9600
Mg	0.0120
Si	0.0080
Ti	0.0015
Cr	0.0035
Mn	0.0015
Fe	0.0070
Cu	0.0040
Zn	0.0025

THEORETICAL CALCULATION OF NEUTRON ATTENUATION
(in a 3/16 inch 6061 Al Alloy Plate)

Element	σ_a (barns)	σ_s (barns)	σ_T (barns)	ρ g/cm^3	A g/mole
Al	0.241	1.4	1.64	2.70	26.9815
Mg	0.069	3.6	3.67	1.74	24.3120
Si	0.160	1.7	1.86	2.40	28.0860
Ti	3.400	14.0	17.40	4.50	47.9000
Mn	13.200	2.3	15.50	7.20	54.9381
Cr	3.100	3.0	6.10	6.92	51.9960
Fe	2.620	11.0	13.60	7.87	55.8470
Cu	3.850	7.2	11.05	8.96	63.5400
Zn	1.100	3.6	4.70	6.59	65.3700

Note: Cross section values are from ANL5800 Handbook and assume maximum error.

$\sigma_{T_{AV}} = 1.8430b$

$\rho_{AV} = 2.78\ g/cm^3$

$A_{AV} = 27.5634\ g/mole$

$$N = \frac{\rho N_A}{A} = \frac{2.78(g/cm^3) \times 6.022045\times10^{23}(atoms/mole)}{27.5634(g/mole)} = 6.0759\times10^{22}\ \frac{atom}{cm^3}$$

$\Sigma_T = N\sigma_{T_{AV}} = 6.0759\times10^{22}\ (atoms/cm^3) \times 1.8430\times10^{-24}(cm^2)$

$\Sigma_T = 0.11198\ cm^{-1}$

Plate Thickness = 0.1875 in. = 0.476 cm

$$\frac{I}{I_o} = e^{-\Sigma_T t} = e^{-(0.11198\ cm^{-1})\ (.476cm)} = 0.9481$$

Fraction of Beam Removed = $1 - \frac{I}{I_o} = 0.0520$

$\boxed{\text{5.2\% of beam is removed}}$

DESCRIPTION OF TRANSMISSION EXPERIMENT

A BF_3 detection system was set up using the PuBe source stored in 100W Randolph. The sample was placed between the source and the detector and two minute counts were taken. Several two-minute counts were also taken with a one millimeter cadmium shield to determine background radiation and also with nothing to impede the beam to determine the unattenuated flux. This data was used to determine the attenuation coefficient via the following formula:

$$A = \frac{(CR\ w/\ Al) - (CR\ w/\ Cd)}{(CR\ w/o) - (CR\ w/\ Cd)}$$

See figure 3 for a tabulation of data collected.

APPARATUS

The following is a description of the equipment used for gathering the transmission data. See figure 1 for a layout diagram and figure 2 for the BF_3 calibration curve.

1 PuBe source
1 N. Wood BF_3 Detector (40 cm) sn-G15820
1 Tennelec Pre-amp sn-884
1 Bertran Nim-Bin Variable High Voltage Power Supply sn-1809
1 Tennelec 211 Linear Amp sn-509
1 Tennelec 441 single Channel Analyzer sn-300
1 Tennelec 545A Timer/Counter sn-103
1 Tektronix O-scope sn-T922 B027660

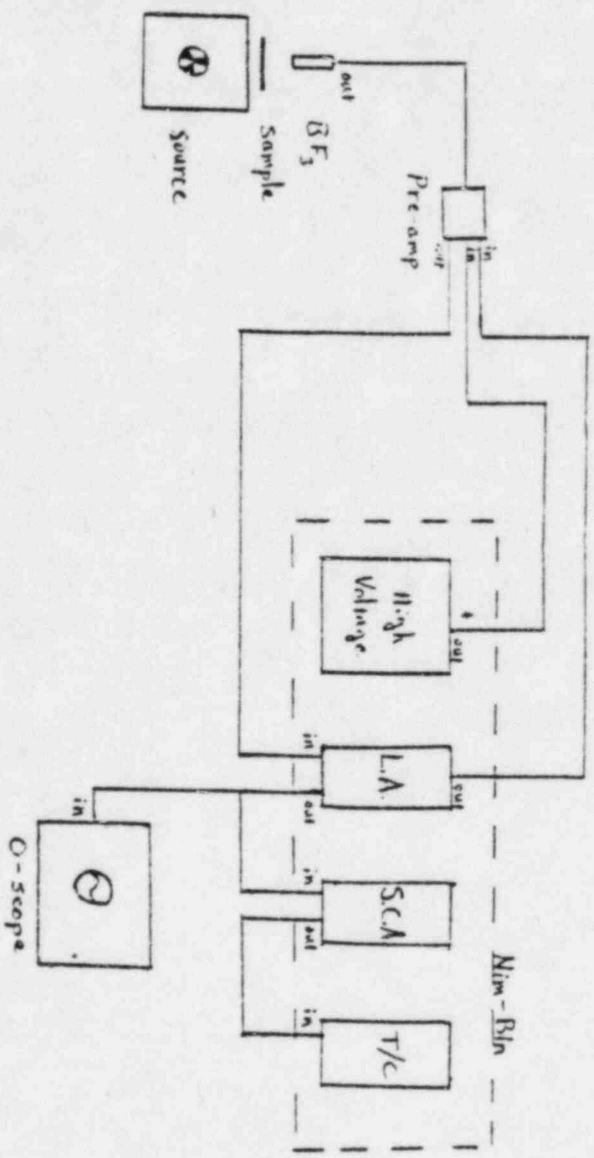
RESULTS

Results from the transmission experiments showed that approximately 19% of the neutrons were attenuated by the aluminum faceplate. This value differs from the theoretical value of 5% that was predicted. The difference is felt to be due to the instrumentation used in the experiment. Source beam collimation and detector geometry are important considerations in transmission experiments and may have been a major contribution to the error in the experimental measurements due to multiple scattering effects and detector vibration. It is the recommendation of the reactor staff that the attenuation due to the new faceplate will not have a significant effect on reactor operations and operations should continue as before the installation.

List of Figures

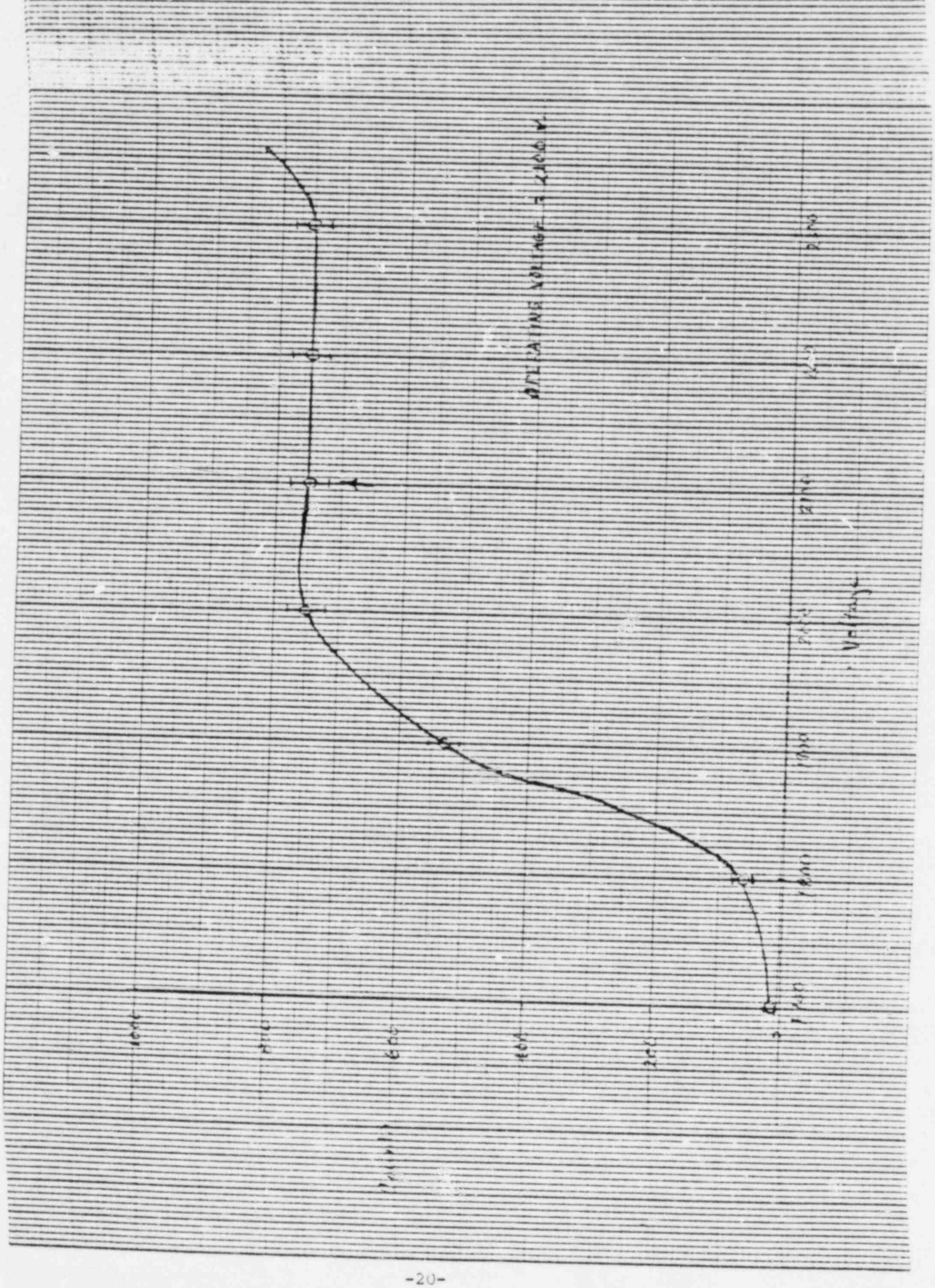
Neutron Transmission Equipment Layout

(Figure 1)



BF3 Calibration Curve

Figure 2



Neutron Transmission Data

(Figure 3)

Instrument Settings

H.V.= 2100 v E-dial= 1.3

c.g.= 4 ΔE-dial= 9.9 (0-10 v scal

f.g.= 1

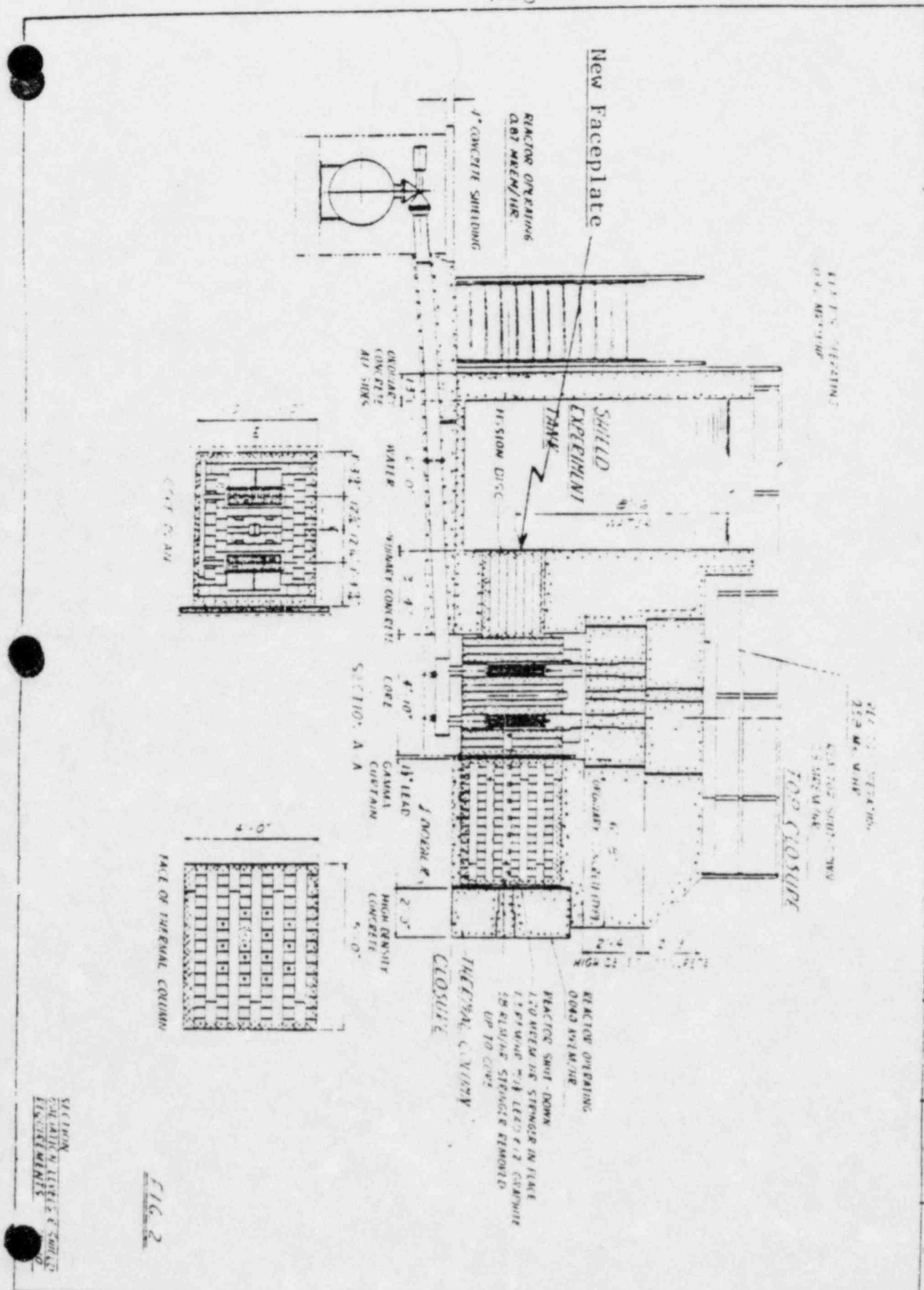
count time= 2 min.

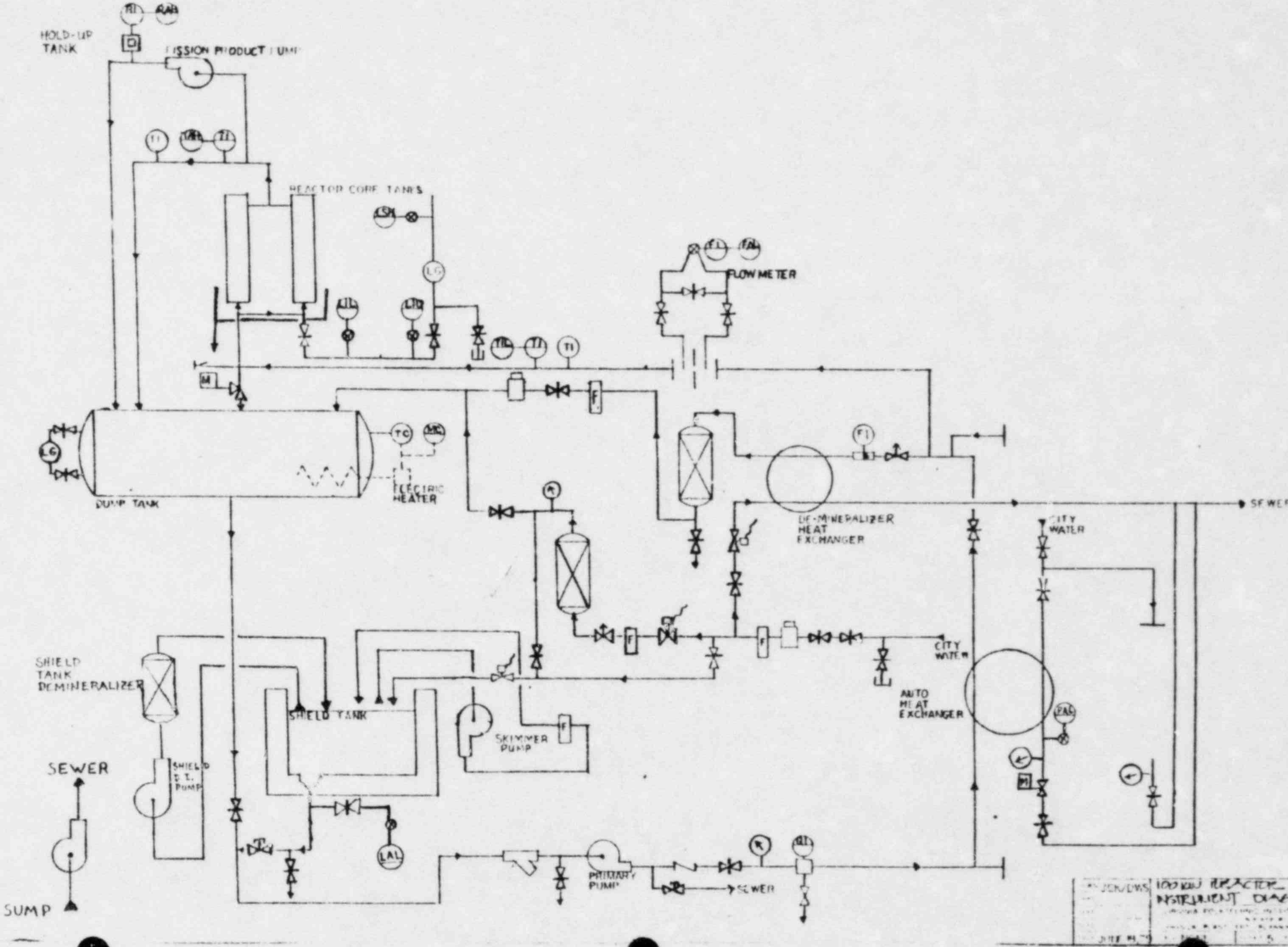
	Counts				
Unshielded	5243±72	5131±72	4986±71	5060±71	4858±70
Cd Shielded	2472±50	2531±50	2564±51	2531±50	2554±51
Al Shielded	4597±68	4534±67	4650±68	4383±66	4645±68
% Not Attenuated	.77±.09	.77±.09	.86±.09	.73±.09	.91±.095

Avg. Not Attenuated = 0.81 ± .07

Faceplate Installation

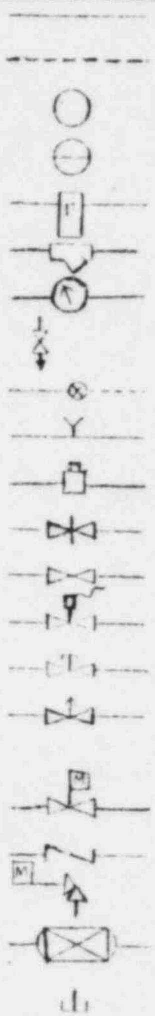
(Figure 4)




HOLD-UP TANK
FISSION PRODUCT PUMP
REACTOR CORE TANKS
FLOWMETER
DUMP TANK
ELECTRIC HEATER
DE-MINERALIZER HEAT EXCHANGER
SEWER
CITY WATER
CITY WATER
AUTO HEAT EXCHANGER
SHIELD TANK DEMINERALIZER
SHIELD TANK
SHIELD D.I. PUMP
SKIMMER PUMP
PRIMARY PUMP
SEWER
SEWER
SUMP
100 KW REACTOR — INSTRUMENT DIAGRAM

LEGEND:

SYMBOLS



- PROCESS PIPING
- INSTRUMENT SIGNAL LINE
- LOCALLY MOUNTED INSTRUMENT
- BOARD MOUNTED INSTRUMENT
- FILTER
- STRAINER
- PRESSURE GAUGE
- SAMPLE TAP
- INSTRUMENT TRANSMITTER
- ADDITION FITTING
- PRESSURE REGULATOR
- GATE VALVE
- GLOBE VALVE
- SOLENOID VALVE
- LOCK VALVE (GATE)
- THROTTLE VALVE (GLOBE)
- [illegible] CONTROL VALVE
- CHECK VALVE
- FAST-ACTING ANGLE GATE VALVE (SPRING - OPENED MOTOR - CLOSED)
- DE - IONIZER
- CAPPED PIPING TERMINATION

INSTRUMENT IDENTIFICATION

FIRST LETTER PROCESS VARIABLE	SECOND LETTER INSTRUMENT FUNCTION	THIRD LETTER INSTRUMENT FUNCTION
T - TEMPERATURE	I - INDICATOR	H - HIGH LIMIT
F - FLOW	C - CONTROL	L - LOW LIMIT
P - PRESSURE	A - ALARM	O - OPERATING LEVEL
L - LEVEL	S - SHUT DOWN	C - CONTROL
M - MANUAL	G - GAGE GLASS	S - SPARE
Q - PURITY		
R - RADIATION		

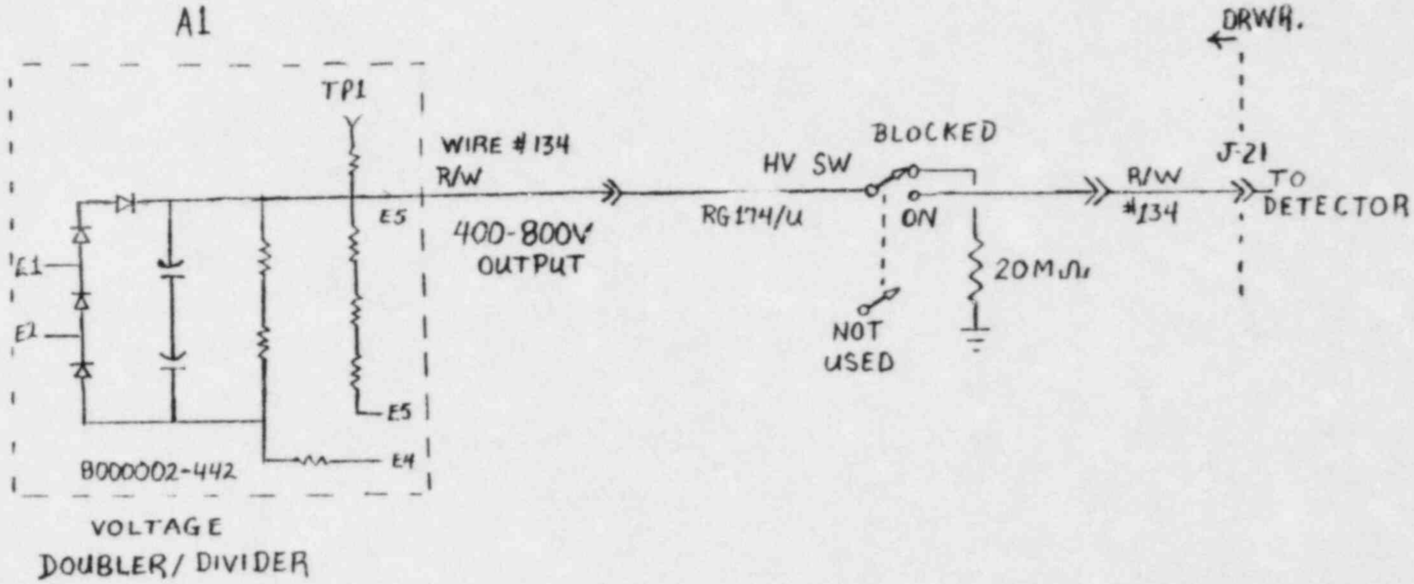
VIRGINIA POLYTECHNIC INST. & STATE UNIVERSITY
PHYSICAL PLANT DEPARTMENT
PLANNING & ENGINEERING DIVISION
328 BURRUSS HALL BLACKSBURG, VA 24061

SHEET 2 of 2

VPI & SU REACTOR
PIPING & INSTRUMENT DIAGRAM : LEGEND

JGK 4/13/79

SOURCE RANGE MODIFICATION



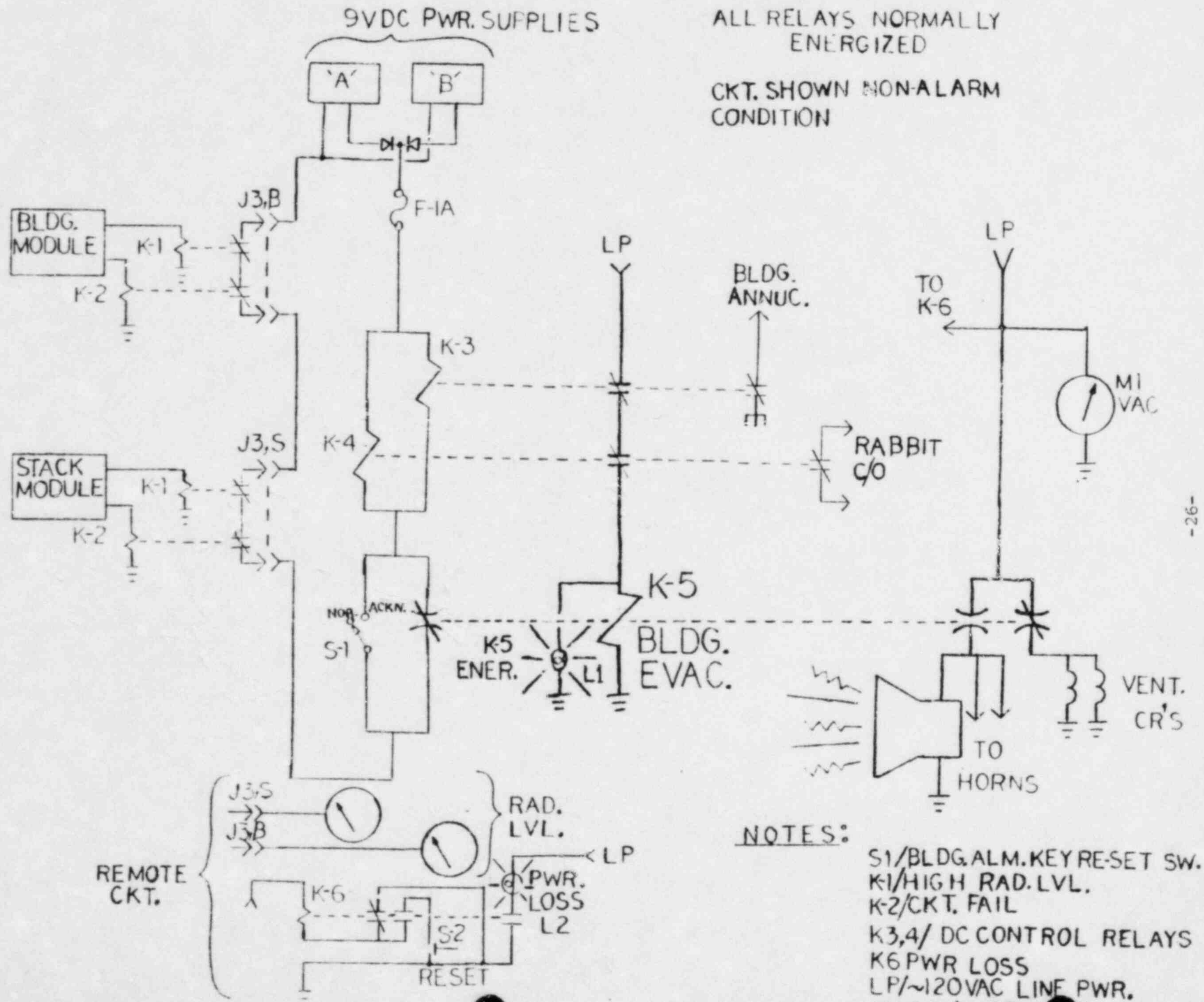
NOTES:

HV SW MOUNTED ON DRWR. FRONT

USE RG 174/U OR EQUIVALENT

DR[illegible]N BY: PH

FUNCTIONAL DIAGRAM \ RAD. MONITOR CONTROL CIRCUITS

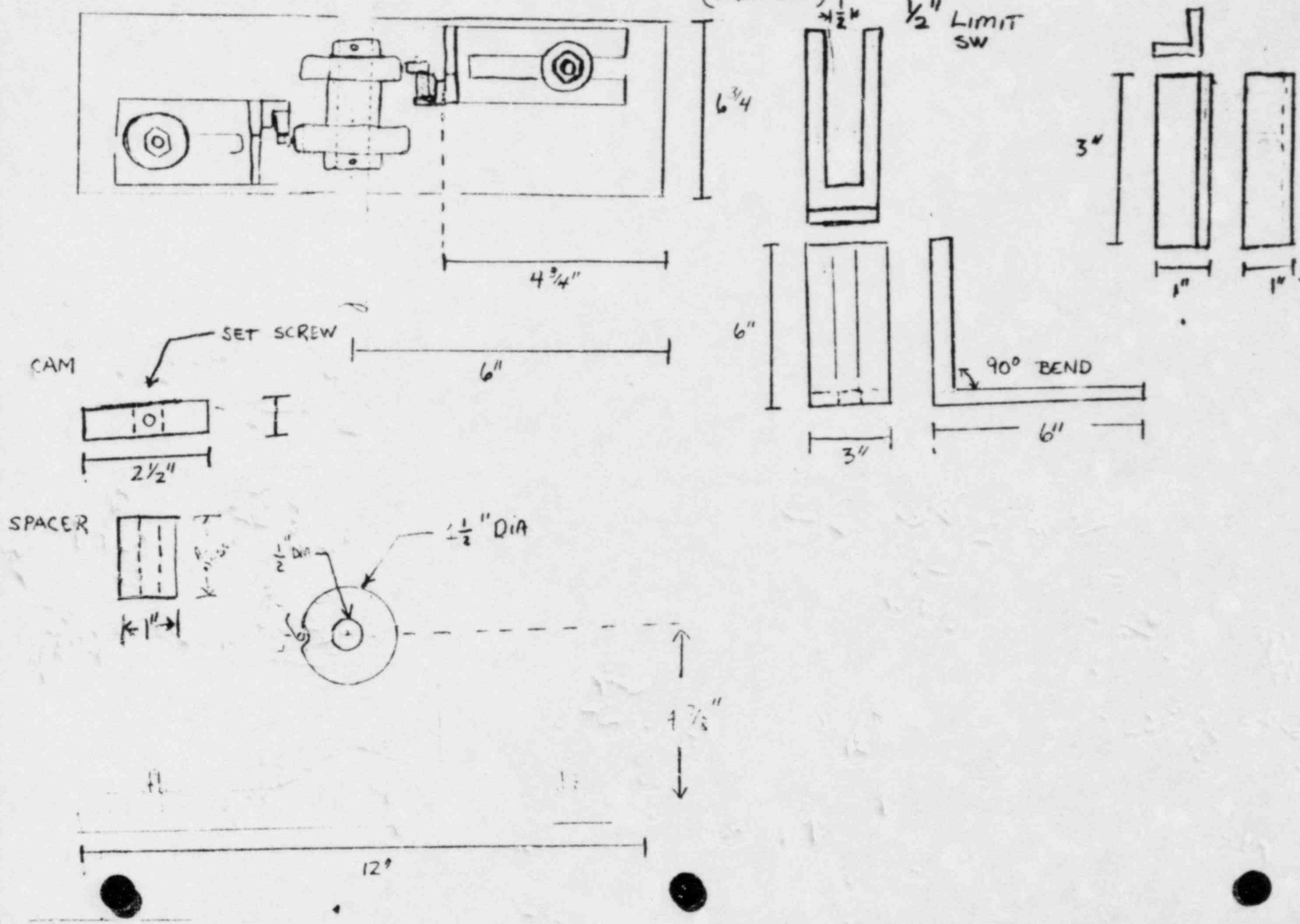


NOTES:

S1/BLDG.ALM.KEY RE-SET SW.
K-1/HIGH RAD. LVL.
K-2/CKT. FAIL
K3,4/ DC CONTROL RELAYS
K6 PWR LOSS
LP/~120VAC LINE PWR.
L1,M1,S1/ON CONSOLE
DWN: PD Belian 12/82

-LIMIT SWITCHES-

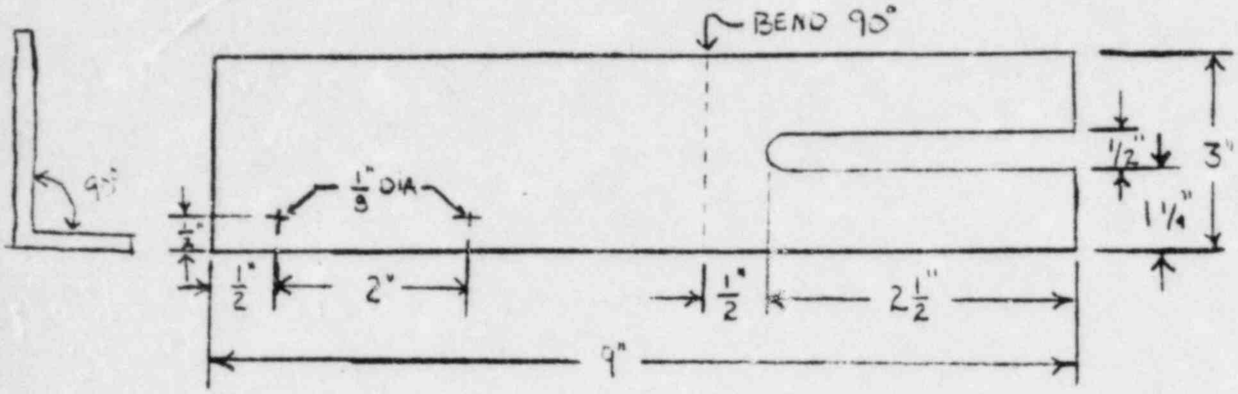
SAFETY ROD ONE
CHANGE
(PROPOSED)



SAFETY ROD ONE
CHANGE
(PROPOSED)

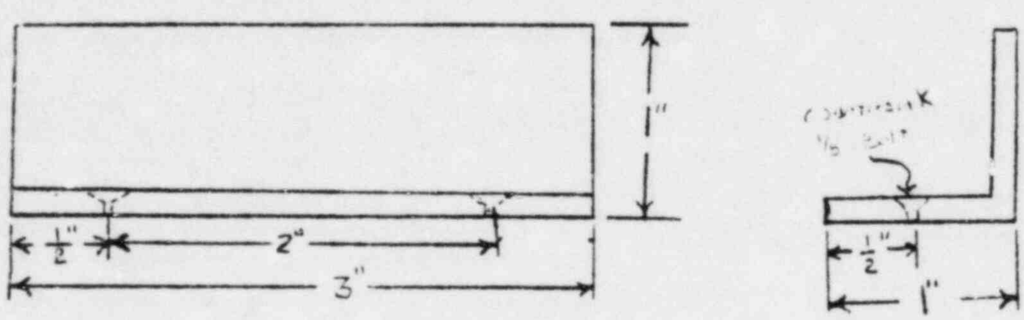
LIMIT SWITCH BASE (ATTACH TO BASE PLATE)

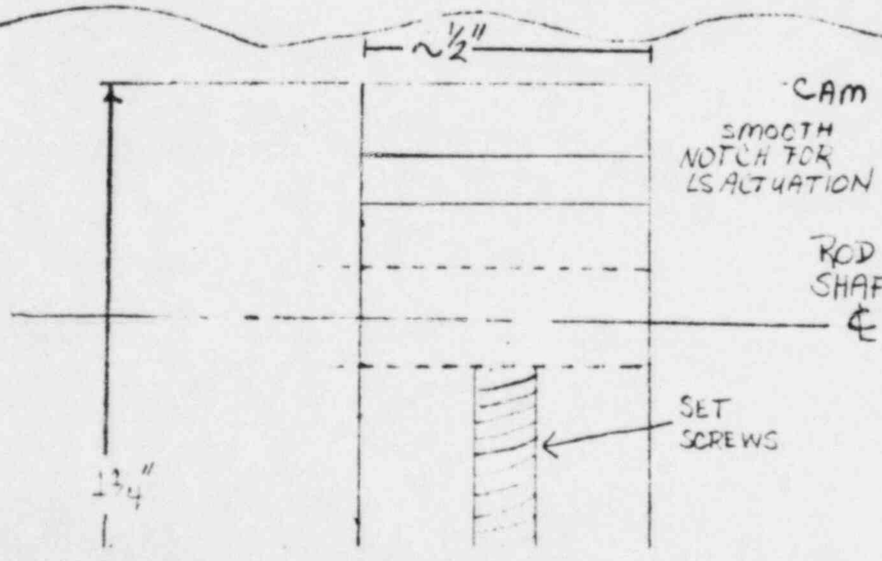
ANGLE BAR (NOT TO SCALE) ALUMINUM



LIMIT SWITCH MOUNT

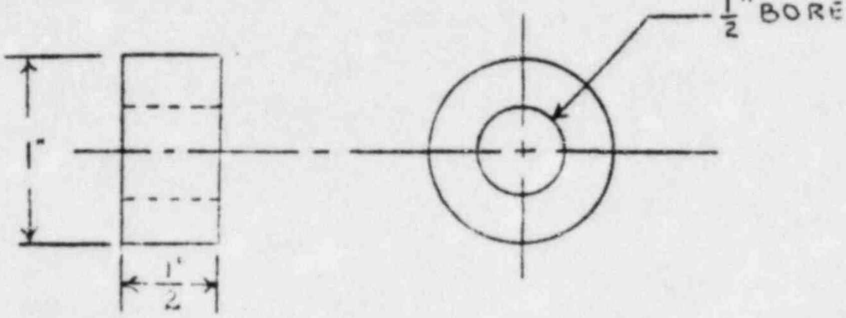
ANGLE IRON ALUMINUM



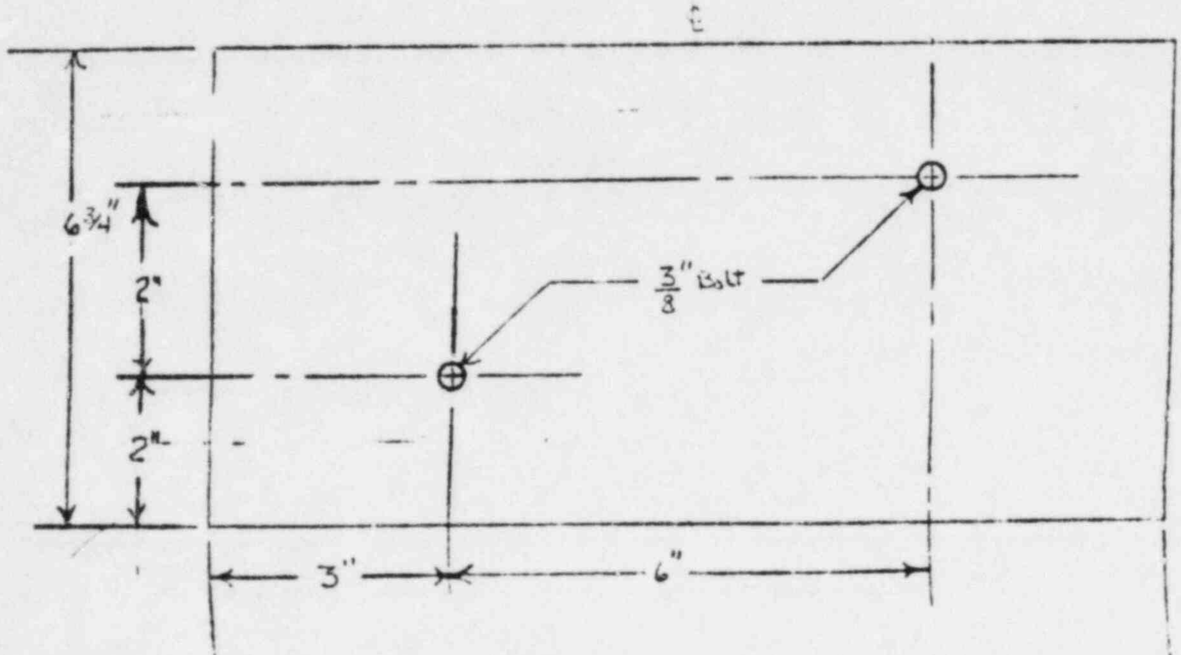


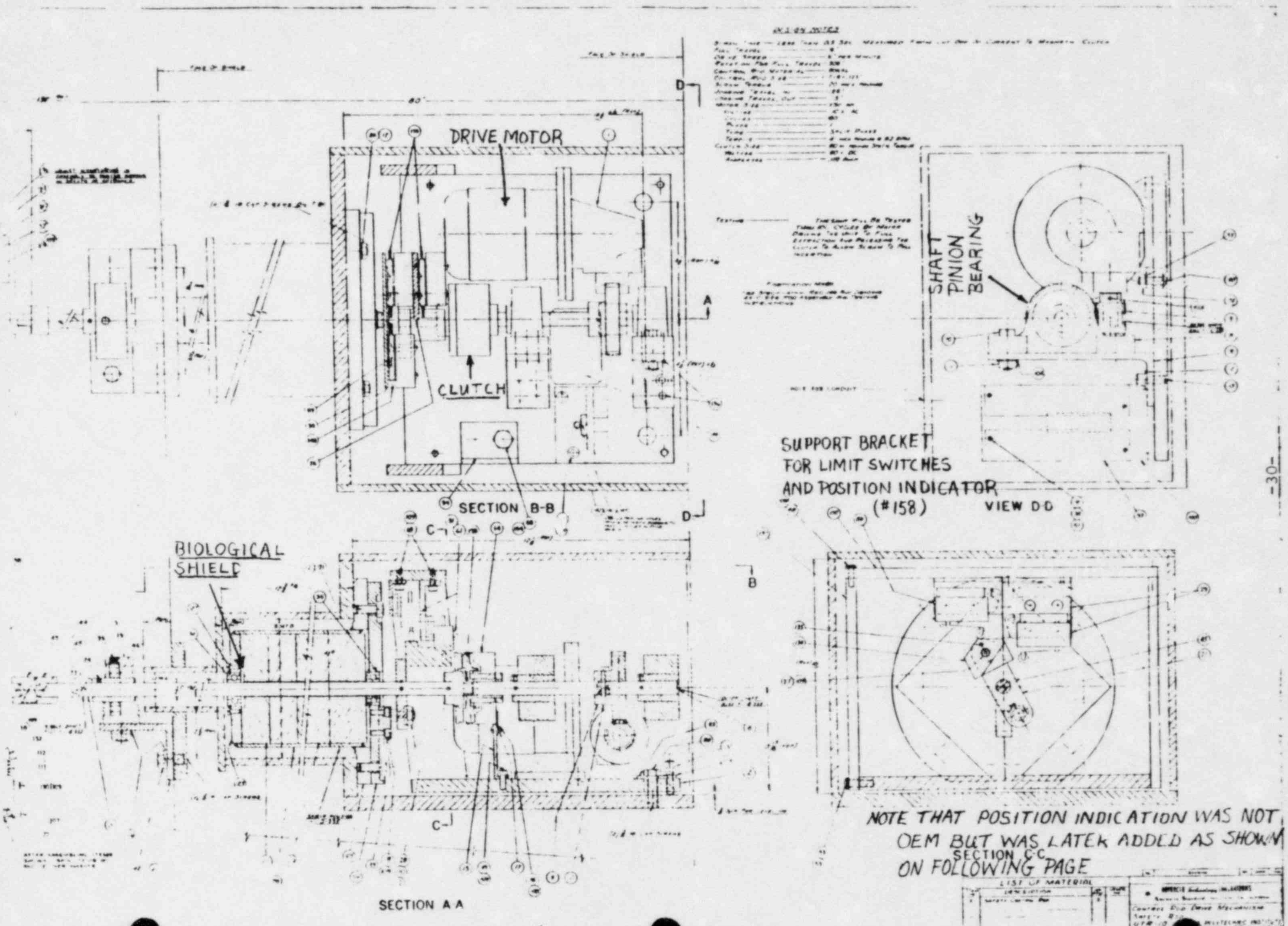
SAFETY ROD ONE
CHANGE
(PROPOSED)

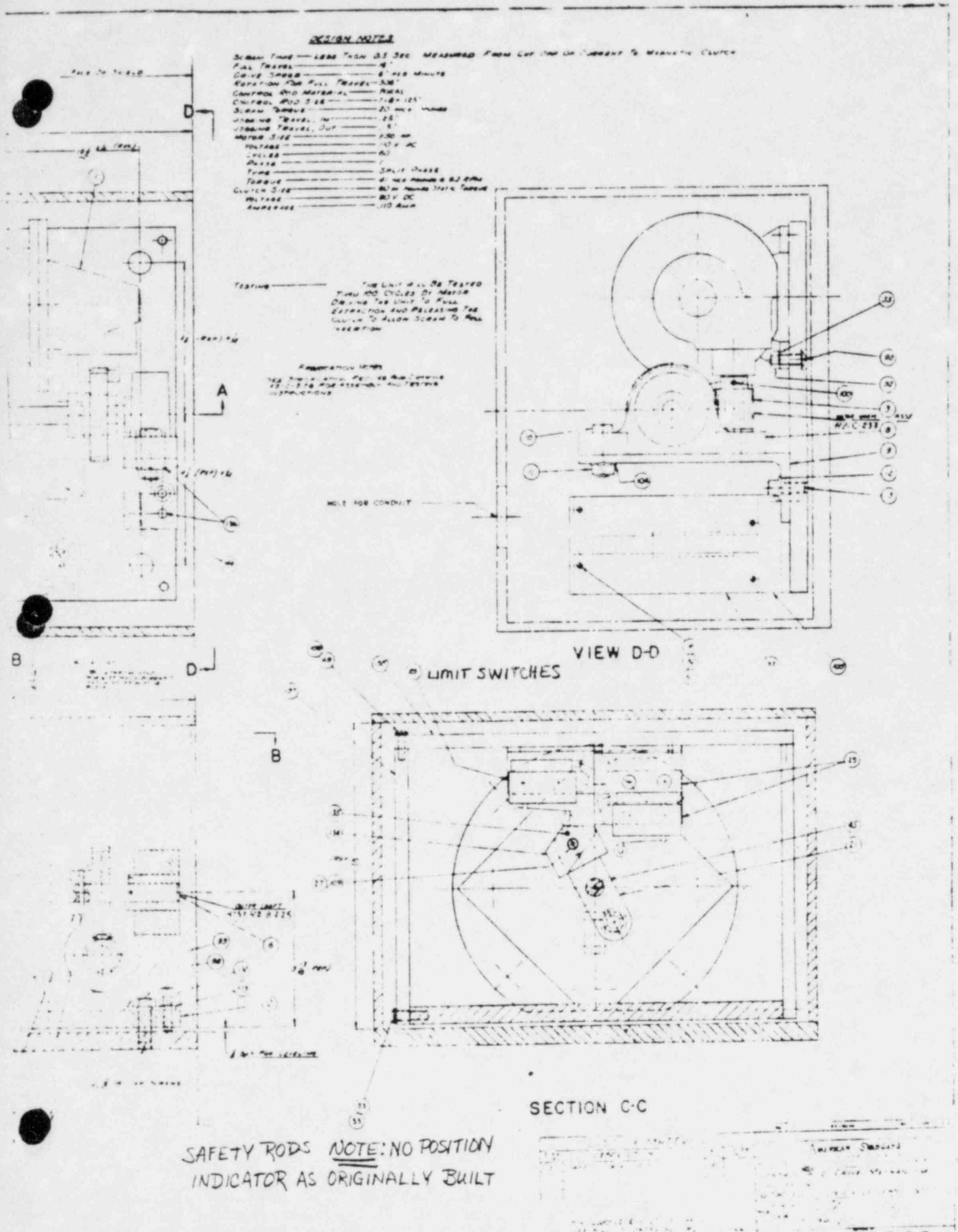
LIMIT SWITCH
(3 per unit) SPACER



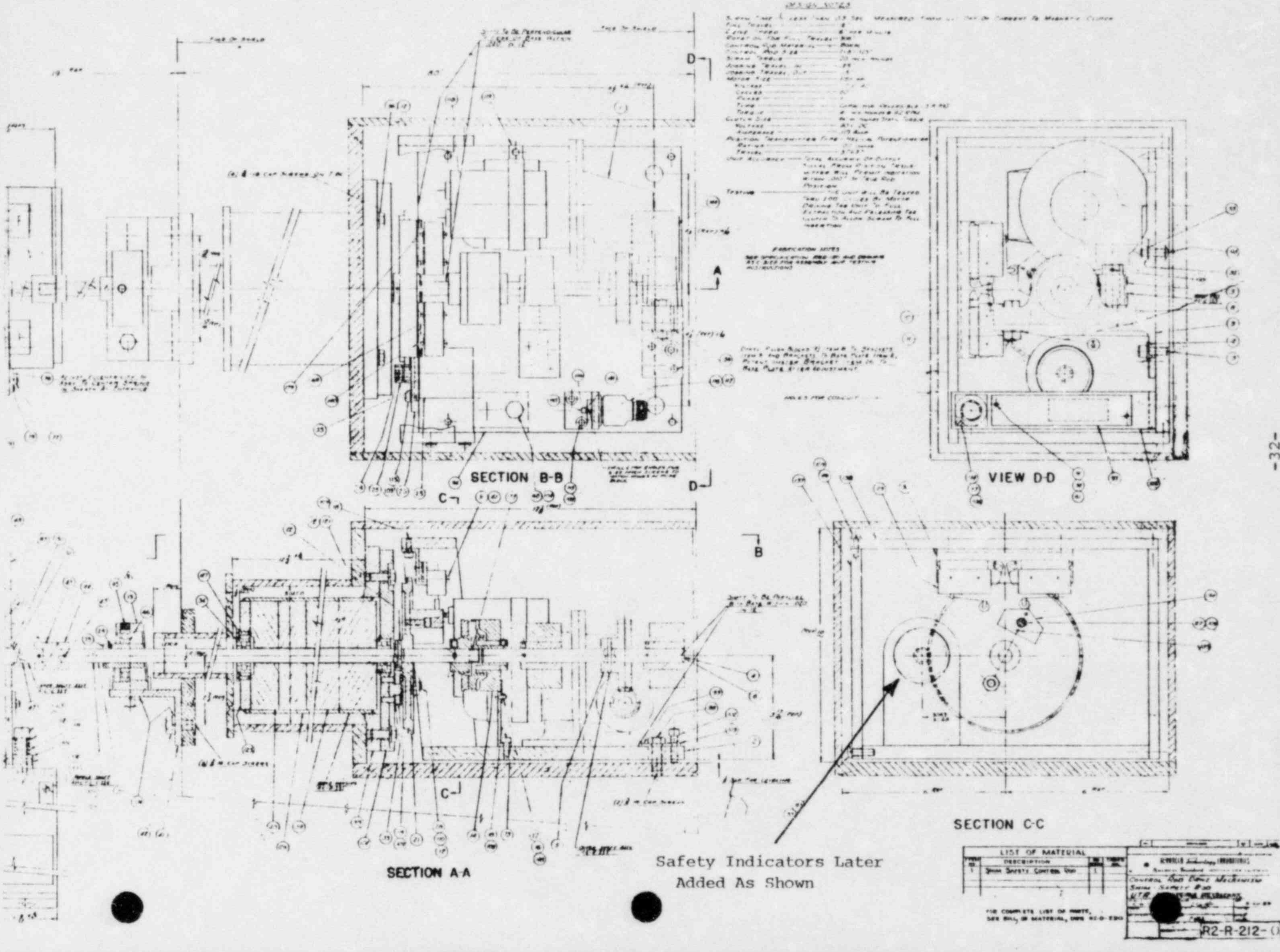
BASE PLATE ALUMINUM (NOT TO SCALE)

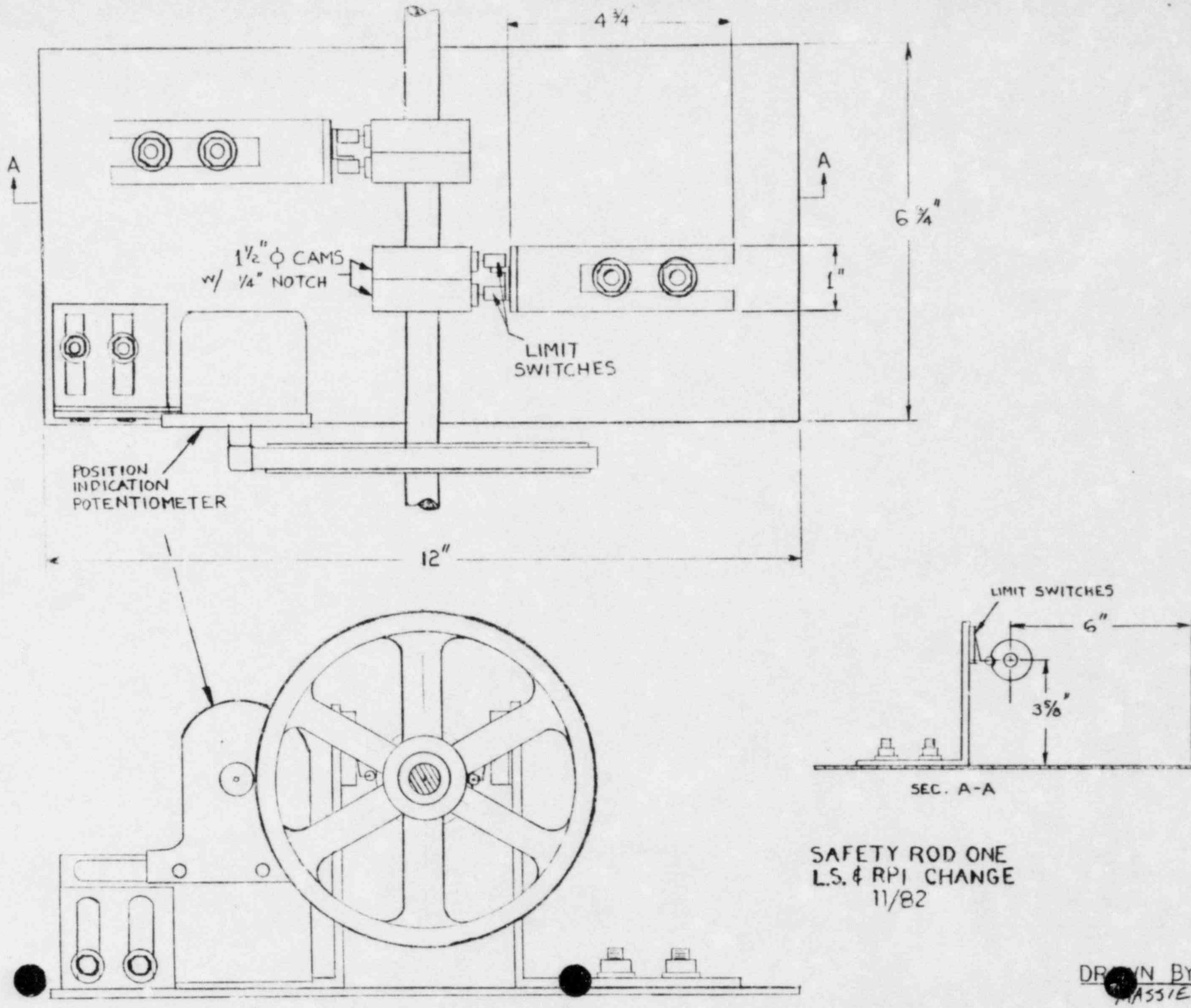



DRIVE MOTOR
CLUTCH
SECTION B-B
BIOLOGICAL SHIELD
SECTION A-A
SHAFT PINION BEARING
SUPPORT BRACKET FOR LIMIT SWITCHES AND POSITION INDICATOR (#158)
VIEW D-D
SECTION C-C
NOTE THAT POSITION INDICATION WAS NOT OEM BUT WAS LATER ADDED AS SHOWN ON FOLLOWING PAGE
LIST OF MATERIAL



SAFETY RODS NOTE: NO POSITION INDICATOR AS ORIGINALLY BUILT


SECTION B-B
VIEW D-D
SECTION A-A
SECTION C-C
Safety Indicators Later
Added As Shown
LIST OF MATERIAL
R2-R-212-0



SAFETY ROD ONE
L.S. & RPI CHANGE
11/82

DRAWN BY:
MASSIE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

DEC 15 1982

Virginia Polytechnic Institute
and State University
ATTN: Mr. T. F. Parkinson, Director
Nuclear Laboratory
Blacksburg, VA 24060

Gentlemen:

Subject: Report No. 50-124/82-01

This refers to the routine safety inspection conducted by Mr. A. K. Hardin of this office on November 17-19, 1982, of activities authorized by NRC License No. R-62 for the Nuclear Research Reactor facility. Our preliminary findings were discussed with you and members of your staff at the conclusion of the inspection.

Areas examined during the inspection and our findings are discussed in the enclosed inspection report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspectors.

During the inspection, it was found that certain activities under your license appear to violate NRC requirements. These items and references to pertinent requirements are listed in the Notice of Violation enclosed herewith as Appendix A. Elements to be included in your response are delineated in Appendix A.

We have examined actions you have taken with regard to previously identified enforcement matters and unresolved items. The status of these items is discussed in the enclosed report.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosures will be placed in the NRC's Public Document Room unless you notify this office, by telephone, within ten days of the date of this letter and submit written application to withhold information contained therein within thirty days of the date of this letter. Such application must be consistent with the requirements of 2.790(b)(1).

The responses directed by this letter and the enclosures are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

Should you have any questions concerning this letter, we will be glad to discuss them with you.

Sincerely,

R.C. Lewis
R. C. Lewis, Director
Division of Project and
Resident Programs

Enclosures:
1. Appendix A, Notice of Violation
2. Inspection Report No. 50-124/82-01

NOTICE OF VIOLATION

Virginia Polytechnic Institute
Nuclear Research Reactor

Docket No. 50-124
License No. R-62

As a result of the inspection conducted on November 17-19, 1982, and in accordance with the NRC Enforcement Policy, 47 FR 9987 (March 9, 1982), the following violations were identified.

A. Technical Specification 8.2.2 requires that the Radiation Safety Committee be responsible for the review of conformity of operations with Technical Specifications. This requirement is implemented by the Virginia Polytechnic Institute Radiation Safety Manual, which requires that the Reactor Safety Committee be responsible for assuring that an annual audit of reactor operations is performed.

   Contrary to the above, the requirement for review of conformity of operations with Technical Specifications was not met in that the annual audit of reactor operations required by the VPI Radiation Safety Manual was not performed for 1981.

   This is a Severity Level V Violation (Supplement I).

B. Technical Specification 8.2.5 requires the Radiation Safety Committee to review changes in the facility or procedures to determine if they constitute an unreviewed safety question.

   Contrary to the above, the Radiation Safety Committee has not performed the required review of procedure IV.15, in that procedure IV.15., which permits manual withdrawal of a control rod for drop time measurement testing, had not been evaluated for unmonitored and unmeasured reactivity input rate that could reduce the margin of safety of Technical Specification.

   This is a Severity Level IV Violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, you are hereby required to submit to this office within thirty days of the date of this Notice, a written statement or explanation in reply, including: (1) admission or denial of the alleged violations; (2) the reasons for the violations if admitted; (3) the corrective steps which have been taken and the results achieved; (4) corrective steps which will be taken to avoid further violations; and (5) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

Date: DEC 15 1982

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

Report No. 50-124/82-01

Licensee: Virginia Polytechnic Institute
and State University
Blacksburg, VA 24060

Facility Name: Nuclear Research Reactor

Docket No. 50-124

License No. R-62

Inspectors: A. K. Hardin — 12/8/82
A. K. Hardin — Date Signed

C W Hehl — 12/8/82
C. W. Hehl — Date Signed

Approved by: [signature] — 12/14/82
P. R. Bemis, Section Chief, Division of Project and Resident Programs — Date Signed

SUMMARY

Inspection on November 17-19, 1982

Areas Inspected

This routine, unannounced inspection involved 32 inspector-hours on site in the areas of review and audit, organization, logs and records, requalification training, procedures, surveillances, experiments and previous unresolved items.

Results

Of the seven areas inspected, no violations or deviations were identified in six areas; two violations were found in one area (review and audit, paragraphs 7 and procedures, paragraph 9).

DETAILS

1. Persons Contacted

Licensee Employees

*A. K. Furr, Director, Safety & Health Program
*T. F. Parkinson, Director, Nuclear Reactor Laboratory
*J. B. Jones, Head, Mechanical Engineering Department
*D. C. Smiley, Campus Radiation Safety Officer
*R. A. Teekel, Chairman, Radiation Safety Committee
*T. S. Smithwick, Reactor Radiation Safety Officers
*P. D. Holian, Reactor Supervisor
*E. R. Ellis, Senior Reactor Operator
*D. R. Krause, Senior Reactor Operator

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized on November 19, 1982, with those persons indicated in paragraph 1 above. The open items and areas of noncompliance were discussed with, and acknowledged by the licensee.

3. Licensee Action on Previous Enforcement Matters

A. (Closed) Unresolved Item (50-124/79-02-05). This unresolved item dealt with manual withdrawal of control rods during performance of the rod drop time measurements. The item was reviewed in August 1981, in IE Inspection Report 81-02, and was left unresolved pending completion of a commitment by the licensee to document the procedure to NRR. As of the current inspection, no action has been taken by the licensee. The item is closed as an unresolved item and changed to noncompliance as discussed in paragraph 9.

B. (Closed) Noncompliance (50-124/81-02-02). On August 17, 1982, the licensee was cited for noncompliance with VPI procedure VI.6, paragraph II.B. The licensee had revised nuclear reactor procedure II.2 without completing the procedurally required reviews and approvals. On August 25, 1981, the licensee responded, stating their corrective and preventive measures. The inspector verified that the procedure change had been reviewed by the Reactor Safety Committee and that preventative measures, committed to in the response, had been implemented.

4. Unresolved Items

Unresolved items were not identified during this inspection.

5. Organization, Logs and Records

a. Organization

The inspector reviewed the organizational changes which have occurred with respect to the VPI Radiation Safety Committee and the formation of a Reactor Safety Committee. The Radiation Safety Committee, which is assigned specific review and audit functions by the facility Technical Specifications, has delegated these specific functions of review and audit to the Reactor Safety Committee, with the Radiation Safety Committee retaining overall review authority. NRC Region II was notified of this organizational change by letter in 1979. The VPI Radiation Safety Manual, revision dated September 22, 1980, supports this present organizational structure and delineates the responsibilities assigned the Reactor Safety Committee, but the listing of Reactor Safety Committee responsibilities falls short of the responsibilities delineated in Technical Specification 8.2. An audit by the inspector of the review and approval activities of Reactor Safety Committee has shown that with exception of one area (see paragraph 7 of this report), the Reactor Safety Committee has met the requirements of Technical Specifications. The inspector's concerns were discussed during the exit interview and the licensee committed to reviewing this area and to bringing the Radiation Safety Manual listing of Reactor Safety Committee responsibilities in line with those delineated in the Technical Specifications. The licensee also committed to reviewing the need for an administrative Technical Specification change in order that the TS more closely reflect the present organization. This is an open item. (50-124/82-01-01)

b. Logs and Records

The inspector reviewed the console logs and found the records complete and traceable. The inspector had no further questions.

6. Requalification Program

The inspector reviewed the requalification program for 1981 and 1982. The examinations, the examined individuals answers, Reactor Supervisor observations and evaluations, and documentation of required reactor control manipulations were satisfactory. Records of the required monthly meetings of qualified operators were reviewed for July 1981 to October 1982 in response to a previous open item in this area. The monthly meeting minutes during the late 1981 and early 1982 contained satisfactory detail but recent minutes have deteriorated in quality. The minutes for the April 1982 meeting was missing. This item was discussed during the exit interview. The licensee stated that the decline in the quality of the minutes probably resulted from the misunderstanding of the requirement by the newly assigned person in charge of this area, and that an increased effort would be expanded in this area. Previously identified open item (50-124/79-02-09) remains open.

7. Review and Audit

The records of Reactor Safety Committee (RSC) meetings for April 1981 through August 1982 were reviewed. The inspector verified that the meetings were conducted in accordance with Technical Specification requirements regarding quorum, membership, and meeting frequency. The RSC reviewed procedure changes, unusual incidents and occurrences, recent modifications and maintenance to the coolant systems and recent changes in the facility staff. A review of the RSC audit functions disclosed the following deficiency.

The Radiation Safety Committee is required by Technical Specification 8.2.2 to review and approve conformity of operations with the Technical Specifications. The VPI Radiation Safety Manual, page 1-4, states that the Reactor Safety Committee shall be responsible for assuring that an annual audit of reactor operations, security, SNM inventory and safeguards are performed. For 1981, an annual audit of reactor operations was not documented, therefore no documentation of the Radiation Safety Committee's required review of the conformity of operations with Technical Specifications was available. This is a violation. (50-124/82-01-02)

8. Plant Tour

A tour of the facility was conducted on November 17, 1982 with the reactor shutdown. Housekeeping in the reactor room and the control room were satisfactory. The inspector noted that portable radiation monitoring equipment in the control room and outside the reactor room displayed current calibration stickers. Previous open item (50-124/81-02-01) is considered closed. The inspector identified no discrepancies in this area.

9. Procedures

The VPI Procedure manual was reviewed. Six major categories of procedures are included in the manual. The inspector observed through a review of Reactor Safety Committee meeting minutes and a review of procedure issue dates that procedures are routinely reviewed for accuracy and applicability, and that the content and scope of the procedures was adequate to control safety related operations.

During an inspection conducted January 30 through February 2, 1979, documented in IE Report 79-02, an unresolved item was established regarding procedure IV. 15, the rod drop test procedure. The procedure permitted manual withdrawal of control rods for testing purposes. The procedure was reviewed and approved by the Reactor Safety Committee. The basis for manual withdrawal of the rods for this test had not been established but was to be included in new proposed Technical Specifications as a part of a license renewal application. As of the current inspection, the licensee had not revised the procedure, nor established a basis for the manual rod withdrawal.

Technical Specification 6.2.2 states that the safety rods and shim rods shall each have a reactivity worth of approximately 0.55 percent delta k/k and further that the maximum reactivity input rate of the safety and shim rods shall not exceed 0.02 percent delta k/k/second. Compliance with this Technical Specification during manual withdrawal of a control rod can not be assured. At the exit interview, the inspector discussed the above findings and the basis for the concerns. The inspector stated that Technical Specification 8.2.5 requires the Radiation Safety Committee to review changes to procedures to determine if they constitute an unreviewed safety question. Procedure IV.15 which allows the manual rod withdrawal was approved on March 5, 1979, however, there was no evidence that the effect of this procedure on the margin of safety in Technical Specification 6.2.2 was evaluated. Based on these findings the event appears to represent noncompliance with Technical Specification 8.2.5. (50-124/82-01-03)

10. Surveillance

Several surveillance test procedures were examined for technical content and adequacy. With the exception of Procedure IV.15, discussed in paragraph 9, the procedures and related administrative requirements were adequate. Surveillance requirements were being accomplished on schedule. No areas of noncompliance or deviation were identified.

11. Experiments

The records of "Active Irradiation Requests", "Experiment Activation Plans" and procedures related to routine experiments were reviewed. No items of noncompliance were identified. However, the criteria by which the licensee measures and determines whether an irradiation is a new experiment was not considered sufficiently inclusive by the inspector. The irradiation request form used by the licensee requires review that the experiment is not flammable, corrosive, or explosive, and does not contain special nuclear material. A limit of not more than one curie or a dose rate of not more than 2 REM at 10 cm is also imposed. If the irradiation request meets the above requirements, the experiment is defined as not a new experiment. The irradiation request form does not require evaluation of reactivity effects or experimental failures which might cause rod or fuel problems. The potential for overlooking a problem with an experiment was discussed at the exit interview. The licensee agreed to review their irradiation request procedure and made appropriate changes. This was left as an open item. (50-124/82-01-04)

COLLEGE OF ENGINEERING

VIRGINIA POLYTECHNIC INSTITUTE AND STATE UNIVERSITY

Blacksburg, Virginia 24061

NUCLEAR REACTOR LABORATORY

January 11, 1983

Mr. R. C. Lewis, Director
Division of Project and Resident Programs
U. S. Nuclear Regulatory Commission, Region II
101 Marietta Street, N.W.
Atlanta, GA 30303

Dear Sir:

In response to your letter of December 15, 1982 (Report No. 50-124/82-01), we acknowledge the two violations cited in Appendix A.

With respect to Item A, while the required annual audits were partially carried out, we admit that our documentation of the audits was unsatisfactory. The reason for this violation was the lack of a formal procedure for carrying out the audits. In order to prevent a recurrence of the violation, we propose to initiate a procedure for reactor audit and review. The proposed procedure is included as Attachment I. We believe that this corrective action will prevent a recurrence of the violation. The proposed procedure will be presented to the Reactor Safety Committee on January 31, 1983, so we should be in full compliance after that date.

In regard to Item B, while the Radiation Safety Committee has reviewed the fact that we were manually withdrawing control rods for drop time measurement testing of control rods, we admit we do not have documentation for a safety analysis being performed on manual withdrawal of a control rod. The reason for the error was the fact that without the moderator in the core (procedure requirement), we have a negative reactivity of 30% $\Delta K/K$ inserted in the core. The maximum reactivity any one control rod will insert when pulled is 0.77% $\Delta K/K$. This fact was apparently taken to meet the safety analysis requirements. We now realize we are in violation and propose to correct this by submitting to the Radiation Safety Committee a Safety Analysis Report covering a manual rod withdrawal with moderator out of the core. The proposed Safety Analysis Report is included in Attachment II. We believe this corrective action will prevent any recurrence of the violation. The proposed Safety Analysis Report will go before the Reactor Safety Committee for approval at the 31 January 1983 meeting and we should be in full compliance after that time.

Sincerely yours,

Thomas F. Parkinson

Thomas F. Parkinson, Director
Nuclear Reactor Laboratory

smw
attachments
cc: R. A. Teekell
P. D. Holian

COLLEGE OF ENGINEERING

VIRGINIA POLYTECHNIC INSTITUTE AND STATE UNIVERSITY

Blacksburg, Virginia 24061

NUCLEAR REACTOR LABORATORY

January 11, 1983

MEMORANDUM

TO: Dr. Roger A. Teekell, Chairman, Reactor Safety Committee

FROM: T. F. Parkinson, Director, Nuclear Reactor Laboratory T F Parkinson

SUBJECT: Reply to Nuclear Regulatory Commission Safety Inspection (Report No. 50-124/82-01)

In order to assure compliance with Technical Specifications 8.2.2, 8.2.5, 6.2.2 and the audit requirements as given in the Radiation Safety Manual, I am submitting the following proposals:

ATTACHMENT I; A proposed Procedure for Reactor Audit and Review.

This proposed procedure outlines the steps necessary to perform the annual audit. Also the accompanying audit report sheet will be submitted to the Reactor Safety Committee (RSC) at the completion of the Audit encompassing any discrepancies or other items worth noting and will be signed by the members of the R.S.C. who performed the audit.

ATTACHMENT II; Proposed Safety Analysis Report, Change in Procedure IV.15, "Measurement of total Control Rod Drop Time" and Change in Technical Specifications 6.6.2.

These proposals are being submitted to the N.R.C. in response to the Notice of Violation received from the last safety inspection and will be included as ITEMS on the agenda of the R.S.C. meeting scheduled for 31 January 1933. I suggest that we implement these proposals at that time so that they may be functioning immediately.

smw

Attachments I and II

cc: P. D. Holian

D. R. Krause

ATTACHMENT I

APPROVED/REVIEWED ____________________
Date

____
Chairman, Reactor Safety Committee

III.3 Reactor Audit and Review (annual)

In accordance with the Radiation Safety Manual, annual audits must be conducted in the following areas: Reactor Operations, Security, Special Nuclear Material Inventory and Safeguards. Accordingly, all procedures, Technical Specification operating logs and other documentation pertaining to the safe operation and of the VPI&SU Nuclear Reactor Laboratory shall be audited annually by three (3) members of the Reactor Safety Committee (R.S.C.) not directly associated with reactor operations. These members shall be designated at the general R.S.C. meeting just prior to the quarter in which the audit is to be performed.

All Special Nuclear Material (S.N.M.) shall be inventoried by:

1. The Reactor Supervisor
2. The University Reactor Radiation Safety Officer
3. One of the three members designated by the R.S.C. to take part in the annual audit.

As for the rest of the areas of the audit, the three members designated to perform the audit will perform the audit with only assistance from the reactor staff.

1. The Audit will consist of a review of the following:

   A Reactor Operations

      i. Reactor Operating logs
      ii. Reactor Procedures
      iii. Reactor Operation/R.S.C. Meeting Minutes
      iv. Requalification procedures/documents
      v. Equipment calibration/maintenance data

   B. Security

      i. Security Procedures/documents

   C. Special Nuclear Material

      i. All S.N.M. stored with the VPI&SU Nuclear Reactor Laboratory or assigned to the Nuclear Laboratory Personnel.

   D. Safeguards

      i. Experiment Procedure/documentation
      ii. Annual Report
      iii. Technical Specifications
      iv. Safety Analysis Report

2. The Audit shall be documented by completing the attached report, with the exception of the S.N.M. inventory, which will documented on the S.N.M. report (NRC 742).

APPROVED/REVIEWED__________________
Date

____
Chairman, Reactor Safety Committee

III.3 (continued)

3. The S.N.M. inventory shall consist of actual "sighting" of all Special Nuclear Material listed in the previous S.N.M. report. (NRC 742) inventory, with adjustments made for any transfers since the previous inventory. Irradiated fuel sighting shall consist of inspection of fuel transfer logs, reactor core inventory, and confirmation of storage of the appropriate number of "hot" fuel plates as recorded in fuel storage logs.

   Confirmation of fuel plate serial numbers shall be by inspection of individual serial numbers from not less than two (2) storage locations chosen at random by the consultant member of the S.N.M. inventory team. (for non-irradiated fuel only.)

   A. The formula used for burnup of U^{235} shall be 4.356×10^{-2} grams/Mw-hr. The formula for burnup + transmutation of U^{235} shall be 4.356×10^{-2} grams/Mw-hr. $\times \frac{98+580}{580}$.

      The figures reported on any completed NRC-732 forms shall be checked by the consulting member of the S.N.M. inventory team.

4. Any significant discrepancies (any excess or missing Items in the S.N.M. inventory) shall be reported immediately to the chairman of the Radiation Safety Committee. Should the S.N.M. discrepancy not be readily resolved, the discrepancy shall be reported to the Nuclear Regulatory Commission within 24 hours.

5. The report results (report sheet and NRC-732) shall be presented at the next general R.S.C. meeting along with recommendations for correction of any discrepancies found.

6. A copy of the results of the audit plus corrections performed for any discrepancies found shall be included in the annual report.

APPROVED/REVIEWED ________________
Date

____
Chairman, Reactor Safety Committee

III.3 Reactor Audit and Review Report Sheet

Areas listed below are to be audited annually and will include the subjects listed in Procedure III.3. Any comments will be placed in the space provided and may be continued as necessary. All areas must be filled in. Special Nuclear Material will use NRC-742.

A. Reactor Operations:

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Continued /__/

B. Security:

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Continued /__/

ATTACHMENT I
Page 4

APPROVED/REVIEWED ____________
Date

____
Chairman, Reactor Safety Committee

III.3 Reactor Audit and Review Report Sheet (continued)

C. Safeguards:

Continued /__/

D. Audit completed by:

Reviewed: ____________
Chairman, Reactor Safety Committee

ATTACHMENT II

COLLEGE OF ENGINEERING

VIRGINIA POLYTECHNIC INSTITUTE AND STATE UNIVERSITY

Blacksburg, Virginia 24061

NUCLEAR REACTOR LABORATORY

January 11, 1983

TECHNICAL SPECIFICATION CHANGE PROPOSAL

RE: License No. R-62, Docket No. 50-124
Safety Analysis Report, Manual Withdrawal of a Control Rod Drop Time Measurements and Analysis for Technical Specifications

SUBJECT: Change in Technical Specifications 6.2.2
Maximum Reactivity Input Rate of Safety and Shim Rods

TYPE: Nuclear

PROPOSAL:

Original wording:

6.2.2 The two safety rods and the shim rod shall each have a reactivity worth of approximately 0.55% ΔK/K. The safety rods must be withdrawn sequentially prior to withdrawal of the shim or regulating rods. The maximum reactivity input rate of the safety and shim rods shall not exceed 0.02% ΔK/K/sec. The maximum time for insertion of the shim and safety rods following initiation of a SCRAM signal shall not exceed 0.8 seconds.

New wording:

6.2.2 The two safety rods and the shim rod shall each have a reactivity worth of approximately 0.55% ΔK/K. The safety rods must be withdrawn sequentially prior to withdrawal of the shim or regulating rods. The maximum time for insertion of the safety and shim rods following initiation of a SCRAM signal shall not exceed 0.8 seconds. The maximum reactivity input rate of the safety and shim rods shall not exceed 0.02% ΔK/K/sec. This reactivity input rate (0.02% ΔK/Ksec) is not applicable when the core is devoid of water moderator.

EFFECTS: See Attached Safety Analysis Report

SUMMARY: See Attached Safety Analysis Report

COLLEGE OF ENGINEERING

VIRGINIA POLYTECHNIC INSTITUTE AND STATE UNIVERSITY

Blacksburg, Virginia 24061

NUCLEAR REACTOR LABORATORY

January 11, 1983

SAFEY ANALYSIS REPORT

RE: License No. R-62, Docket No. 50-124

SUBJECT: Manual Withdrawal of a Control Rod for Control Rod Drop Time Measurements and Analysis for Technical Specifications

TYPE: Mechanical/Nuclear

PROPOSAL: We propose that the manual withdrawal of a single control rod under the following conditions:

(1) Reactor Shutdown/Control Rods inserted
(2) Core devoid of water moderator
(3) Dump Valve Interlock enabled and activated;

will not cause a Reactivity Addition Accident,
will not impede or hinder the Reactor Safety Systems,
will not exceed Safety System limiting settings,
will not cause an uncontrolled radioactive release and
that the reactivity input rate of 0.02% ΔK/K/sec.
is not applicable when the the core is devoid of water moderator,

Therefore, this proposed manual withdrawal of a single control rod should be allowed in the Control Rod Droptime Measurement Procedure (IV.15) in order to expedite trouble shooting time prior to dismantling the core.

EFFECTS: The Argonaut Research Reactor at VPI & SU has a shutdown condition defined as: all rods inserted, the water moderator "dumped" to the dump tank and the console key removed. By "dumping" the moderator, we void the core region of coolant/moderator and achieve a negative 30.0% ΔK/K reactivity. The control rods then add an additional 2.16% ΔK/K of negative reactivity. When using Procedure IV.15, the reactor is not technically shutdown; however, as specified in procedure IV.15, the Dump Valve Interlock must be activated and this condition prevents filling the core tanks.

A Reactivity Addition Accident cannot occur because with the moderator "dumped" the core reactivity is negative by 30.0% ΔK/K. Thus with manual withdrawal of a single rod (maximum reactivity of any rod is 0.77% ΔK/K) a shutdown reactivity of -29.24% ΔK/K is maintained.

$$(-30.0\%\ \Delta K/K + 0.77\%\ \Delta K/K = -29.24\%\ \Delta K/K)$$

EFFECTS: (continued)

Therefore, even with a single complete rod ejection the minimum shutdown margin of 0.5% ΔK/K will not be exceeded and a manual rod withdrawal will have no adverse effects on the core.

A manual rod withdrawal with core devoid of water moderator will not impede or hinder any Reactor Safety System owing to the fact that the safety system will be energized, operable and in compliance with any specific provisions listed in Technical Specifications for the VPI & SU Reactor, while the manual withdrawal is taking place.

As shown in the following analysis; no Safety System limiting settings will be exceeded:

(1) Neutron Countrate; not applicable, core devoid of water moderator

(2) Coolant/moderator;

A) Temperature not applicable, core devoid of water moderator

B) Flow

C) Operating level

(3) Reactor Room Ventilation; No effect - fans remain on

(4) Safety Rods 1 & 2 fully withdrawn;

has bypass provision - only withdrawn singly

(5) Reactor period/power set points;

never reached due to -29.24% ΔK/K reactivity in core with core devoid of water moderator

(6) Automatic controller servo set point;

never effected, reactor not operating

(7) Regulating Rod at upper or lower limit;

rod not moved past upper or lower limits - also manual stops present

(8) Activation of manual SCRAM switch (remote/manual);

not applicable - core not operating (-29.24% ΔK/K)

(9) Shield tank level;

not applicable, core not operating

(10) Earthquake SCRAM;

not applicable core not operating

EFFECTS: (continued)

(11) Radiation levels; operable - with core not operating levels will not change

(12) Radiation level fission products monitor;

not applicable - core devoid of water moderator, no flow to detector.

SUMMARY: As mentioned in the previous statements, the reliability or safety of our facility will not be degraded nor will a a safety hazard be posed by manually withdrawing a single rod when the core is devoid of water moderator. By not exceeding the Safety System limiting settings we are protected against an uncontrolled radioactive release and therefore the requirements set forth in the proposal are met. We propose that the manual rod withdrawal be included in the Control Rod Droptime Procedure (IV.15) and that a reactivity input rate of 0.02% $\Delta K/K$/sec. not be applicable when the core is devoid of water moderator.

APPROVED/REVIEWED 3/5/79
Date

[signature]
Chairman, Radiation Safety Committee

IV.15 Measurement of Total Control Rod Drop Time Procedure Revisions

Revision	Date	Description
Rev. 1	3-5-79	1. The procedure was completely rewritten with the following standard format: A. Initial Conditions B. Precautions and Limitations C. Procedure D. Final Conditions
Rev. 2	5-19-80	2. Limitation B.5 was changed so rod maintenance is required if rod drop time exceeds .65 seconds.
Rev. 7		1. Less personnel required for performance; SRO now required. 2. Newer oscilloscope specified. 3. Allows for performance following shutdown. 4. Limitation changes so troubleshooting required if any drop time - exceeds .65, allows for other corrective attempts prior to actual control rod maintenance. 5. Deletes routine performance using manual rod withdrawal. 6. New data sheet.

APPROVED/REVIEWED ____________
Date

____
Chairman, Radiation Safety Committee

IV.15 Measurement of Total Control Rod Drop Time

A. Initial Conditions

Rev. 7

1. Two individuals with the following minimum qualifications are available:

   (a) nuclear senior operator
   (b) reactor operator

2. The reactor control console switch key is installed, the console is on and manned by a reactor operator.

3. The dump valve disable interlock is activated and all control rods are fully inserted.

4. Log any jumpers installed or removed from reactor instrumentation in the jumper log.

5. The graphite stack, core tanks and surrounding core region are at thermal equilibrium at $\geq 130^{\circ}F$.

Rev. 7

6. A Tektronix oscilloscope model 2213 or equivalent is available.

7. If this is performed subsequent to a reactor run shutdown checks have been performed without removing the console key.

   CAUTION: Do not leave the console unattended with the key inserted.

B. Precautions and Limitations

1. This procedure will be coordinated and performed by a senior reactor operator.

2. Observe all appropriate electrical safety precautions.

3. Use a 2 to 3 way AC adapter on the AC input power, to the oscilloscope to insure the oscilloscope power cord is not grounded to earth ground.

4. The total control rod drop time is a Technical Specification limit (para. 6.2.2) and shall not exceed 0.8 seconds.

Rev. 7

5. If control rod drop times exceed .65 seconds, inform the Reactor Supervisor. Check the alignment between the drive mechanism and the shaft. Operate the mechanism and listen for any abnormal sounds which may indicate bad bearings and/or re-alignment of the mechanism. If this still does not correct the problem and the mechanism appears to be functioning properly disconnect it from the control rod shaft. Rotate the mechanism by hand to ascertain if it requires maintenance. If this is required, perform procedures III.2 and IV.16.

APPROVED/REVIEWED ____________
Date

____
Chairman, Radiation Safety Committee

IV.15 C. Procedure

1. Set the controls on the oscilloscope as follows:

   (a) Sync: ext.
   (b) Sweep: 100M.S./cm.
   (c) Stability: preset, adjust as necessary
   (d) Triggering level: + and near 0
   (e) Main Sweep: normal
   (f) Volts/Div.: 1 volt/cm sensitivity minimum

   NOTE: Connection from the trigger input to the reactor control console should be through a short length of shielded cable. Connection from the vertical amplifier input to the console may be through a 10X probe or a short length of shielded cable. The TB numbers refer to terminal connections on the rear of the console.

2. Make the following connections for each measurement:

   (a) oscilloscope probe grounded to TB9-80
   (b) trigger input TB9-3
   (c) vertical amplifier input thru a.01μF capacitor)

      1. safety rod No. 1 TB11-14
      2. safety rod No. 2 TB11-20
      3. shim rod TB11-26

   (d) Just prior to making the drop time test, the intensity control on the oscilloscope should be adjusted for a light trace. This will be a vertical line four or more centimeters high and should be located at the left calibration line on the CRT graticule. The triggering level control should be set on the positive side and as close to zero as is possible without accidental triggering due to noise. When the manual scram button is pushed the oscilloscope sweep circuit will be triggered and an intense trace will proceed from left to right across the screen of the CRT. The point at which the trace collapses to a central line is the end of the drop time.
   (e) NOTE: With the respective jumpers installed allowing rod withdrawal you must manually reset the scram to ensure scram bus continuity (clutch voltage applied).

3. Shim rod:

   (a) Insure safety rods and regulating rod are fully inserted.
   (b) Install a jumper between TB9-75 and TB9-16. This disables the start-up interlock on the safety rods allowing the shim and regulating rod to be withdrawn.
   (c) Insure the regulating rod is maintained fully inserted.
   (d) Withdraw the shim rod fully until both analog and digital indicators verify the rod is fully withdrawn.
   (e) Perform step 5, repeat as necessary.
   (f) Remove the jumper between TB9-75 and TB9-16.

APPROVED/REVIEWED ____________
Date

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Chairman, Radiation Safety Committee

IV.15 C. (continued)

4. Safety rods:

   (a) For safety rod 1 verify all other rods are fully inserted.
   (b) Install a jumper between TB9-75 and TB9-11. This disables the respective start-up interlocks allowing withdrawal of safety rod 1.
   (c) Withdraw safety rod 1 fully until the top light energizes and both analog and digital indicators verify the rod is fully withdrawn.
   (d) Perform step 5, repeat as necessary.
   (e) Remove the jumper between TB9-75 and TB9-11.
   (f) For safety rod 2 verify all other rods are fully inserted.
   (g) Install a jumper between TB9-75 and TB9-14. This disables the respective start-up interlocks allowing withdrawal of safety rod 2.
   (h) Withdrawal safety rod 2 fully until the top light energizes and both analog and digital indicators verify the rod is fully withdrawn.
   (i) Perform step 5, repeat as necessary.
   (j) Remove the jumper between TB9-75 and TB9-14.

5. On command by the senior reactor operator, the reactor operator will scram the reactor. The senior reactor operator will note the drop time by deserving the oscilloscope.

D. Final Conditions

1. Disconnect the oscilloscope from the back of the reactor control console.

2. Remove the reactor control console key. Verify all rods are inserted and complete shutdown checklist.

APPROVED/REVIEWED ____________________
Date

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Chairman, Radiation Safety Committee

NRL-012

REV 7

- VIRGINIA TECH RESEARCH REACTOR -

Total Control Rod Drop Time Data Sheet

IV.15 Step C.6 (time indicated in seconds)

Run No. #	S.R. No. 1	S.R. No. 2	SHIM ROD
1			
2			
3			
Average			

No drop time shall exceed .65 seconds

COMMENTS: Include maintenance performed on any rod if required.

Procedure performed by: ____________________

Date: ____________________

Reviewed by: ____________________

Date: ____________________

COLLEGE OF ENGINEERING

VIRGINIA POLYTECHNIC INSTITUTE AND STATE UNIVERSITY

Blacksburg, Virginia 24061

NUCLEAR REACTOR LABORATORY

January 28, 1983

Mr. R. C. Lewis, Director
Division of Project and Resident Programs
U. S. Nuclear Regulatory Commission, Region II
101 Marietta Street, N.W.
Atlanta, GA 30303

Dear Sir:

This is in reference to our written reply to an inspection conducted at our facility on December 15, 1982. Our reply stated that the necessary changes would be presented to the Reactor Safety Committee prior to January 31, 1983.

Due to the large number of other changes to be presented at this meeting with the required safety analysis, review, and documentation — a January 31 date is not possible.

The Reactor Safety Committee Meeting has been rescheduled to February, 1983 to allow sufficient time to complete the remainder of the desired changes. Following Safety Committee approval, the proposed changes will then be implemented (procedural — not requiring a Technical Specification change) or submitted for approval (those items requiring a Technical Specification change).

Sincerely yours,

Peter D Holian

Peter D. Holian
Reactor Supervisor
Nuclear Reactor Laboratory

smw
cc: Dr. T. F. Parkinson, Director, Nuclear Reactor Laboratory
Dr. R. A. Teekell, Chairman, Reactor Safety Committee

COLLEGE OF ENGINEERING

VIRGINIA POLYTECHNIC INSTITUTE AND STATE UNIVERSITY

Blacksburg, Virginia 24061

NUCLEAR REACTOR LABORATORY

July 12, 1982

Mr. James P. O'Reilly
Region II
U. S. Nuclear Regulatory Commission
101 Marietta St., N.W.
Atlanta, GA 30303

RE: License Number R-62, Docket Number 50-124

Dear Mr. O'Reilly:

This letter is being sent to you at the request of Mr. Austin Hardin. Enclosed for your reference is a copy of my letter to Mr. Hardin dated June 30, 1982, which describes the incident of June 24, 1982. Mr. Hardin has expressed concern as to whether a violation of Technical Specifications occurred. I do not believe that a violation occurred due to the following reason. Section 6.2.4 of our Technical Specifications states, "The reactor shall be subcritical by a minimum margin of 0.5% delta k/k when the safety or shim rod of maximum reactivity worth and the regulating rod are fully withdrawn from the core." According to our latest calibrations (these values were applicable during the incident), the reactivity values applicable to our reactor are given in the table below.

Component	Reactivity Worth
Safety Rod 1	0.67% $\frac{\Delta k}{k}$
Safety Rod 2	0.61% $\frac{\Delta k}{k}$
Shim Rod	0.77% $\frac{\Delta k}{k}$
Regulating Rod	0.108% $\frac{\Delta k}{k}$
Moderator Worth	30.0% $\frac{\Delta k}{k}$
Maximum Excess Reactivity (Including Experiments)	0.398% $\frac{\Delta k}{k}$

The shutdown reactivity margin can be calculated as follows:

$$\left(\frac{\Delta k}{k}\right)_{SD} = \left(\frac{\Delta k}{k}\right)_{controllable} - \left(\frac{\Delta k}{k}\right)_{excess}. \qquad (1)$$

Therefore, for our reactor, the following values are obtained:

$$\left(\frac{\Delta k}{k}\right)_{SD} = 31.76\%$$

The $\left(\frac{\Delta k}{k}\right)_{controllable} = 32.158\% \frac{\Delta k}{k} = \left(\frac{\Delta k}{k}\right)_{rods} + \left(\frac{\Delta k}{k}\right)_{H_2O}$

and $\left(\frac{\Delta k}{k}\right)_{excess} = 0.398\%$. The Technical Specification limit is $0.6\% \frac{\Delta k}{k}$.

As you can see by these values, with Safety Rod 1 stuck out, we were still shut down by a significant margin and well within the 0.5% margin required by Technical Specifications. Additionally, this was a normal shutdown requiring no safety system action, and the standard procedure for a stuck rod was carried out.

If you have any questions or require additional information, don't hesitate to call me.

Sincerely yours,

T F Parkinson

T. F. Parkinson, Director
Nuclear Reactor Laboratory

TFP:jd
Enc.: Letter, T.F. Parkinson (VPI) to Austin Hardin (NRC), June 30, 1982
cc: Dr. R. A. Teekell, Chairman, Reactor Safety Committee
Mr. Austin Hardin, U.S. Nuclear Regulatory Commission
Mr. Peter Holian, VPI Nuclear Reactor Laboratory

June 30, 1982

Mr. Austin Hardin
Region II,
U. S. Nuclear Regulatory Commission
101 Marietta St., N.W.
Suite 3100
Atlanta, GA 30303

Dear Mr. Hardin:

This report is being forwarded to you as per your request following the telephone conversations held on June 24, 1982. It concerns the safety rod one malfunction following a normal reactor shutdown which occurred earlier that day.

Enclosed are a description of the mechanism operation and drawings for your reference.

On the morning of June 24, 1982, a normal reactor start-up was being performed for a training program being conducted for South Carolina Electric and Gas reactor operators. Following withdrawal of safety rod one, the top limit switch failed to pick up. A technician was sent to check the limit switch and it was found to be out of adjustment. Apparently over a period of time, the top mounting plate for the limit switches had shifted slightly.

The limit switch was readjusted, the support switch bracket (Part #138, Dwg. R2-R-214) tightened, and the start-up continued without further incident. <u>However, the adjustment was not rechecked</u>.

During the subsequent normal shutdown safety rod one failed to insert. The immediate actions for a stuck rod were carried out and the rod was manually inserted. Investigation revealed that the rod was maintained in its withdrawn position by the limit switch. The rotating element of the limit switch caused the point of the cam to catch between it and the plate the rotating element itself is mounted to. The plate the rotating element is attached to buckled from the weight of the rod until it came to a rest.

The limit switch was replaced and the rod cycled 10 times to recheck the replacement. Operation was satisfactory.

The proposed corrective action to prevent future reoccurrences of this nature is as follows: (1) The necessity to recheck any reactory related equipment following adjustments will be stressed at the next scheduled reactor operator meeting; (2) Several proposed design changes for actuation of the limit switches are being considered.

If you have any questions or require additional information, don't hesitate to call me.

Sincerely,

T F Parkinson

T. F. Parkinson, Director
Nuclear Reactor Laboratory

TFP:jd
Enc.
cc: Dr. R. A. Teekell, Chairman
Reactor Safety Committee