ANNUAL OPERATIONS REPORT

Nuclear Research Reactor

Virginia Polytechnic Institute and State University

January 1, 1980 - December 31, 1980

by

Alan P. Curtner, Supervisor

Alan P. Curtner

Approved by

Dr. T.F. Parkinson, Director

T F Parkinson

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APPENDICES

I. Reactor Operations

Reactor operating parameters for 1980 were as follows:

Quarter	Jan-Mar	Apr-June	July-Sept	Oct-Dec
Kilowatt hours	22873	23997	14535	8873
Hours critical	268.5	282.3	230.9	123.4
Ar-41 Released, mCi	36894	38511	23328	14241
Number of startups	59	52	51	43
Unscheduled shutdowns	2	1	4	0

Yearly totals

Kilowatt hours	70278
Hours critical	905.1
Ar-41 released, mCi	112974
number of startups	205
unscheduled startups	7

II. Tabulation of Unscheduled Scrams

Momentary loss of power to the building (electrical storms)	6
Spurious 5 second period, electrical noise within the intermediate range instrument	1

III. Reactor Operator Changes

Mr. Leo Eskin successfully completed a senior reactor operator exam and was issued a license effective November 21, 1980. Mr. William J. Bryan and Mr. Douglas H. Rockwell successfully completed reactor operator exams in October 1980. Mr. Bryan was issued a license effective November 21, 1980 and a license for Mr. Rockwell has been withheld pending a final review of his Medical Examination Report.

The following individuals have permanently left the facility:

Mr. C. B. Holland, June 18, 1980
Mr. J. L. Nelson, June 5, 1980

IV. Personnel Changes

Mr. Christopher B. Holland, who was employed as a half-time staff reactor operator, permanently left the facility in June 1980. Mr. Leo Eskin who was also employed half-time, assumed a full time reactor operator position in June 1980.

Mr. Eskin resigned his full time position in September 1980 to begin graduate studies; he plans to maintain his senior operator license while at Virginia Tech.

Mr. Dennis Prater was hired in November 1980 as a full time senior reactor operator, filling the position left vacant by Mr. Eskin. He is currently in operator training and a licensing exam is scheduled for April 1981.

V. Quarterly Scram Time Tests (seconds)

Quarter	June-Mar	Apr-June	July-Sept	Oct-Dec	Average
Safety #1	0.46	0.42	0.48	see	0.45
Safety #2	0.51	0.53	0.53	note	0.52
Shim-Safety	0.43	0.42	0.43	below	0.43

Note: Oct-Dec scram time tests were not performed.

The tests were scheduled for December 20, 1980. The last reactor run for the year was performed on December 19, 1980. The reactor staff was on vacation and the reactor shutdown for the remainder of the year. The tests could not be scheluled earlier due to special testing with a 13-plate fuel element. (See section IX of this report.) Annual in-core maintenance is scheduled for the first week of January 1981 and scram time tests will be performed after that evoluation and prior to the next reactor run.

VI. Health Physics

Area surveys and wipe tests were made on a minimum of a quarterly basis. There were no significant changes in observed radiation levels during the year.

The only radioactive waste released to the environment was Ar-41 through the ventilation stack. The total amounts released per quarter are shown on the operations summary.

VII. Control Rod and Fuel Inspections

The control rods and reactor fuel were inspected during March 1980. There was no evidence of cracking, pitting, or unusual wear or corrosion. Control rod worths, stringerworths, and reactivity input rates were measured. No significant changes were noted.

VIII. Amendment No. 4 to facility license No. R-62

An application for an amendment to Appendix A, Technical Specifications, license No. R-62 was submitted on September 17, 1980. The NRC granted a temporary amendment from December 1 to December 31, 1980 (see Appendix A).

IX. 13-Plate Fuel Element Tests

Experiments with a 13-plate fuel element in various core locations were performed from December 15 to December 19, 1980. Measured reactivity effects are summarized in Appendix B. A complete report will be available in first quarter 1981.

X. Reactor license R-62 Renewal

Reactor license R-62 expired on November 16, 1979. An application for license renewal was submitted in a timely manner allowing continued operation on the existing license until a review of license renewal documentation is completed by the NRC.

The reactor facility submitted the following documents to the NRC in August 1980:

A. Financial Qualifications of the Applicant

B. Environmental Impact Appraisal

C. Safety Analysis Report

D. Technical Specifications

E. Reactor Facility Emergency Plan

F. Operator Requalification Program

G. Physical Security Plan

The NRC indicated by recent correspondence that a review of these documents would be conducted in early 1981.

XI. Facility Changes

A. Back Flow Preventer Valves

Two parallel BFP valves were installed on the 2 inch inlet cooling line to the reactor on July 30, 1979. Subsequent operational problems developed (Appendix C).

The BFP valves were removed from the system on February 5, 1980 by the Physical Plant Department and replaced with straight pipe. The earlier operational problems have been alleviated.

B. Reactor Control Console Changes

The following changes were made during February 1980:

1. The secondary pressure low warning light has been disconnected and replaced with a secondary pressure low annunciator alarm on the new annunciator panel.

2. The negative pressure in the reactor room is sensed and an annunciator now alarms on the reactor control console. This alarm will detect a failure in the reactor room ventilation fans. There is a 30 second delay between loss of negative pressure and the annunciator sounding to provide for normal entering and exiting of the reactor room during reactor operation.

3. The water in process pit warning light has been disconnected and replaced with a water in process pit annunciator alarm on the new annunciator panel. The alarm will sound when the sump in the process pit is approximately 75% full.

C. Cooling Tower

A Marley Cooling Tower model no. 8805A has been installed on the roof of Robeson Hall in November 1980 as part of the continuing 500KW power upgrade program. Piping and electrical conduit installation was begun in December. The existing primary and secondary systems will not be modified until complete review by the Reactor Safety Committee, and if appropriate by the Nuclear Regulatory Commission.

D. Physical Security System

A new intrusion detection system with an alarm at the Campus Security Division Office has been installed in December. This system fulfills new requirements of 10 CFR part 73 (73.67) Physical Protection of Plants and Materials.

XII. New Experiment - Shield Tank

A new experiment was approved by the Reactor Safety Committee and initial measurements began in September 1980. The purpose of the experiments is to measure the integral spectrum in a "dummy" fuel assembly which consists of clusters of UO_2 (natural uranium) fuel rods contained in zircalloy -4 tubes. (See Appendix D for details)

XIII. Equipment Failures

Tabulation of Failures by Month

Jan. - Regulating rod indication meter sticking - repaired
Regulating rod feedback control potentiometer - replaced

Feb. - Power level chart recorder, pinched cable harness - repaired
Chemistry Laboratory heating system pipe rupture, water entered nuclear instrument cableways, dried cables, replaced damaged connectors

May - Radiation Monitoring System, vacuum tube failure - replaced
Power level chart recorder, pinched cable harness - replaced

June - Solid State Relay failure, safety system - repaired
Intermediate Range, compen. volt. power supply - replaced
Source Range preamplifier - replaced

July - Source Range Fission Chamber - replaced
Air Particulate Fission Product Monitor, vacuum pump - replaced
Shim Rod up/down switch - replaced

Sep. - Safety Rod #1 and #2 up switch - replaced

Nov. - Reactor Coolant Flow Monitor, transformer in amp. module - replaced
Radiation Monitoring System, alarm reset switch - replaced

APPENDIX A

Amendment No. 4 to Facility License No. R-62

COLLEGE OF ENGINEERING

VIRGINIA POLYTECHNIC INSTITUTE AND STATE UNIVERSITY

Blacksburg, Virginia 24061

NUCLEAR ENGINEERING GROUP

September 17, 1980

Mr. James R. Miller, Chief
Standardization and Special Projects Branch
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington DC 20555

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

Dear Sir:

In our preliminary analysis for operation at 500 Kw(t), we have determined that the present fuel element configuration will need to be modified to add more reactivity. Our current technical specifications limit us to operations with no more than 12 fuel plates per element and with a maximum k (excess) of 0.6%. We request a temporary waiver of the technical specifications to permit tests of one 13-plate element without exceeding the limit on k (excess). Testing would be carried out at low power (< 1 kw(t)) and should not require more than 30 days. If possible we would like to perform the low-power physics tests during the period from Dec. 1-31, 1980. The specific tests to be done would involve measuring the change in k_{eff} when the 13-plate element is exchanged for a 12-plate element. Sub-critical multiplication measurements will be made during the approach to critical to insure that we do not exceed the limit on k (excess). One or more dummy fuel plates will be substituted in the 12-plate elements as required.

Since we will not exceed the k (excess) limit of 0.6%, we believe that no unreviewed safety hazard will exist. This measurement is classified as a "New Experiment"; therefore, it will be submitted to the Reactor Safety Committee for their prior review and approval.

The attached Safety Analysis indicates that if all twelve 12-plate elements were replaced with 13-plate elements, the thermal utilization change, $\Delta f/f$, would be 0.0039. Assuming that the non-leakage probability does not change, $\frac{\Delta k_e}{k_e} = \frac{\Delta k_\infty}{k_\infty} = \frac{\Delta f}{f}$. Therefore, on the average, substituting one 13-plate element for a 12-plate element would result in

$$\frac{\Delta k_e}{k_e} = \frac{0.0039}{12} = \underline{0.0325\%}$$

For the exchange of a particular fuel element, the above average change would be modified by the statistical weight,

$$W = \frac{\phi_{ij}^{2}}{\bar{\phi}_{core}^{2}}$$, where $\bar{\phi}_{core}$ is the average thermal flux in the core and

ϕ_{ij} is the average flux for a fuel element located at the $(i,j)^{th}$ coordinate. The objective of the proposed tests is to validate the calculation of the change in k_{eff} when a 13-plate element is exchanged with a 12-plate element located at the $(i,j)^{th}$ coordinate position.

If you have any questions about this request, please call me at (703) 961-6510.

Sincerely yours,

TF Parkinson

T. F. Parkinson, Director
Nuclear Reactor Laboratory

Attachment

TFP/nfl

cc: Reactor Safety Committee
A. P. Curtner (Reactor Supervisor)
Mark Embrechts

Safety Analysis

Reactivity Change with 13-Plate Fuel Element

Mark Embrechts

The method of calculation is that used in reference 1; we wish to calculate the change in thermal utilization, f, if all the 12-plate fuel elements are replaced with 13-plate elements. Let f and f' be, respectively, the value of f with 12-plate and 13-plate fuel elements.

$$\text{Then} \quad f = \frac{\Sigma a_1 \int_{v_1} \phi_1(r)dv}{\Sigma a_1 \int_{v_1} \phi_1(r)dv + \Sigma a_2 \int \phi_2(r)dv} \qquad (1)$$

where the subscripts 1 and 2 refer to the fuel and moderator, respectively. For a unit cell in the plate-type fuel assembly,

$$f^{-1} = 1 + \frac{v_2\ \Sigma a_2}{v_1\ \Sigma a_1}\ F + (E-1) \qquad (2)$$

$$\text{where } F = \overline{\phi}_2/\overline{\phi}_1 = \kappa_1\ R_1 \coth \kappa_1\ R_1 \qquad (3)$$

$$\text{and } E = \kappa_2\ (R_2 - R_1) \coth (R_2 - R_1) \qquad (4)$$

In equations 3 and 4, R_1 and R_2 are the half-thicknesses of the fuel plate and cell, respectively;

$$\kappa_1^2 = \Sigma a_1/D_1 \text{ and } \kappa_2^2 = \Sigma a_2/D_2.$$

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(1) Reactor Physics of UTR-10, Report ASEA-19, Advanced Technology Laboratories, June 1957.

The values of the constants are collected in Table I.

Table I

Thermal Utilization Constants

Constant	12-Plate Element	13-Plate Element
R_1	0.040 in. (0.1016 cm)	0.040 in. (0.1016 cm)
R_2	0.240 in. (0.6096 cm)	0.220 in. (0.5500 cm)
Σa_1	2.6532 cm^{-1}	2.6532 cm^{-1}
D_1	0.03857 cm^{-1}	0.03857 cm^{-1}
Σa_2	0.022 cm^{-1}	0.022 cm^{-1}
D_2	0.7767 cm^{-1}	0.7767 cm^{-1}
κ_1	0.1024 cm^{-1}	0.1024 cm^{-1}
κ_2	0.0171 cm^{-1}	0.0171 cm^{-1}
F	0.9997	0.9997
E	1.000025	1.000020
f	0.9601	0.9640

Thus the reactivity change if all 12-plate elements are replaced with 13-plate elements would be

$$\frac{\Delta k_e}{k_e} \doteq \frac{\Delta k_\infty}{k_\infty} = \frac{\Delta f}{f} = \frac{f' - f}{f} = 0.0039$$

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

Received 24 Nov 1980
TH

Docket No. 50-124

NOV 21 1980

Mr. T. F. Parkinson
Director, Nuclear Laboratory
Virginia Polytechnic Institute and
  State University
Room 108 Robeson
Blacksburg, Virginia 24061

Dear Mr. Parkinson:

The Commission has issued the enclosed Amendment No. 4 to Facility License No. R-62 for your Argonaut-type reactor. The amendment consists of changes to the Facility License and to Appendix A of the Technical Specifications in response to your application dated September 17, 1980.

This amendment allows testing of one 13-plate fuel element between December 1 and December 31, 1980, providing the power level remains below 1 KWt and the maximum excess reactivity is limited to 0.6% $\Delta k/k$. This is a temporary amendment which expires on December 31, 1980.

A copy of the related Safety Evaluation is also enclosed.

Sincerely,

James R. Miller, Chief
Standardization and Special
  Projects Branch
Division of Licensing

Enclosures:
1. Amendment No. 4
2. Safety Evaluation

cc w/enclosures:
See next page

cc w/enclosure(s):

Mayor of the City of Blacksburg
City Hall
Blacksburg, Virginia 24061

Mr. J. B. Jackson
Commonwealth of Virginia
Council of the Environment
903 Ninth Street Office Bldg.
Richmond, Virginia 23219

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

VIRGINIA POLYTECHNIC INSTITUTE AND STATE UNIVERSITY

DOCKET NO. 50-124

AMENDMENT TO FACILITY LICENSE

Amendment No. 4
License No. R-62

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

   A. The application for amendment by Virginia Polytechnic Institute and State University (the licensee) dated September 17, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   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.

   F. Publication of notice of this amendment is not required since it does not involve a significant hazards consideration nor amendment of a license of the type described in 10 CFR Section 2.106(a)(2).

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of facility License No. R-62 is hereby amended to read as follows:

   2.C.2 Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 4, 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 the date of its issurance and shall expire on December 31, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller, Chief
Standardization and Special
Projects Branch
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: November 21, 1980

UNIT[illegible] STATES
NUCLEAR REGULATORY COMMISSION
WASHINGT[illegible], D. C. 20555

SAFETY EVALUATION BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 4 TO

FACILITY LICENSE NO. R-62

VIRGINIA POLYTECHNIC INSTITUTE AND STATE UNIVERSITY

DOCKET NO. 50-124

Introduction

By letter dated September 17, 1980, the Virginia Polytechnic Institute and State University (the licensee) requested an amendment to Appendix A of Facility License No. R-62. The requested amendment would grant a temporary waiver of the Technical Specifications to permit tests of one 13-plate element. The proposed testing would be carried out at low power (power level less than 1 KWt), the reactivity would not exceed the technical specification limit of $\Delta k/k$ of 0.6% $\Delta k/k$ and should not require more than 30 days (1 to 31 December, 1980). The objective of the specific tests involved is to measure the change in K_{eff} when the 13-plate element is changed for a 12-plate element. These tests will provide data for preliminary analysis for operation at 500 KW, as the present configuration will have to be modified by adding an additional plate in each subassembly in order to operate at the higher power level.

Background

The current Technical Specifications for the reactor core (p. 4, No. 6.1.1) require that "standard fuel elements shall be of the flat-plate type with twelve plates to each element" and (p. 4, No. 6.1.3) "Maximum excess reactivity above cold clean critical shall be limited to 0.6% $\Delta k/k$ including positive reactivity from experiments." These measurements are classified as a "New Experiment." One or more dummy fuel plates must be substituted in the 12-plate elements as necessary to satisfy the excess reactivity requirement.

Evaluation

To support the request for operation with one 13-plate subassembly, the applicant performed an approximate calculation of the reactivity change (based on thermal utilization only) associated with a loading consisting of 12 subassemblies of 13 plates each. This calculation indicates that the total reactivity worth change should be less than 0.4%. This amendment will allow the determination of the reactivity addition due to one 13-plate element at various positions in the core. The licensee will remove the appropriate number of plates elsewhere to counteract the reactivity increase. In addition, subcritical multiplication measurements will be performed during the approach to critical to insure that the limit on k(excess) is not exceeded. During the proposed measurements, the reactor will be operated at power level less than 1 KWt, hence it is not expected that any other technical specifications related to the thermal quantities will be exceeded.

Environmental Considerations

We have determined that this amendment will not result in any significant environmental impact and that is does not constitute a major Commission action significantly affecting the quality of the human environment. We have also determined that this action is not one of those covered by 10 CFR Section 51.5(a) or (b). Having made these determinations, we have further concluded that, pursuant to 10 CFR Section 51.5(d)(4), an environmental impact statement or environmental impact appraisal and negative declaration need not be prepared in connection with issuance of this amendment.

Conclusions

We conclude that the proposed technical specifications waiver for the experiment with one 13-plate fuel subassembly is acceptable. We base this conclusion on the following:

(1) the $\Delta k/k$ (excess) will not be exceeded from that required by the technical specifications;

(2) the power during the measurements will be less than about 1 KWt;

(3) the technical specification waiver is of limited duration and applicable only to these measurements; and

(4) subcritical multiplication measurements will be made during the approach to critical.

We have further determined that this amendment will not increase the probability of occurrence of an accident analyzed in the Safety Analysis Report, nor does it increase the consequences of such an accident. We have 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 considered, and does not

POOR ORIGINAL

involve a significant decrease in a safety margin, 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 operation in the proposed manner, 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 to the health and safety of the public.

Dated: November 21, 1980

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ATTACHMENT TO LICENSE AMENDMENT NO. 4

FACILITY LICENSE NO. R-62

DOCKET NO. 50-124

Revise Appendix A as follows:

Remove Page	Add Page
4	4

6.0 Reactor Core

6.1 Fuel:

6.1.1 Standard fuel elements shall be of the flat-plate type with twelve plates to each element. For a period of 30 days i.e., December 1 to December 31, 1980 one fuel assembly may contain thirteen fuel plates. Each fully-loaded fuel plate shall be approximately 26-inches long by 3-inches wide by 0.080-inch thick, which includes 0.020-inch aluminum cladding on each side, containing a nominal 22 grams of U-235 as uranium-aluminum alloy. The fuel plates shall be separated by 0.40 inch and mechanically joined at the top and bottom. Half and quarter-load fuel plates and aluminum dummy plates may be used to adjust the core loading. *

6.1.2 Low power fuel elements shall be the same as the standard fuel elements except for 0.010-inch cladding and a different [illegible] lifting attachment. The low power elements shall not be operated above 100 watts.

6.1.3 Twelve fuel elements, loaded six to each core tank, shall make up a core loading. Maximum excess reactivity above cold clean critical shall be limited to 0.6% delta k/k including positive reactivity from experiments.

6.1.4 The moderator temperature coefficient of reactivity shall be negative and have a minimum absolute reactivity value of 6.8×10^{-5} delta k/k/°C at any operating temperature. The moderator void coefficient of reactivity shall be negative and have a core averaged value of 0.184% delta k/k per 1% void.

6.2 Control Rods

6.2.1 Four 1/8" thick boral control rods, 2 safety, 1 shim and 1 regulating of the windowshade type shall be positioned in slots machined in the graphite external reflector adjacent to the outside face and near each outside corner of the core. All control rods shall be inspected for cracks and deformations at least once a year.

*This authorization shall expire on December 31, 1980.

APPENDIX B

13 Plate Fuel Element Measured Reactivity Effects

4.0 Measured Reactivity Effects

4.1 Approach to Critical Experiments

A thirteen plate fuel element loaded with 7 dummy aluminum plates and 6 fully loaded fuel plates was first placed in core location E-4. A centrally located fuel element was replaced since it is in the most reactive region of the core. Shifting the 13 plate element to any other core location after it is fully loaded should result in a decrease in total excess-K for that particular location. A summary of the U-235 loading in all fuel elements is given in table 4.1.3. and 4.1.4.

II.1 Approach to Critical Procedure in the Reactor Procedure Manual was followed. One fully loaded fuel plate was added to the element between each critical mass determination. (tables 4.1.1 and 4.1.2, figures 4.1.1 through 4.1.9). The existing source range fission chamber and a BF_3 detector located in the north beam port were used for obtaining count rates.

Criticality was predicted and did occur with the addition of the eleventh fuel plate. The approach to critical procedure was followed when adding the twelfth and thirteenth fuel plate to insure 0.6% $\Delta k/k$ total excess was not exceeded. (figures 4.1.8 and 4.1.9) Graphite stringers were removed as necessary which prevented exceeding the total excess-K limit.

4.2 Critical Experiments

Reactivity measurements with the fully loaded 13 plate element were performed in core locations E-4, E-5, E-6, W-1, and W-2. The total net gain in excess-K by replacing a 12 plate with a 13 plate element was measured. The total actual excess-K with the 13 plate element in each location was also measured. The total theoretical gain in excess-K if no graphite stringers had been removed was calculated. The results are summarized in table 4.2.1. The total worth of the shim rod and the regulating rod were measured with the element in each core location (table 4.2.1). Integral rod worth was measured at 4", 6", 8", 10", 12", and 16" withdrawal for the reg. rod and shim rod in location E-4 and for the shim rod in locations E-5, E-6 W-1 and W-2. (figures 4.2.1 through 4.2.6).

The original plan was to measure diagonally opposite locations in the core. Five out of six of the locations were measured. The experiment was terminated on location W-3 when it could not be assured a total excess-K of 0.6% $\Delta K/K$ would not be exceeded. The best extrapolated estimate that could be obtained when approaching critical indicated a possible total excess-K of 0.601% $\Delta k/k$. Reactivity measurements were performed to an accuracy of $\pm$ 0.001% $\Delta k/k$ using a micro-processor based reactimeter.

All removable graphite stringers were out of the core. Reducing the existing reactivity of the core would have required disassembling a "hot" 12 plate fuel element and replacing one of the 12 plates with a dummy aluminum plate. Therefore, it was decided to terminate the experiment due to time constraints and the tedious process that would have been encountered in manipulating "hot" fuel.

APPROVED/REVIEWED 8/13/79
Date

II.1 Approach to Critical Procedure Data Sheet

Chairman, Radiation Safety Committee

ELEMENTS PLATES

Run #	Fuel Addition	Total Fuel Loading	Counting Rate CPS Fiss.Ch	Counting Rate ~~μμa~~	Counting Rate CPM Spare Detector	Counting Rate	Reciprocal Multiplication Fiss.Ch	Reciprocal Multiplication ~~μμa~~	Reciprocal Multiplication SPARE DETECTOR	Reciprocal Multiplication	Pred. Crit. Mass	Next Fuel Add.	# Rods Out
1	1 PLATE	11 + 6	30	N/A	583		1.0	N/A	1.0		N/A		0
"	"	"	32	—	625		0.94	—	0.93		N/A		1
"	"	"	65	—	1123		0.46	—	0.52		N/A		2
"	"	"	85	—	1637		0.35	—	0.36		N/A		All
2	1 PLATE	11 + 7	30	N/A	630		1.0	N/A	1.0		N/A		0
"	"	"	33	—	670		0.94	—	0.94		N/A		1
"	"	"	65	—	1289		0.46	—	0.49		N/A		2
"	"	"	100	—	2000		0.30	—	0.31		11 + 11		All
3	1 PLATE	11 + 8	31	N/A	690		1.0	N/A	1.0		N/A		0
"	"	"	33	—	819		0.94	—	0.84		N/A		1
"	"	"	75	—	1446		0.41	—	0.48		N/A		2
"	"	"	110	—	2476		0.28	—	0.28		11 + 12		All
4	1 PLATE	11 + 9	30	N/A	744		1.0	N/A	1.0		N/A		0
"	"	"	32	—	919		0.94	—	0.81		N/A		1
"	"	"	80	—	1717		0.38	—	0.43		N/A		2
"	"	"	170	—	3521		0.18	—	0.21		11 + 11		ALL

ELEMENTS PLATES

TABLE 4.1.1

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APPROVED/REVIEWED 8/13/71
Date

II.1 Approach to Critical Procedure Data Sheet

Chairman, Radiation Safety Committee

ELEMENTS PLATES

Run #	Fuel Addition	Total Fuel Loading	Counting Rate Fiss.Ch (CPS)	~~µµa~~	Spare Detector (CPM)		Reciprocal Multiplication Fiss.Ch	~~µµa~~	SPARE DETECTOR		Pred. Crit. Mass	Next Fuel Add.	# Rods Out
5	1 PLATE	11 + 10	31	N/A	856		1.0	N/A	1.0		N/A		0
"	"	"	41	N/A	1098		0.76	—	0.78		N/A		1
"	"	"	100	—	2118		0.31	—	0.40		N/A		2
"	"	"	290	—	6985		0.29	—	0.23		11 + 11		All
6	1 PLATE	11 + 11	33	N/A	950		1.0	N/A	1.0		N/A		0
"	"	"	36	N/A	1220		0.92	N/A	0.78		N/A		1
"	"	"	120	—	2985		0.27	—	0.32		11+11		2
													All
													0
													1
													2
													All

ELEMENTS PLATES

TABLE 4.1.2

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Log of Fuel in Reactor

listed quantities in grams

Slab "A" ↑ **Thermal Column**

W-1 Plate#	Mass	W-2		W-3		W-4		W-5		W-6	
TDE-29	22.46	11-7	21.39	9-12	21.75	9-4	21.75	10-1	22.04	4-27	21.49
TDB-05	21.53	13-1	22.67	12-31	22.38	4-31	21.52	10-2	22.22	13-27	22.58
TDG-08	21.50	11-9	21.39	12-2	22.27	12-23	22.46	10-4	22.13	4-16	21.51
TDB-17	22.42	13-2	22.61	2-31	21.48	2-20	21.48	10-5	22.04	13-28	22.57
TDA-06	22.37	11-12	21.37	12-3	22.23	12-26	22.43	10-7	22.22	4-21	21.51
TDB-30	22.42	13-4	22.61	3-1	22.03	12-28	22.43	10-9	22.32	12-1	22.61
TDG-06	21.87	4-29	21.56	12-19	22.06	2-24	21.48	2-30	21.48	9-28	22.32
TDS-14	22.35	13-8	22.62	12-8	22.26	9-3	21.75	10-12	22.22	12-9	22.55
TDE-04	22.25	4-28	21.53	9-2	21.75	12-30	22.44	10-14	22.13	11-16	21.38
TDB-23	22.27	13-11	22.63	12-15	22.27	2-25	21.48	10-18	22.22	12-13	22.47
TDB-22	22.24	11-10	21.62	10-10	21.76	12-4	22.35	10-19	22.22	11-13	21.37
TDS-11	21.81	13-12	22.61	12-16	22.26	12-5	22.33	4-23	21.34	12-22	22.42

A1	265.49	A2	264.61	A3	264.50	A4	263.9	A5	264.58	A6	264.78

E-1		E-2		E-3		E-4		E-5		E-6	
13-21	22.58	11-19	21.39	12-6	22.31	4-15	21.47	11-21	21.38	10-22	22.22
3-2	22.06	13-31	22.64	2-26	21.48	13-14	22.55	13-13	22.63	10-24	22.22
3-3	22.01	4-8	21.42	12-7	22.36	4-17	21.47	11-5	21.48	10-25	22.13
3-4	22.08	13-3	22.51	2-27	21.48	13-15	22.59	13-18	22.64	10-26	22.22
3-5	22.21	4-9	21.47	10-31	23,13	4-19	21.50	11-18	21.43	10-27	22.13
3-7	22.14	13-5	22.60	2-28	21.48	13-20	22.56	13-19	22.61	4-14	21.48
3-8	22.04	4-10	21.42	12-18	22.40	4-20	21.49	11-30	21.45	10-29	22.22
3-9	21.99	13-7	22.58	3-6	23.24	2-32	21.48	13-23	22.62	10-30	22.04
3-10	22.12	4-12	21.44	2-29	21.48	13-22	22.60	11-28	21.55	10-11	22.04
3-12	22.01	13-9	22.58	12-21	22.38	4-22	21.47	13-25	22.62	10-32	22.13
3-13	22.03	10-28	22.04	12-25	22.35	4-26	21.48	11-31	21.55	3-19	21.95
3-14	22.14	13-10	22.53	12-27	22.35	13-26	22.58	13-30	22.61	3-20	21.97

B1	265.41	B2	264.62	B3	266.44	B4	26[illegible] 24	B5	264.57	B6	264.75

Slab "B" ↓ Shield Tank

TABLE 4.1.3

13 PLATE FUEL ELEMENT U-235 LOADING

Plate No.	Serial No.	Grams, U-235
1	TDA20	21.95
2	TDE21	22.56
3	TDJ07	21.87
4	TDH04	21.92
5	TDE07	21.88
6	TDK09	22.68
7	TDK06	22.54
8	TDG17	22.55
9	TDJ05	21.83
10	TDB14	21.97
11	TDC23	22.52
12	TDA04	21.96
13	TDE26	22.50
		288.73 gr. total

Table 4.1.4

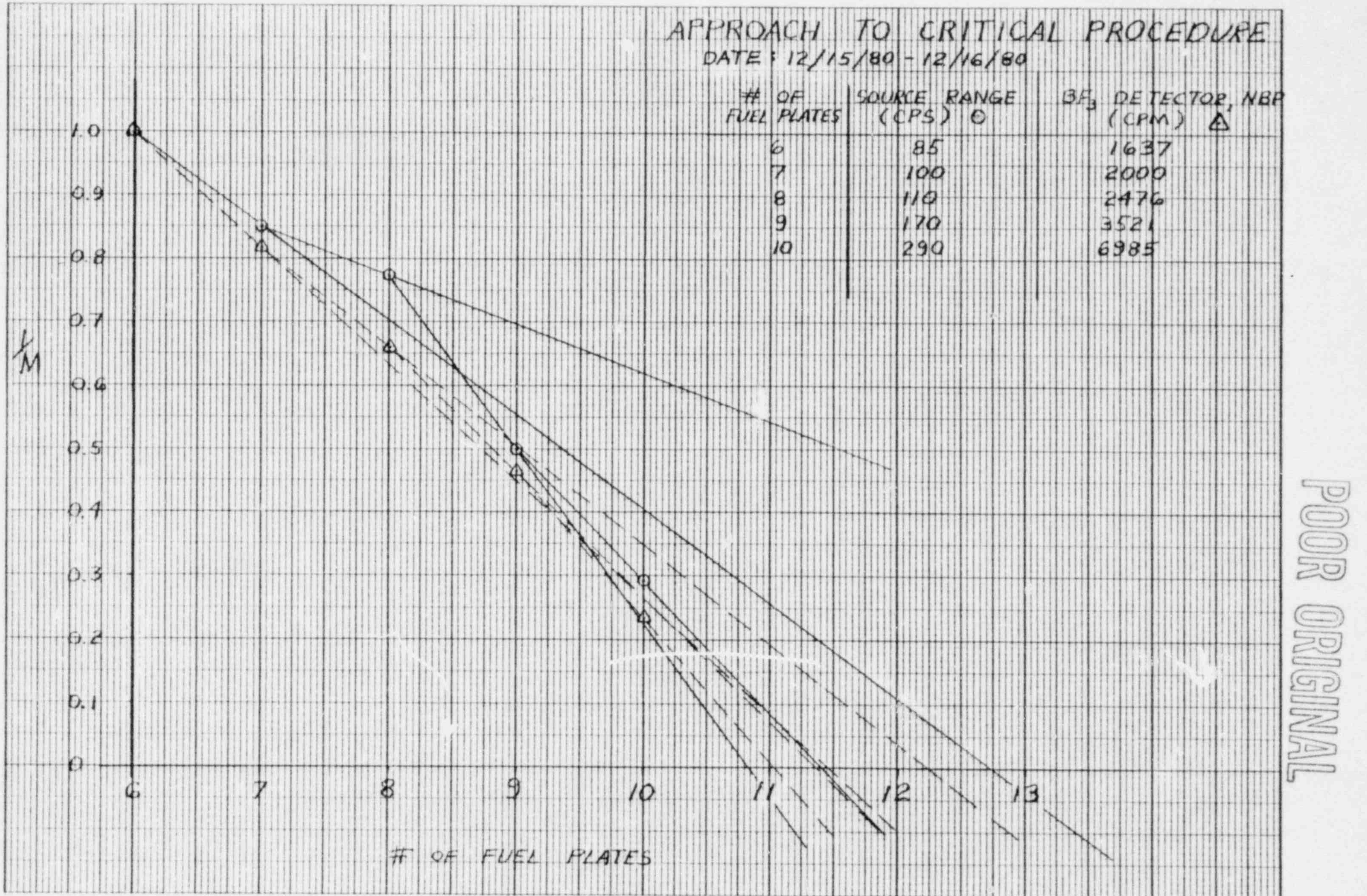

APPROACH TO CRITICAL PROCEDURE
DATE: 12/15/80 - 12/16/80
OF FUEL PLATES
SOURCE RANGE (CPS) ⊙
BF_3 DETECTOR, NBP (CPM) △
6 85 1637
7 100 2000
8 110 2476
9 170 3521
10 290 6985
1/M
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
6
7
8
9
10
11
12
13
OF FUEL PLATES

FIG. 4.1.1

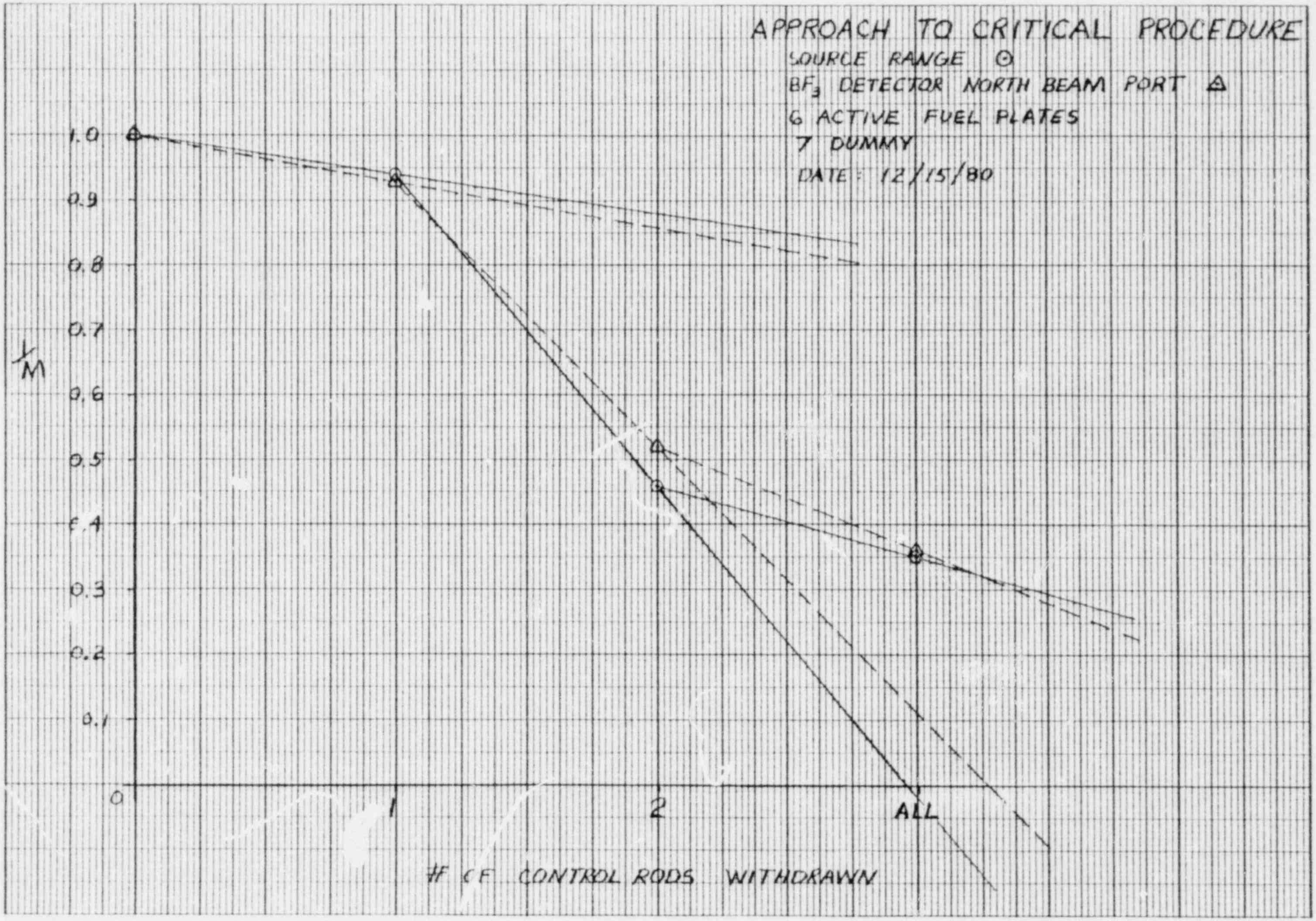

APPROACH TO CRITICAL PROCEDURE
SOURCE RANGE ⊙
BF_3 DETECTOR NORTH BEAM PORT △
6 ACTIVE FUEL PLATES
7 DUMMY
DATE: 12/15/80
1/M
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
1
2
ALL
OF CONTROL RODS WITHDRAWN

FIG. 4.1.2

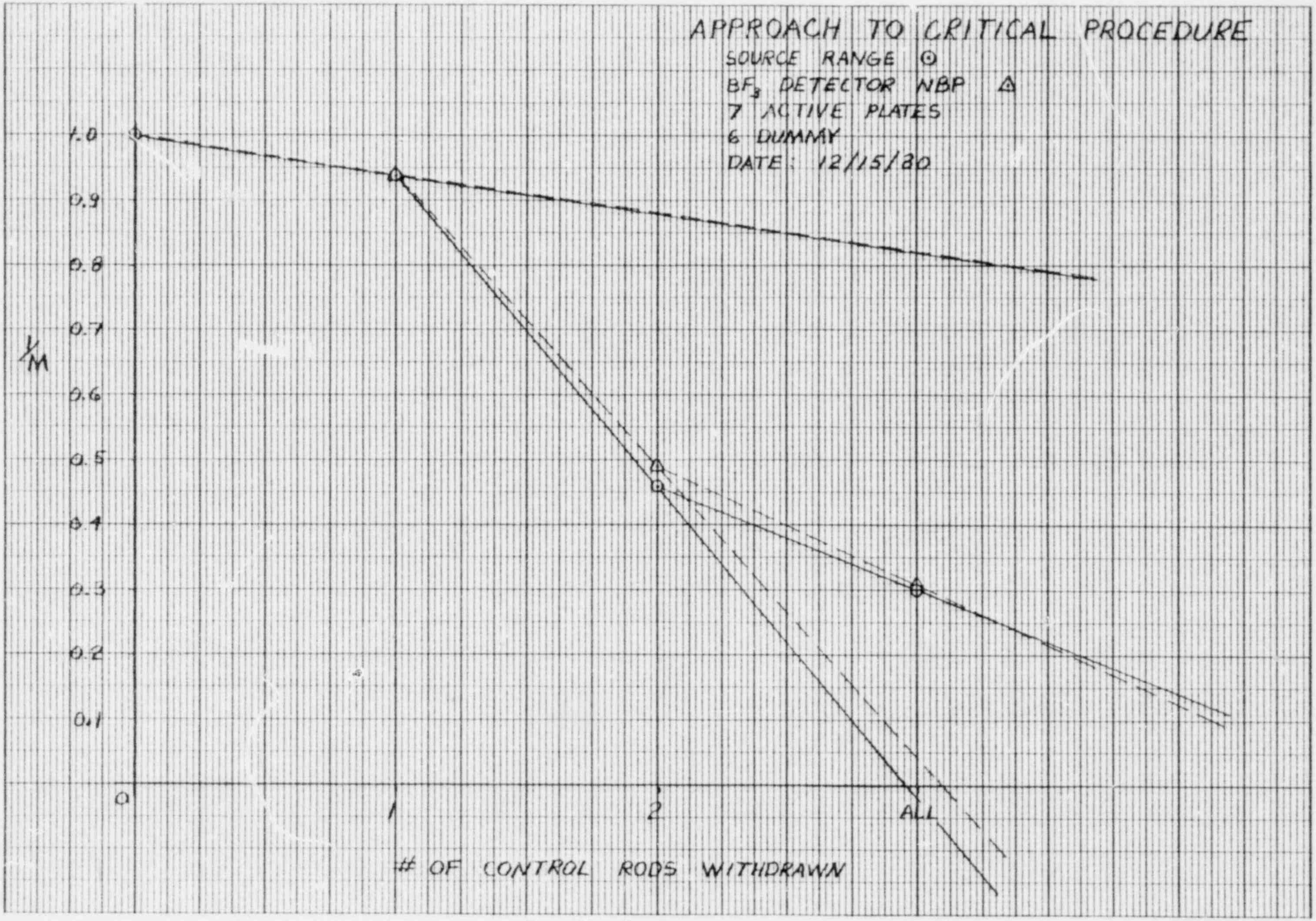
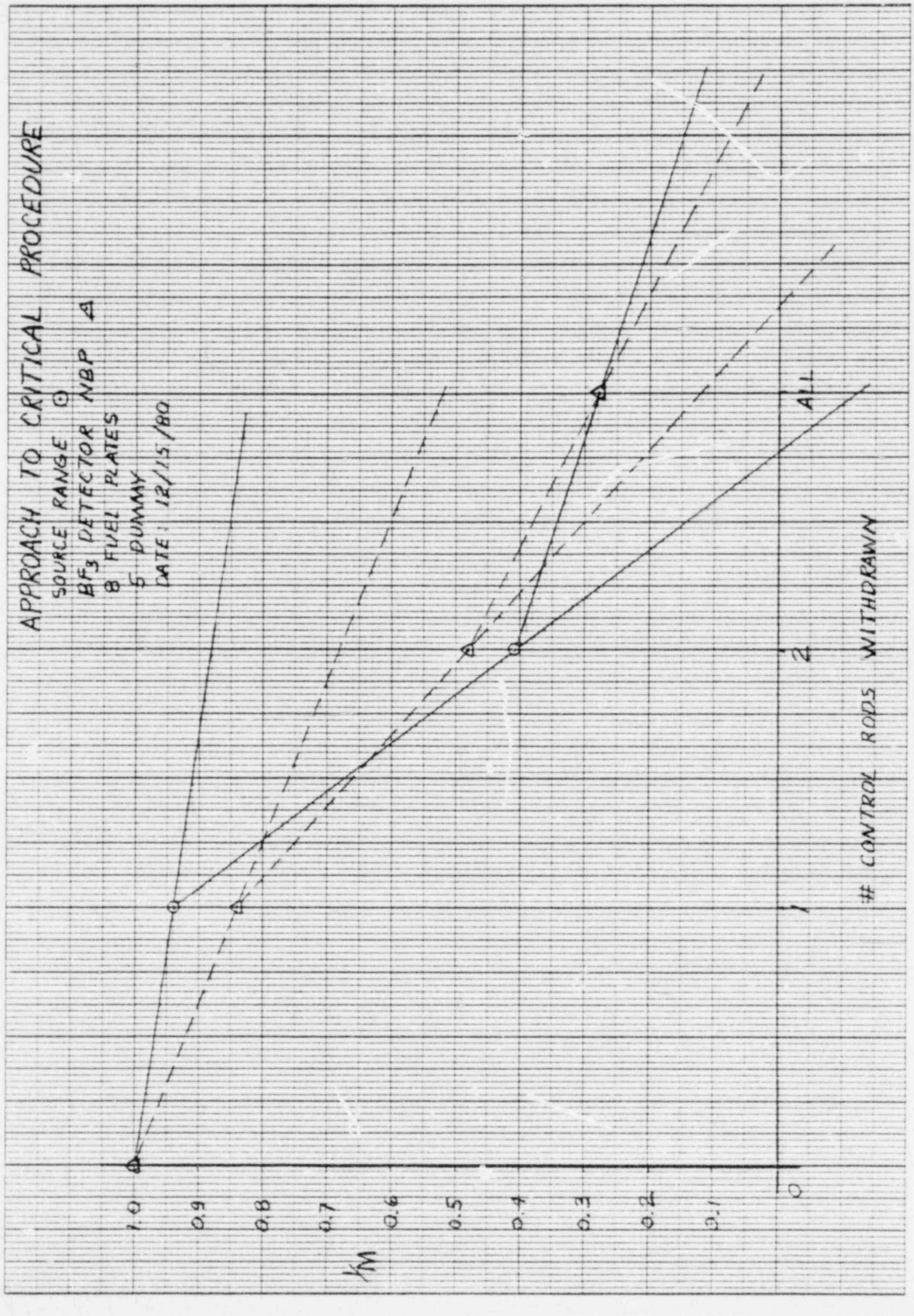

APPROACH TO CRITICAL PROCEDURE
SOURCE RANGE ⊙
BF_3 DETECTOR NBP △
7 ACTIVE PLATES
6 DUMMY
DATE: 12/15/80
1/M
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
1
2
ALL
OF CONTROL RODS WITHDRAWN

FIG. 4.1.3



APPROACH TO CRITICAL PROCEDURE
SOURCE RANGE ⊙
BF_3 DETECTOR NBP △
8 FUEL PLATES
5 DUMMY
DATE: 12/15/80
1/M
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
1
2
ALL
CONTROL RODS WITHDRAWN

FIG. 4.1.4

POOR ORIGINAL

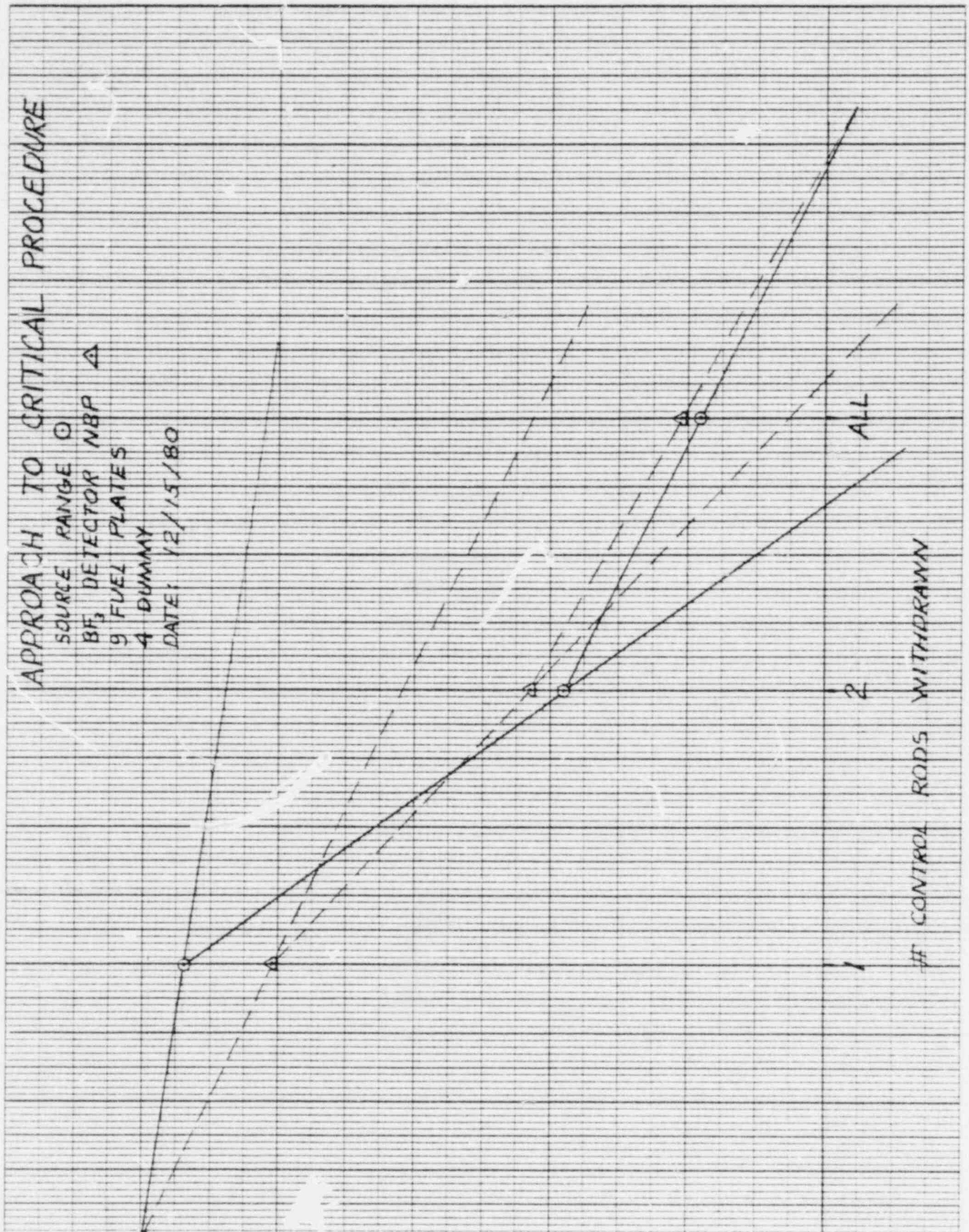


FIG. 4.1.5

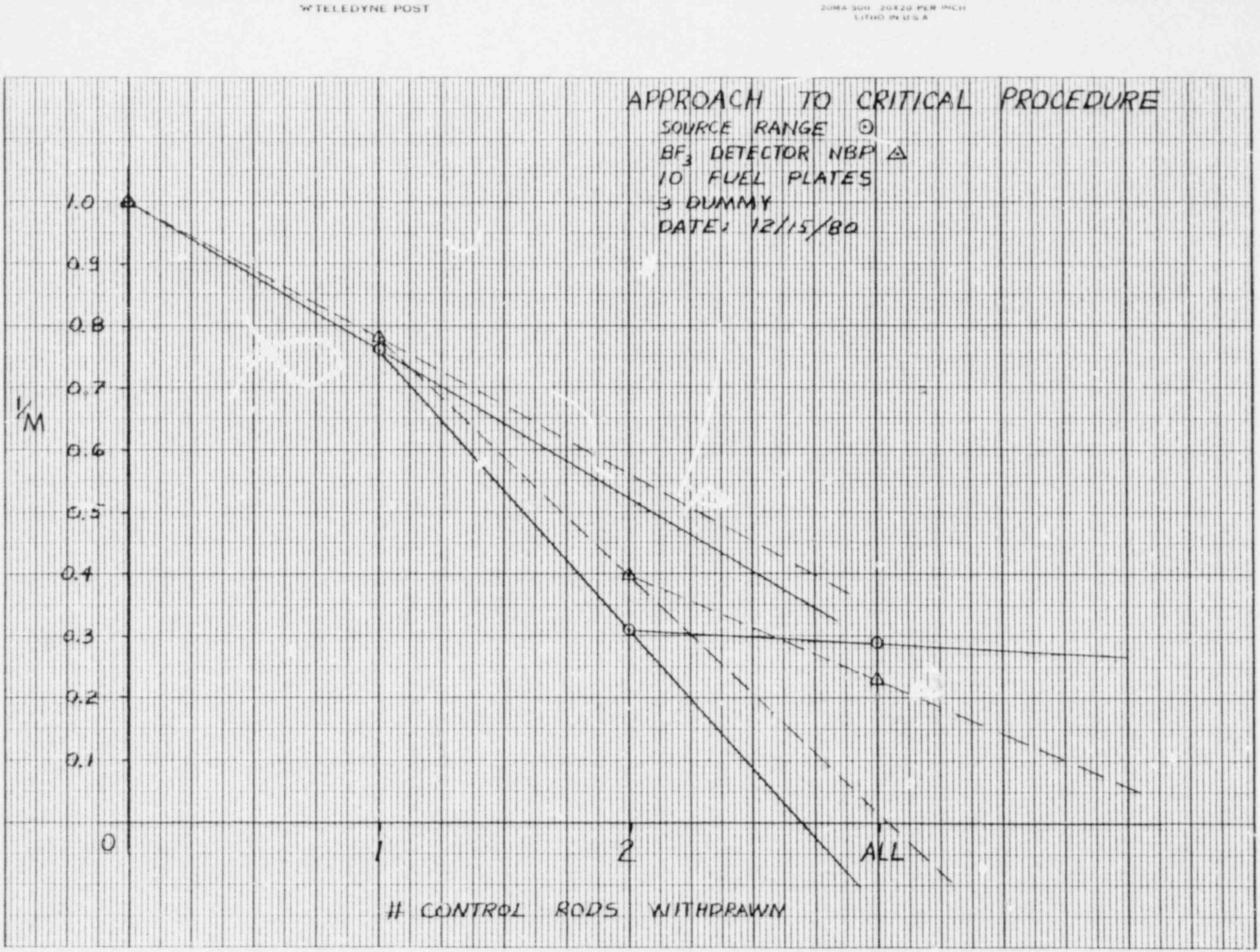

W TELEDYNE POST
20MA-500 20X20 PER INCH
LITHO IN U.S.A.
APPROACH TO CRITICAL PROCEDURE
SOURCE RANGE ⊙
BF_3 DETECTOR NBP △
10 FUEL PLATES
3 DUMMY
DATE: 12/15/80
1/M
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
1
2
ALL
CONTROL RODS WITHDRAWN

FIG. 4.1.6

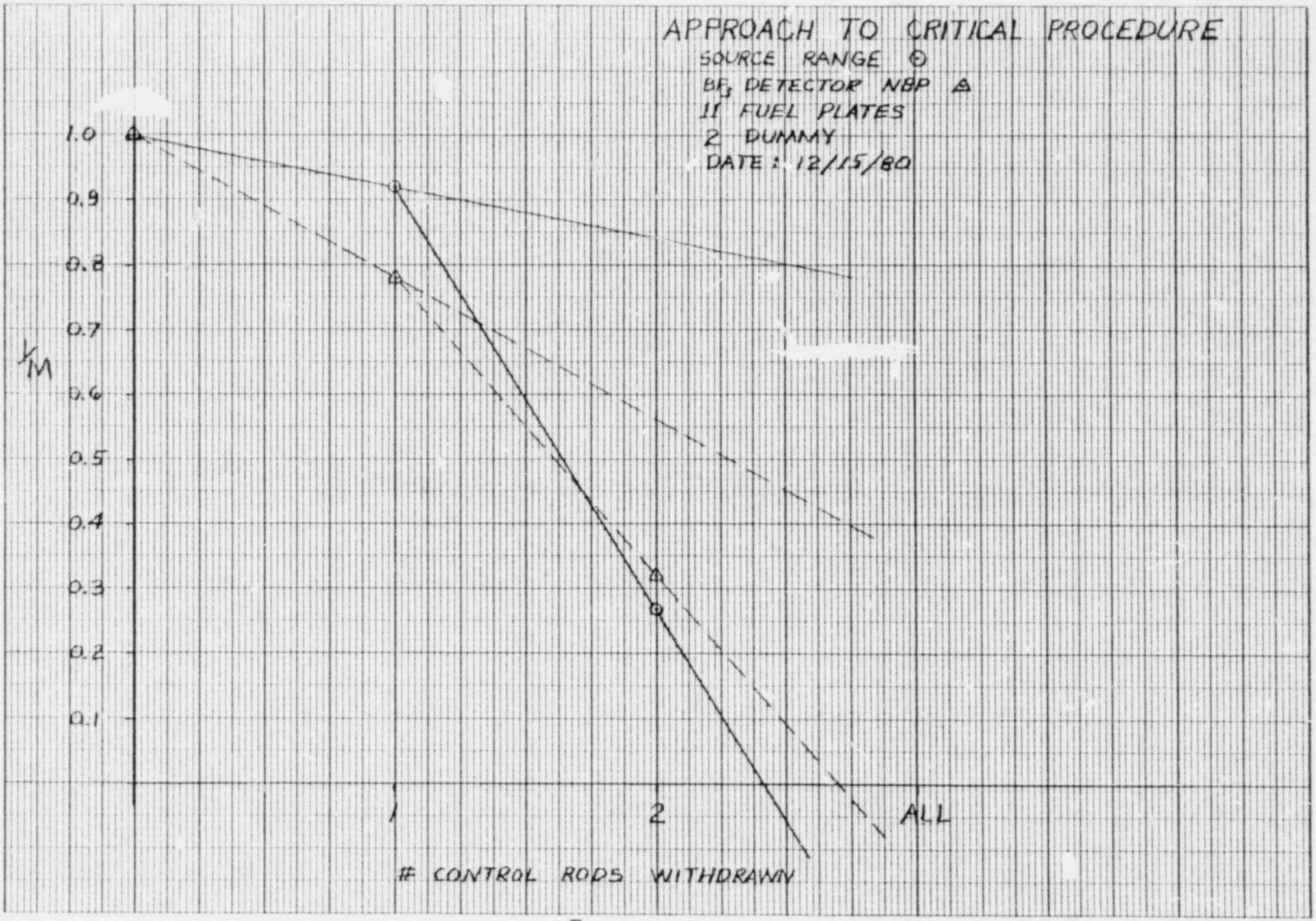

APPROACH TO CRITICAL PROCEDURE
SOURCE RANGE ⊙
BF_3 DETECTOR NBP △
11 FUEL PLATES
2 DUMMY
DATE: 12/15/80
1/M
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
1
2
ALL
CONTROL RODS WITHDRAWN

FIG. 4.1.7

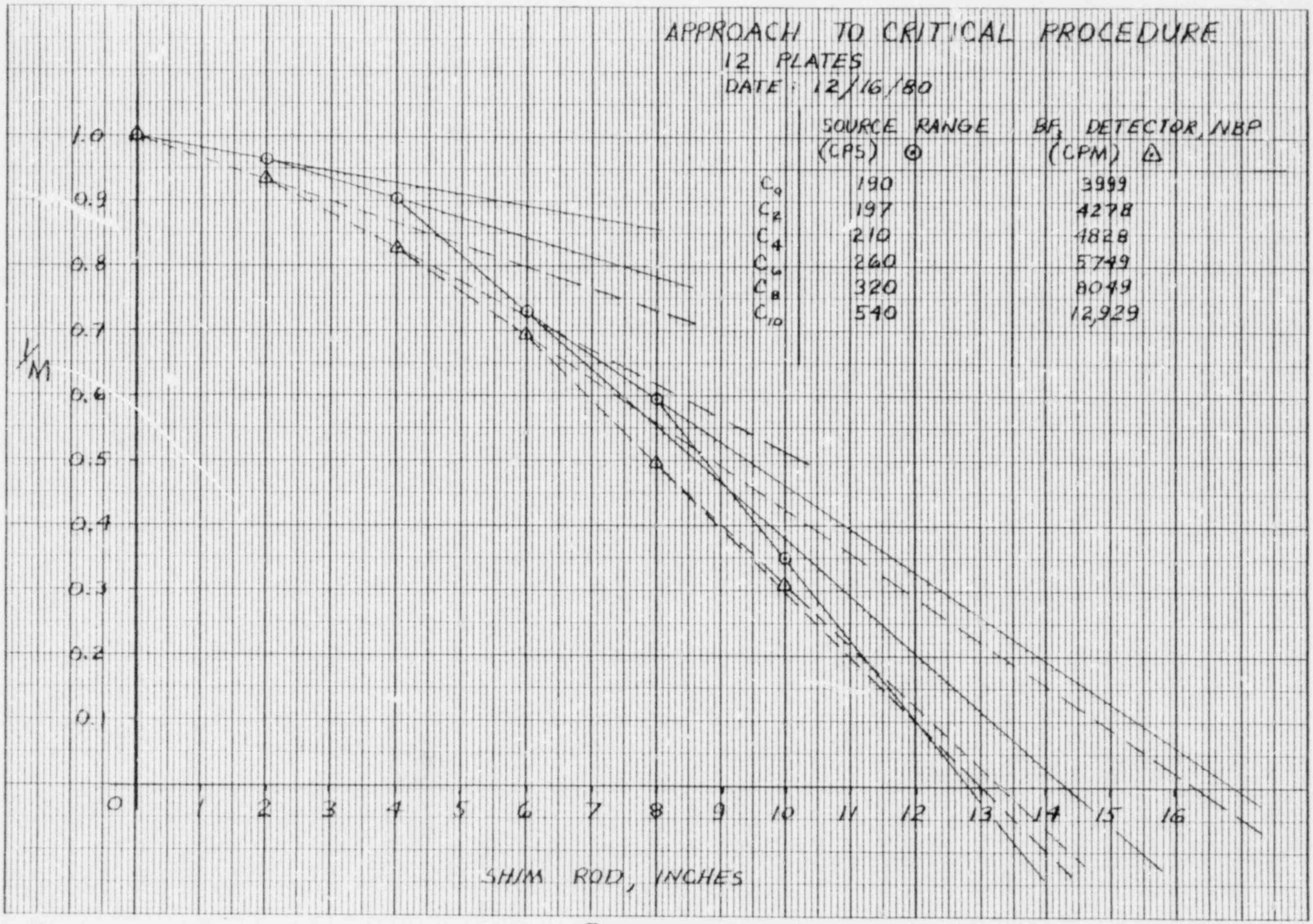

APPROACH TO CRITICAL PROCEDURE
12 PLATES
DATE: 12/16/80
SOURCE RANGE (CPS) ⊙
BF3 DETECTOR, NBP (CPM) △
C0 190 3999
C2 197 4278
C4 210 4828
C6 260 5749
C8 320 8049
C10 540 12,929
1/M
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
SHIM ROD, INCHES

FIG. 4.1.8

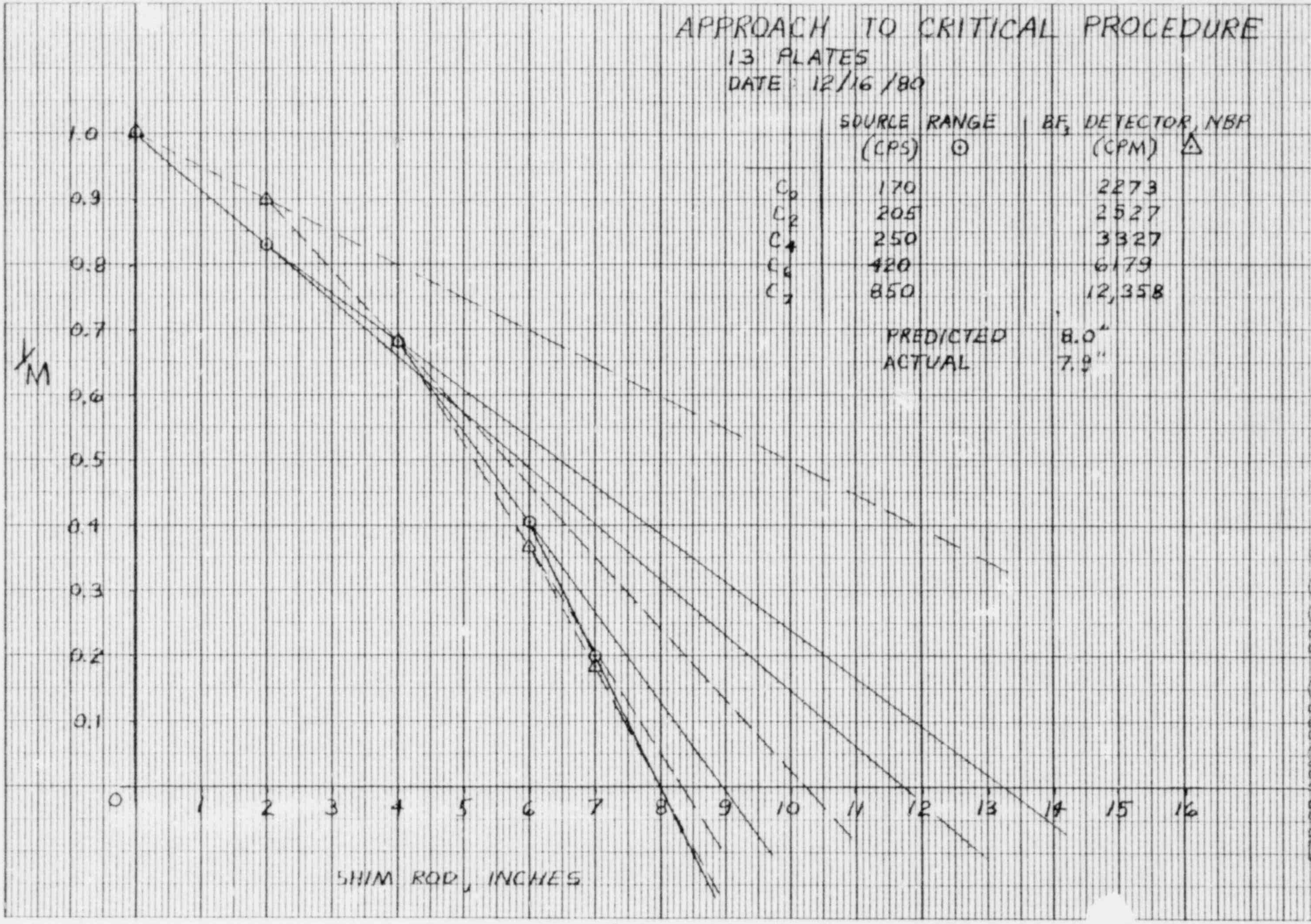

APPROACH TO CRITICAL PROCEDURE
13 PLATES
DATE: 12/16/80
SOURCE RANGE (CPS)
BF3 DETECTOR, NBP (CPM)
C0 170 2273
C2 205 2527
C4 250 3327
C6 420 6179
C7 850 12,358
PREDICTED 8.0"
ACTUAL 7.9"
1/M
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
SHIM ROD, INCHES

FIG. 4.1.9

MEASURED REACTIVITY EFFECTS, 13 PLATE ELEMENT

all values in %ΔK/K

accuracy: ± 0.001 %ΔK/K

Core Location	Total Net Gain, Replacing 12 plate Element With A 13 Plate Element	Total Theoretical Excess-K If No Stringers Removed For compensation	Total Excess-K 13 Plate Element In Core, (Maximum Allowed .6%)	Total Shim Rod Worth	Total Regulating Rod Worth
F [illegible]	0.304	0.783	0.568	0.888	0.115
E-5	0.326	0.805	0.590	0.878	0.113
E-6	0.146	0.625	0.410	0.800	0.111
W-1	0.103	0.582	0.471	0.756	0.119
W-2	0.160	0.639	0.383	0.682	0.126

Table 4.2.1

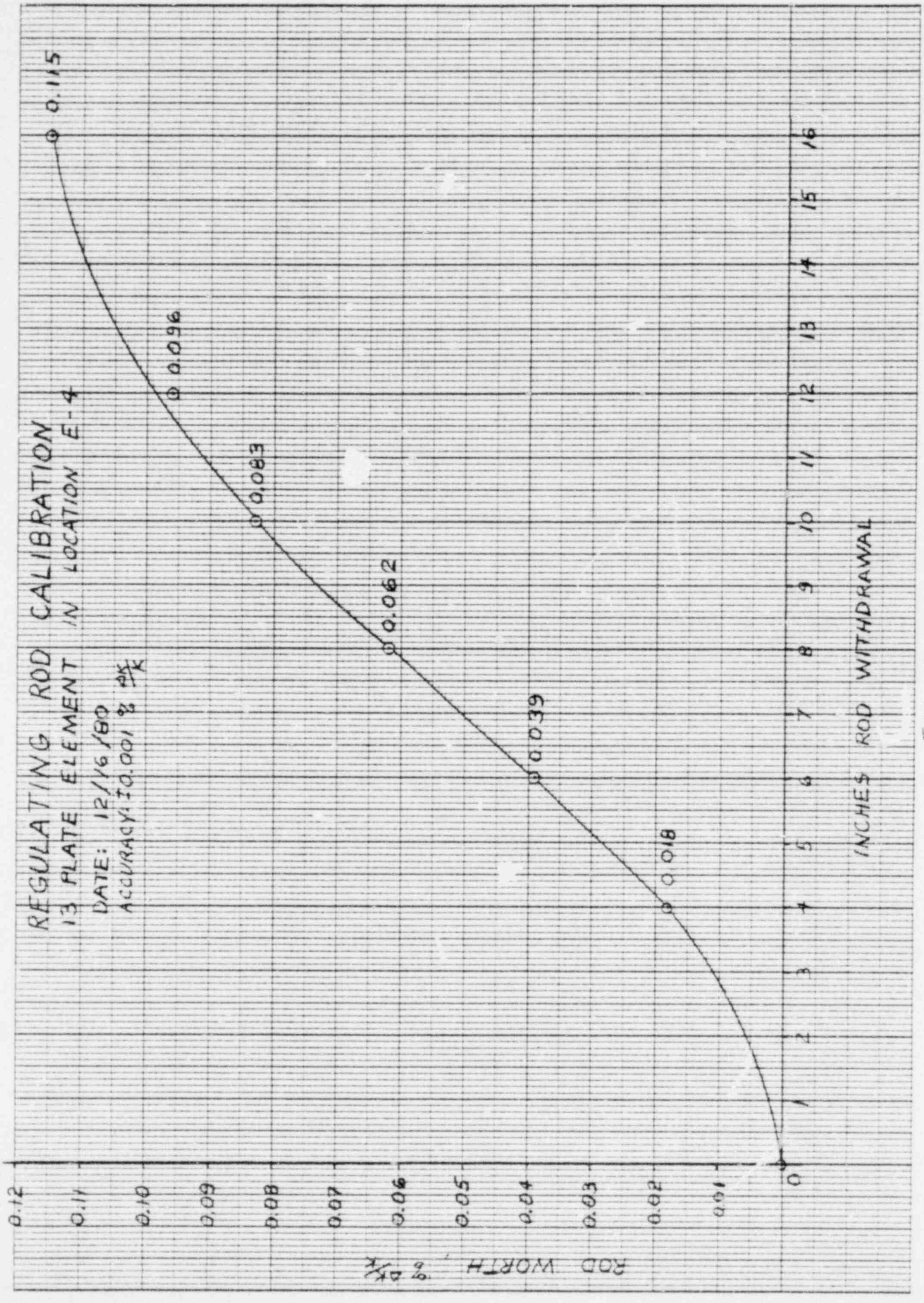

REGULATING ROD CALIBRATION
13 PLATE ELEMENT IN LOCATION E-4
DATE: 12/16/80
ACCURACY: ±0.001 % Δk/k
0.115
0.096
0.083
0.062
0.039
0.018
INCHES ROD WITHDRAWAL
ROD WORTH, % Δk/k
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
0.01
0.02
0.03
0.04
0.05
0.06
0.07
0.08
0.09
0.10
0.11
0.12

FIG. 4.2.1

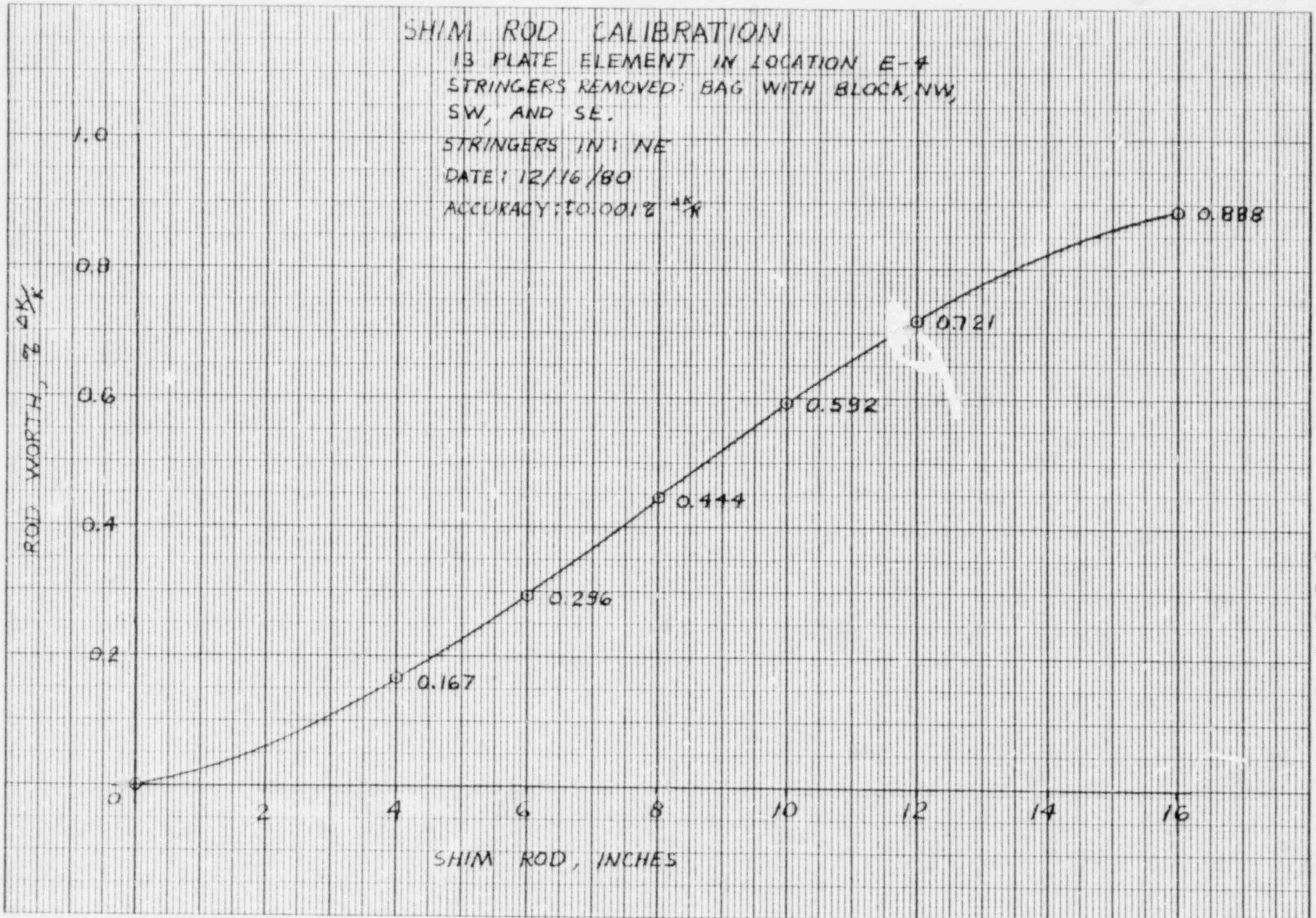

SHIM ROD CALIBRATION
13 PLATE ELEMENT IN LOCATION E-4
STRINGERS REMOVED: BAG WITH BLOCK, NW, SW, AND SE.
STRINGERS IN: NE
DATE: 12/16/80
ACCURACY: ±0.001% ΔK/K
0.888
0.721
0.592
0.444
0.296
0.167
ROD WORTH, % ΔK/K
1.0
0.8
0.6
0.4
0.2
0
2
4
6
8
10
12
14
16
SHIM ROD, INCHES

FIG. 4.2.2

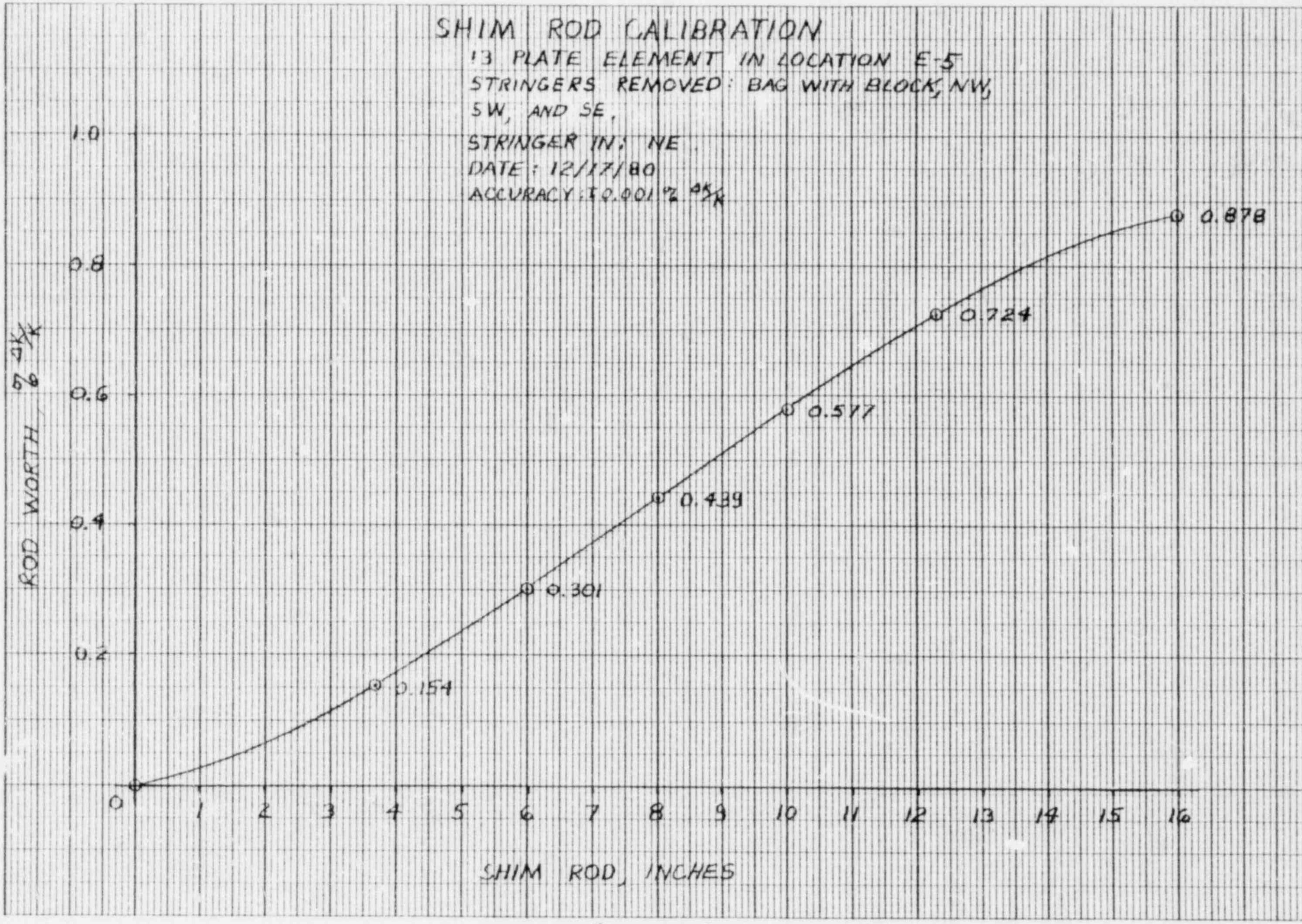

SHIM ROD CALIBRATION
13 PLATE ELEMENT IN LOCATION E-5
STRINGERS REMOVED: BAG WITH BLOCK, NW, SW, AND SE.
STRINGER IN: NE
DATE: 12/17/80
ACCURACY: ±0.001 % Δk/k
ROD WORTH % Δk/k
1.0
0.8
0.6
0.4
0.2
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
SHIM ROD, INCHES
0.154
0.301
0.439
0.577
0.724
0.878

FIG. 4.2.3

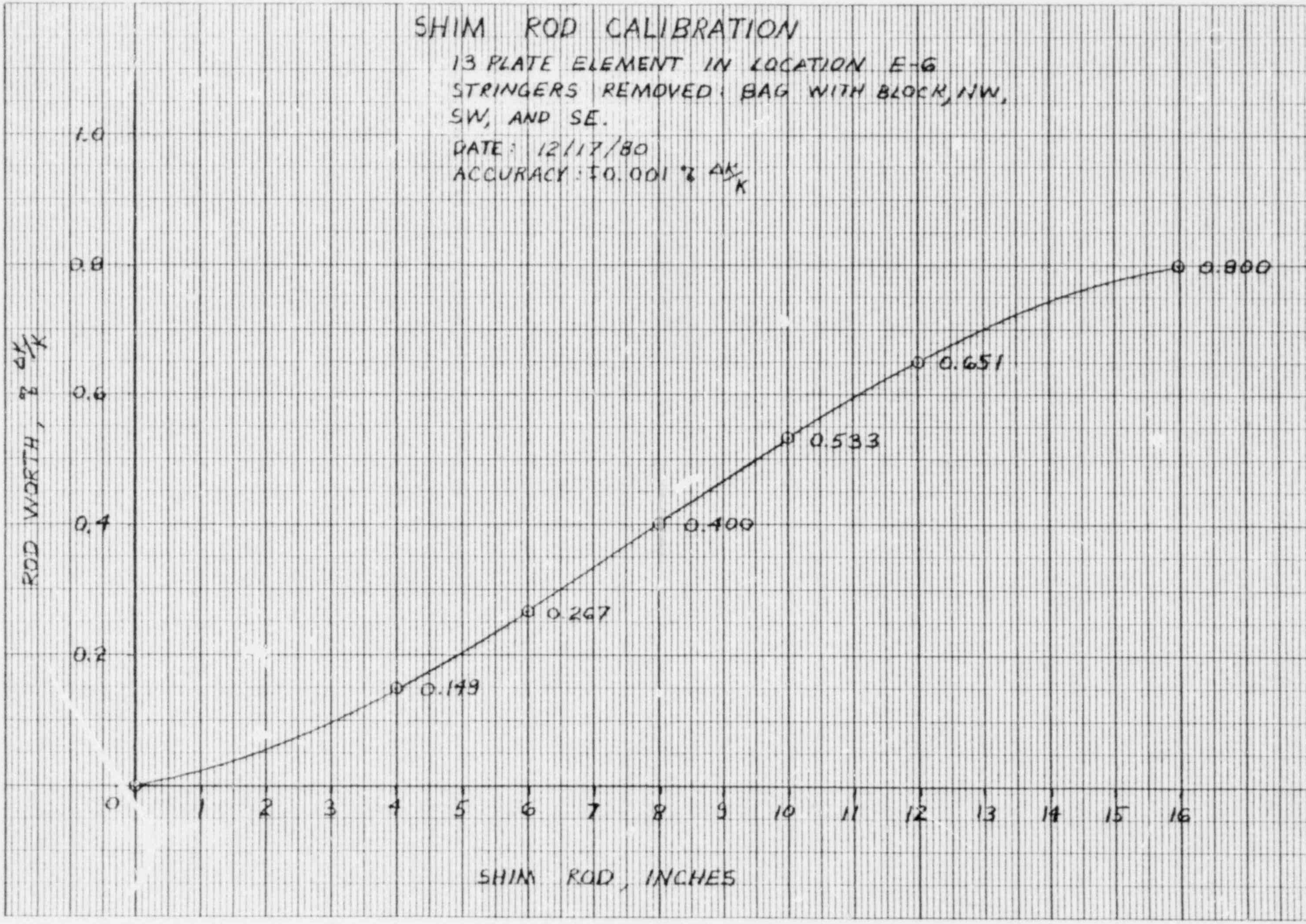

SHIM ROD CALIBRATION
13 PLATE ELEMENT IN LOCATION E-6
STRINGERS REMOVED: BAG WITH BLOCK, NW,
SW, AND SE.
DATE: 12/17/80
ACCURACY: ±0.001 % Δk/k
ROD WORTH, % Δk/k
1.0
0.8
0.6
0.4
0.2
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
0.149
0.267
0.400
0.533
0.651
0.800
SHIM ROD, INCHES

FIG. 4.2.4

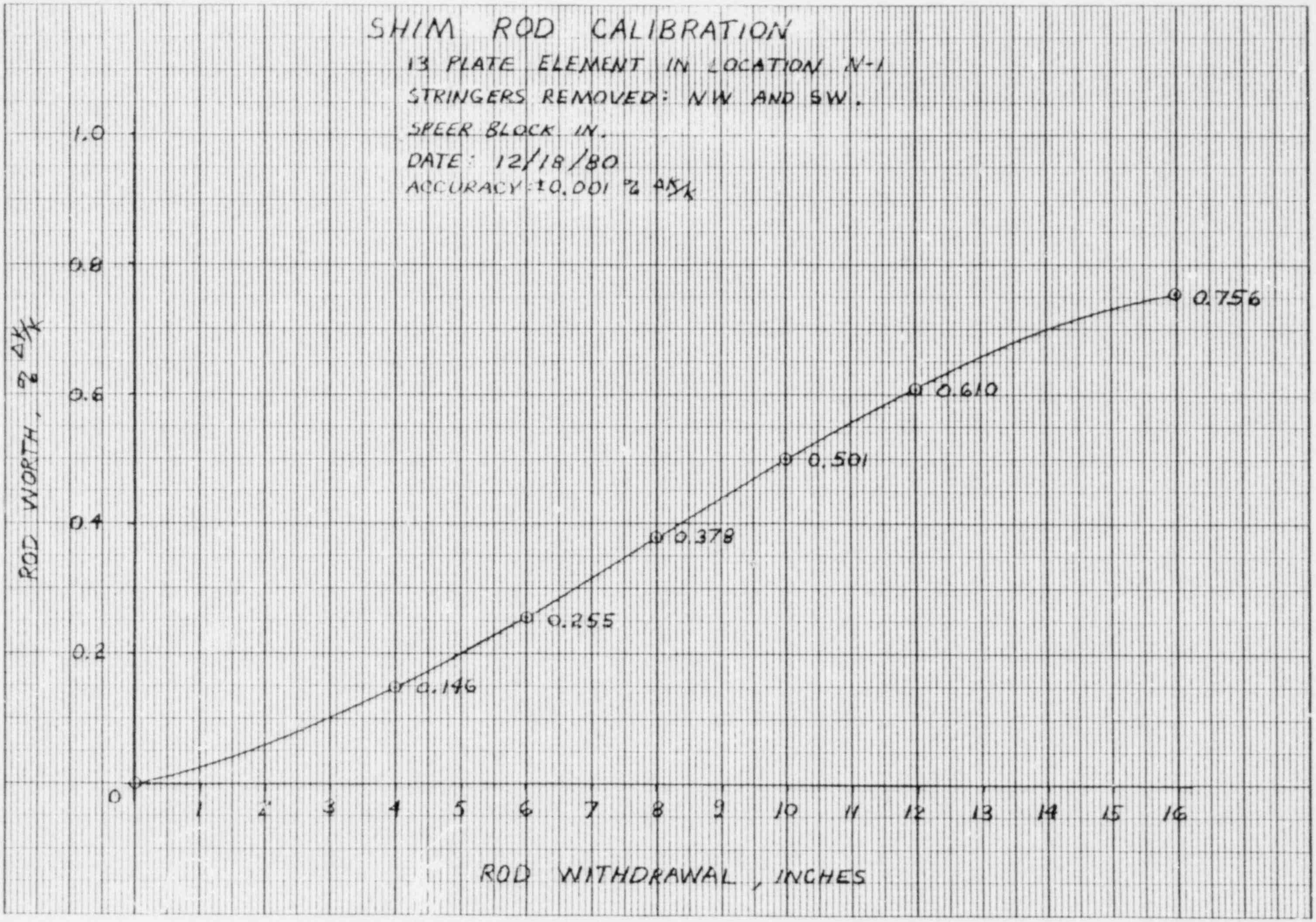

SHIM ROD CALIBRATION
13 PLATE ELEMENT IN LOCATION N-1
STRINGERS REMOVED: NW AND SW.
SPEER BLOCK IN.
DATE: 12/18/80
ACCURACY ±0.001 % ΔK/K
ROD WORTH, % ΔK/K
1.0
0.8
0.6
0.4
0.2
0
0.756
0.610
0.501
0.378
0.255
0.146
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
ROD WITHDRAWAL, INCHES

FIG. 4.2.5

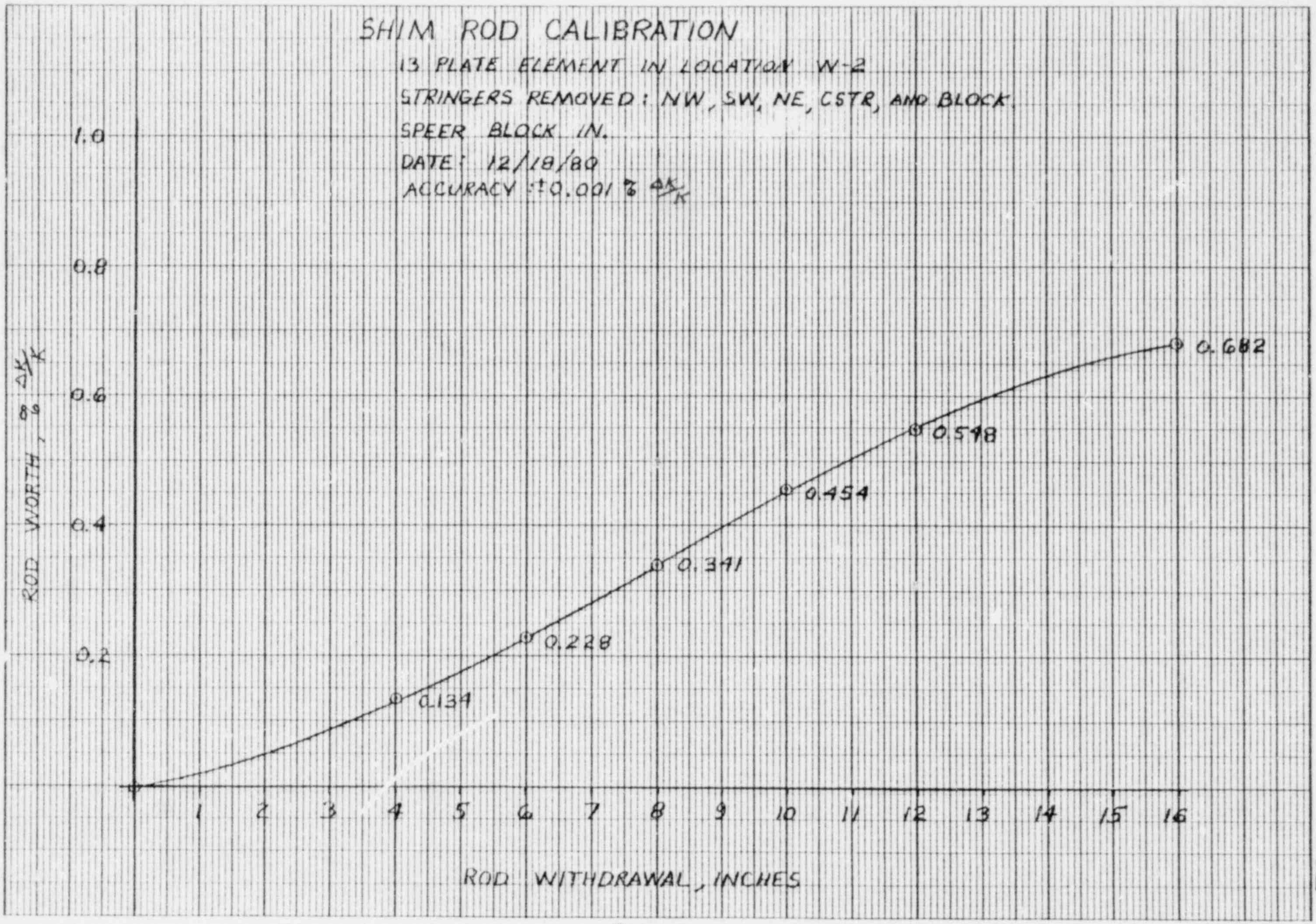

SHIM ROD CALIBRATION
13 PLATE ELEMENT IN LOCATION W-2
STRINGERS REMOVED: NW, SW, NE, CSTR, AND BLOCK.
SPEER BLOCK IN.
DATE: 12/18/80
ACCURACY ±0.001 % Δk/k
ROD WORTH, % Δk/k
1.0
0.8
0.6
0.4
0.2
0.682
0.548
0.454
0.341
0.228
0.134
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
ROD WITHDRAWAL, INCHES

FIG. 4.2.6

APPENDIX C

Back Flow Preventer Valves

file - outgoing

COLLEGE OF ENGINEERING

VIRGINIA POLYTECHNIC INSTITUTE AND STATE UNIVERSITY

Blacksburg, Virginia 24061

NUCLEAR ACTIVATION ANALYSIS LABORATORY

January 4, 1980

Mr. James R. Clemens
Physical Plant Department
Mechanical Services
6A Maintenance Building
Campus

Dear Mr. Clemens:

On July 30, 1979 Physical Plant installed 2-parallel 1½ inch back flow preventer valves on the 2 inch inlet cooling line that supplies water to the heat exchangers required to remove heat generated in the operation of the nuclear research reactor located in Robeson Hall.

The installation of this size back flow preventer has adversely affected the operation of the nuclear research reactor when operated at 100% power. The back flow preventer valves have sufficiently reduced the pressure and flow of secondary cooling water to the reactor such that temperature limitations imposed on us by our technical specifications were reached requiring us to reduce power to less than 95% on certain days during the remaining summer months. 100% power operation is required for performing Neutron Activation Analysis, a major on-going analytical service used by many researchers on campus and numerous other outside organizations around the country.

I anticipate we will not be able to operate the nuclear reactor at its full potential to provide needed research services during the months of May through August because of this reduction in cooling capability. I would desire that the back flow preventer valves be completely removed, restoring the system to its original configuration. However, it is my understanding now that they are required by new federal regulations. So I would like to propose an alternate solution.

I respectfully request that the 2-parallel 1½ inch back flow preventers be replaced with 2-parallel 3" or greater back flow preventers before May, 1980 in an attempt to alleviate this problem.

I might add we noted a temperature increase when a back flow preventer was installed on Robeson Hall water supply earlier in the year. This was before back flow preventers were installed on the individual supply of water to the reactor. This building back flow preventer pushed the temperature to just below our operating limit. When the other valves were installed in series with the building valve, it compounded the problem and increased our temperature beyond the limit we could satisfactorily operate.

Your attention to this problem would be greatly appreciated.

Sincerely yours,

Alan P. Cutner
Reactor Supervisor

APC/fbl

cc: T. F. Parkinson
A. K. Furr
A. H. Krebs
J. B. Jones

APPENDIX D

New Experiment in the Shield Tank

COLLEGE OF ENGINEERING

VIRGINIA POLYTECHNIC INSTITUTE AND STATE UNIVERSITY

Blacksburg, Virginia 24061

NUCLEAR ACTIVATION ANALYSIS LABORATORY

April 8, 1980

To: Reactor Safety Committee

From: T. F. Parkinson and H. J. Boado

Subject: Safety Analysis of New Experiment in the Shield Tank

I. Introduction

It is proposed to perform a series of experiments in the reactor shield tank. The purpose of the experiments is to measure the integral spectrum in a "Dummy" fuel assembly which consists of clusters of UO_2 (natural uranium) fuel rods contained in Zircalloy-4 tubes. Initial experiments will be done with D_2O moderator while subsequent experiments will utilize other moderator-coolants such as H_2O and benzene (C_6H_6). Integral spectra will be determined by activating bare and cadmium-covered foils inserted inside a few of the fuel rods and in the moderator. The types of foils to be used include such materials as In, Au, Lu, U-235 and Pu-239, i.e., both 1/v and non-1/v detectors.

The experiments will be under the supervision of T. F. Parkinson. Co-investigators include H. J. Boado (Research Associate) and R. Cogow (Graduate Research Assistant). All these personnel have passed the exam to qualify for working with radioactive materials.

II. Description of the Experiments

A drawing of the D_2O tank containing the Dummy fuel assembly is shown in Fig. 1. This tank will be positioned adjacent to the graphite duct in the shield tank as shown in Fig. 2 and Fig. 3. After an irradiation is completed, the D_2O tank will be placed on a platform suspended from the side of the shield tank wall. (See Fig. 2). Fuel rods containing foils will then be removed, after monitoring the radiation levels from the UO_2 fuel. Individual foils will then be recovered and counted using either a NaI (Tl) scintillation detector, a Ge (Li) detector or a G-M counter.

III. Safety Considerations

A. Effects on Reactor Operations

Since fission neutrons will be produced in the fuel assembly, it is necessary to insure that the experiments will have negligible effect on reactor operations. Prior work using the pulsed neutron technique has shown that k_{eff} is low for the fuel assembly [1]. Values obtained were 0.30 ± 0.03 and 0.111 ± 0.004, respectively, for H_2O and D_2O moderators. The fuel assembly is isolated from the reactor core by 2 meters of graphite so that any coupling with the reactor core will most likely be negligible. A reactivity balance will be performed to insure that the coupling is negligible. However, the location of the Dummy fuel assembly adjacent to the reactor neutron detectors may cause sufficient flux perturbation to affect the detector readings. It should be noted, however, that prior experiments[2] have demonstrated that very strong absorbers placed next to these detectors did not affect their readings. Nevertheless, measurements will be made of detector readings at several power levels with and without the Dummy fuel assembly in the shield tank. From a comparison of the readings of detectors in the shield tank and in the thermal column, it can be determined whether the flux perturbation due to the Dummy assembly is significant. If significant differences are noted then a graphite slab will be inserted between the Dummy assembly and the reactor neutron detectors as shown in Figs. 2 and 3. The measurements will then be repeated and the thickness of graphite necessary to isolate the Dummy assembly from the neutron detectors will be determined. All subsequent experiments with the Dummy assembly will then be done with the appropriate thickness of graphite.

B. Radiological Safety

Radiation levels due to the irradiated Dummy assembly will be carefully monitored after each irradiation to insure that personnel exposures will be minimized. Typical irradiations are expected to be for 30 minutes at a power level of 100 KW or less.

A "worst-case" calculation has been performed of the radiation levels to be expected from the irradiated UO_2 pellets. We assumed that the fuel pellets were irradiated for 1 hour at a power level of 100 KW, corresponding to an incident thermal neutron flux of 4×10^{10} cm^{-2} sec^{-1}. A one-hour wait time was then assumed. The estimates of the gamma activities and radiation levels resulting from fission products and from Np -239 were made for a single UO_2 pellet. Each fuel tube contains 50 pellets and the entire DUMMY assembly contains 37 fuel tubes. In order to evaluate the <u>total</u> activity, it is necessary to know the average flux in the assembly. This was estimated from the relation

$$\bar{\phi} = \frac{2\pi \phi_o \int_{z=o}^{z=L} e^{-\mathcal{K}z} dz \int_{r=o}^{r=R_1} J_o\left(\frac{\alpha r}{R}\right) r\,dr}{\pi R^2 L} \tag{1}$$

where

R_1 = 5.3 cm (the radius of the UO_2 cluster)
R = 18.0 cm (the extrapolated radius)
L = 70.0 cm (the extrapolated length)
$\mathcal{K}$ = 0.091 cm^{-1} (the inverse relaxation length)
α = 2.404

The detailed calculation of the activity and dose rate from a single UO_2 pellet are given in the Appendix. The results are A_1 = 52 mCi and (DR) = 3.4 m Rem/hr at 10 cm. Then the <u>total</u> activity and dose rate were evaluated from equations (2) and (3):

$$A_{Tot} = A_1 \frac{\bar{\phi}}{\phi_o} N$$

$$(DR)_{Tot} = (DR)_1 \frac{\bar{\phi}}{\phi_o} N$$

here N = the total number of pellets and $\bar{\phi}/\phi_o$ = 0.15 from eqn. 1. The final estimates of total activity and dose rate are:

$$A_{Tot} = 14.4 \text{ Ci}$$

$$(DR)_{Tot} = 0.94 \text{ Rem/hr}$$

In order to validate the calculations, a single UO_2 pellet will be irradiated for a known time at a known flux. The radiation levels from the UO_2 pellet will then be measured for several wait times and compared to the calculated dose rates.

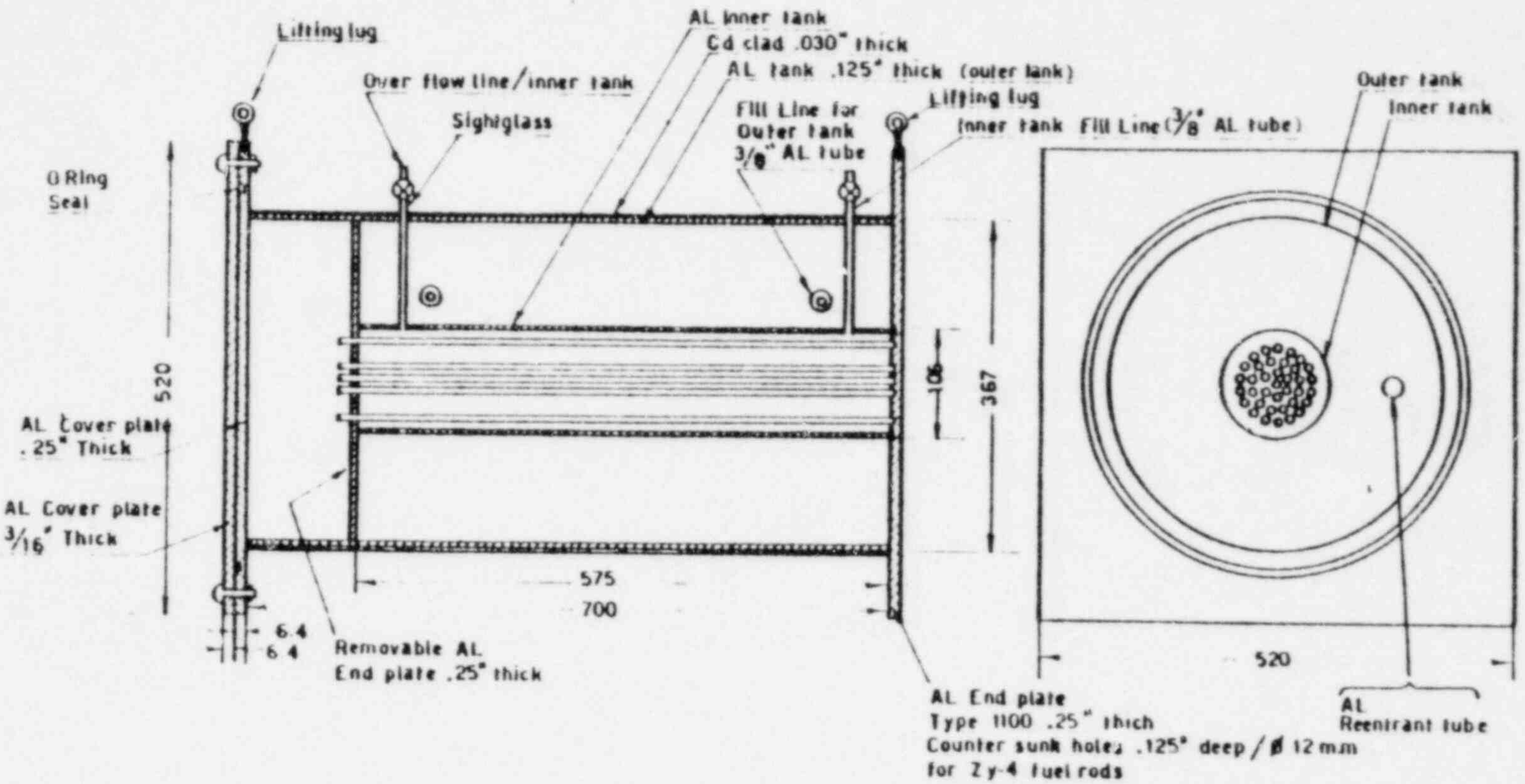
References:

1. Lolich, J. V., and Boado, H. J., "Measurement and Calculation of the Eigenvalues 'α' and 'k' for a Highly Subcritical Assembly" (Submitted to Atomkernenergie)

2. Minutes of the Radiation Safety Committee.

T. F. Parkinson

H. J. Boado



Lifting lug
Over flow line/inner tank
Sightglass
AL Inner tank
Cd clad .030" thick
AL tank .125" thick (outer tank)
Fill Line for Outer tank 3/8" AL tube
Lifting lug
Inner tank Fill Line (3/8" AL tube)
Outer tank
Inner tank
O Ring Seal
520
AL Cover plate .25" Thick
AL Cover plate 3/16" Thick
106
367
575
700
6.4
6.4
Removable AL End plate .25" thick
520
AL End plate
Type 1100 .25" thich
Counter sunk holes .125" deep / Ø 12 mm
for Zy-4 fuel rods
AL
Reentrant tube

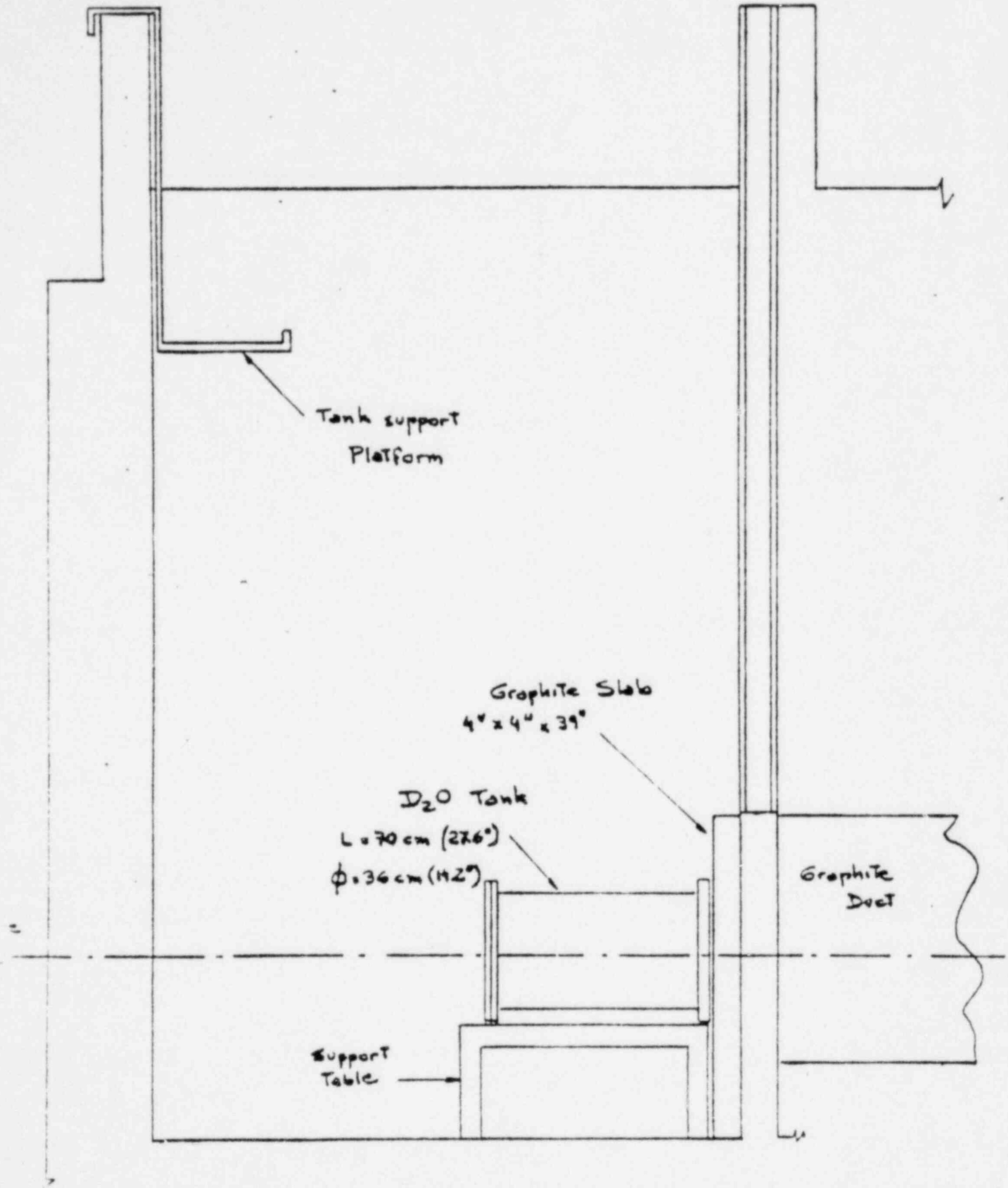


Fig. Shield Tank Elevation
(Cross Section XX)

Scale: 3/4" = 5'0"

POOR ORIGINAL

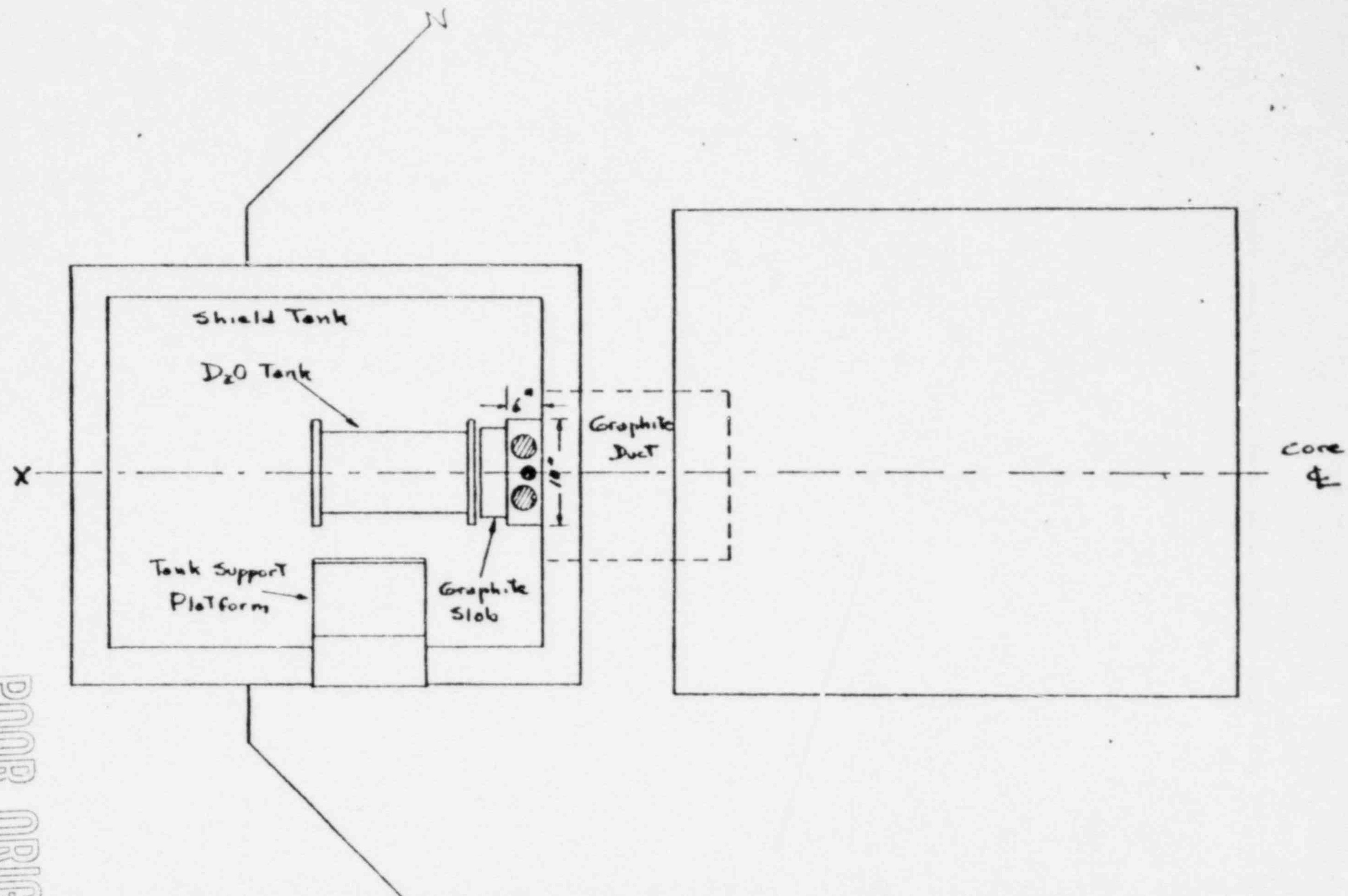


Fig. 2 : Plan view of experiment

Scale: 1/2" = 1'0"

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NUCLEAR ACTIVATION ANALYSIS LABORATORY

July 21, 1980

TO: Reactor Safety Committee

FROM: T. F. Parkinson and H. J. Boado

SUBJECT: New Experiment in the Shield Tank

References: (1). Memo (April 8, 1980) from T. F. Parkinson and H. J. Boado to Reactor Safety Committee
(2). Minutes; Reactor Safety Committee Meeting; May 19 and 23, 1980

I. Introduction

In reference (1) a general description and a safety analysis of the new experiments were presented. The RSC approved the experiments at its May 19 and May 23 meetings subject to the preparation of more detailed procedures for the experiments. A question was also raised about disposal of the radioactive wastes after the experiment is dismantled. The purpose of this memo is to respond to the comments of the RSC.

II. Waste Disposal

The major radioisotope waste produced will be fission products from the natural uranium dioxide fuel pellets. Most of the fission products will be retained within the pellets. Gaseous fission products will be retained within the Zircalloy housing tubes except for the small amounts released when activation foils are removed from the fuel rods after each irradiation. For this operation, a glove box will be used to handle the irradiated fuel tubes and pellets. At the completion of an irradiation the fuel tubes will be loaded into the Al tank and it will be stored in the shield tank.

The duration of these experiments will be at least two years and possibly as long as 5-10 years. The fuel rods were obtained on loan from CNEA (Argentina) and will be retained for an indefinite period. After our experiments are completed the fuel rods can be stored in one of the fuel storage pits. We do not anticipate having to return the fuel rods to CNEA within the next 10 years. By this time the fission product activity will have decayed to very low levels.

Some solid radioactive waste will be induced by activation of the Al double-walled tank. This tank was fabricated from high-purity Type 1100 aluminum to minimize induced activation. At the conclusion of the experiments, the tank will be stored in the shield tank or in a shielded container to allow the activation products to decay. The tank will then be shipped off-site to a low-level waste disposal ground in accordance with NRC-DOT regulations.

The Al tank will be coated with clear Krylon paint to minimize corrosion which could possible cause the release of radioactive corrosion products into the shield tank water.

III. Determination of Neutron Detector Effects and Reactivity Effects of the Al Tank and Fuel Assembly

A. Neutron Detector Effects

Two of the neutron detectors for the reactor instrumentation system are located in vertical tubes placed at the end of the shield tank adjacent to the reactor. These detectors are : (1) a compensated ion chamber (CIC) which supplies the signal for the Intermediate Flux Monitor, and; (2) a compensated ion chamber which supplies the signal for the Keithley picoammeter. Prior experiments have demonstrated that very strong absorbers and voids placed next to these detectors did not affect their readings. To confirm that the double-walled tank has a negligible effect, a set of readings on <u>all</u> neutron detector channels will be taken at several power levels (e.g. 1 watt, 10 watts, 100 watts, 1000 watts) <u>before</u> the Al tank is inserted in the shield tank. The same set of readings will be repeated for the following conditions:

(1) Al tank filled with H_2O but no UO_2.
(2) Al tank filled with H_2O and 7 UO_2 fuel rods.
(3) Al tank filled with H_2O and 19 UO_2 fuel rods.
(4) Al tank filled with H_2O and 37 UO_2 fuel rods.

If the presence of the Al tank shows any significant perturbation on the neutron detectors, then a graphite slab will be introduced between the Al tank and the detectors to insure that there is no effect.

B. Reactivity Effects

During the experiment sequence of Section III A., the control rod settings will also be recorded. From this data the change in k_{eff} will be evaluated. If positive changes are found, k_{eff} will be plotted <u>vs</u>. number of fuel rods so as not to exceed the maximum allowed limit of 0.60% $\Delta k/_k$. Independent measurements have shown that the UO_2 assembly has higher k_{eff} with H_2O moderator than with D_2O moderator.

IV. Review of Experimental Results

A summary report of all the above measurements will be submitted to the Reactor Safety Committee for their review.

T F Parkinson

T. F. Parkinson

H. J. Boado

NEUTRON DETECTOR AND REACTIVITY EFFECTS, Al TANK AND FUEL ASSEMBLY DATA SHEET

	INLET TEMP (°C)	ROD HEIGHT		POWER LEVEL		
		SHIM (%)	REG (%)	S.R. (CPS)	I.R. (nA)	KEITHLEY (AMPS)
REACTOR POWER: 0.1 WATT						
TANK REMOVED	84	44.2	50.0	2×10^5	$<10^3$	3.15×10^{-11}
TANK IN, H_2O ONLY	84	43.6	50.0	2×10^5	$<10^3$	3.05×10^{-11}
TANK IN, H_2O, 7 RODS	84	44.3	50.0	2×10^5	$<10^3$	3.0×10^{-11}
TANK IN, H_2O, 19 RODS	84	44.5	50.0	2×10^5	$<10^3$	3.05×10^{-11}
TANK IN, H_2O, 37 RODS	84	44.0	50.0	2×10^5	$<10^3$	3.15×10^{-11}
REACTOR POWER: 1.0 WATT						
TANK REMOVED	84	44.2	50.0	7×10^5	1.7×10^3	7.5×10^{-11}
TANK IN, H_2O ONLY	84	43.5	50.0	7×10^5	1.8×10^3	7.5×10^{-11}
TANK IN, H_2O, 7 RODS	84	44.5	50.0	7×10^5	1.8×10^3	7.75×10^{-11}
TANK IN, H_2O, 19 RODS	84	43.8	50.0	7×10^5	1.7×10^3	7.75×10^{-11}
TANK IN, H_2O, 37 RODS	84	44.3	50.0	7×10^5	1.9×10^3	7.75×10^{-11}
REACTOR POWER: 10.0 WATT						
TANK REMOVED	84	44.2	50.0	$>10^6$	6×10^3	2.34×10^{-10}
TANK IN, H_2O ONLY	84	44.5	50.0	$>10^6$	6×10^3	2.3×10^{-10}
TANK IN, H_2O, 7 RODS	84	45.0	50.0	$>10^6$	6×10^3	2.3×10^{-10}
TANK IN, H_2O, 19 RODS	84	44.8	50.0	$>10^6$	6×10^3	2.3×10^{-10}
TANK IN, H_2O, 37 RODS	84	44.0	50.0	$>10^6$	6×10^3	2.35×10^{-10}
REACTOR POWER: 100.0 WATT						
TANK REMOVED	84	44.2	50.0	$>10^6$	8×10^4	2.55×10^{-9}
TANK IN, H_2O ONLY	84	44.8	50.0	$>10^6$	5×10^4	2.7×10^{-9}
TANK IN, H_2O, 7 RODS	84	44.6	50.0	$>10^6$	5×10^4	2.7×10^{-9}
TANK IN, H_2O, 19 RODS	84	44.4	50.0	$>10^6$	5×10^4	2.7×10^{-9}
TANK IN, H_2O, 37 RODS	84	44.4	50.0	$>10^6$	5×10^4	2.55×10^{-9}

MEASUREMENTS MADE 9/3/80 THROUGH 9/12/80
REACTOR RUN #'S 4283 THROUGH 4285