YAEC-87

REACTORS—POWER

UNITED STATES ATOMIC ENERGY COMMISSION

QUARTERLY PROGRESS REPORT FOR THE PERIOD APRIL 1, 1958 TO JUNE 30, 1958

By
I. H. Coen
R. W. Garbe

July 30, 1958

Atomic Power Department
Westinghouse Electric Corporation
Pittsburgh, Pennsylvania

Technical Information Service Extension, Oak Ridge, Tenn.

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YAEC-87

Yankee Atomic Electric Company

Research And Development Program

QUARTERLY PROGRESS REPORT

for the period

April 1, 1958 to June 30, 1958

by

I. H. Coen
R. W. Garbe

Large Plant Engineering

For The Yankee Atomic Electric Company
Under Research and Development Subcontract
No. 1 of USAEC-YAEC Contract AT(30-3)-222

July 30, 1958

APPROVED: A. E. Voysey

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Project Manager

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ABSTRACT

This report contains a technical description of the research and development work accomplished and the progress made during the period from April 1, 1958 to June 30, 1958, under the Research and Development Program as presented in Report YAEC-41, Rev. 2. An evaluation and the resulting conclusions of work performed are presented for each project in which definitive progress has been made.

The principle progress during the second quarter of 1958 consisted of:

1. Accelerating the planning, scheduling and coordination of the process-water and in-pile test loop irradiation programs.

2. Performing two-enrichment non-cycled core studies to determine the improvement in the radial hot channel factor over that which would apply for a uniform core.

3. Recalculating the contribution of control to the temperature coefficient using better thermal constant values and an improved method of obtaining control rod worths.

4. Performing a burnup analysis of the reference control rod program to determine the effect of the program on the nuclear hot channel factor.

5. Comparing the decontaminating abilities of two promising ion-exchange resins using a synthetic corrosion product mixture.

6. Establishing a basic permanganate-citrate decontamination procedure which will provide a safe and efficient method of descaling the stainless steel primary coolant piping.

7. Investigating alternate fuel assembly designs to eliminate the possibility of excessive thermal bowing of the assembly which could result in impeding the movement of an adjacent control rod.

8. Determining flow and power output at any instant during a two-pump loss-of-coolant-flow accident.

9. Calculating the effect of local boiling on pressure drop and coolant flow distribution and the maximum possible volume of local boiling.

10. Performing visual flow studies and pressure drop measurements on a one-twelfth scale plastic reactor vessel model.

11. Investigating methods of improving the creep strength of silver-indium-cadmium control rod alloy by increasing grain size.

12. Initiating the preparation of a digital computer code to represent the transient behavior of the steam pressurizer.

13. Experimentally determining reactor parameters by measuring the reactor response to a variable by reactor period and control rod position change and by obtaining flux measurements using a variety of foil and wire detectors.

14. Modifying the in-pile test loop based on information obtained from visits to the loop fabricator and the Materials Testing Reactor.

15. Disassembling, identifying and examinimg the second group of process-water specimens removed from the MTR. Performing post-irradiation measurements, collecting fission gases, and metallographically examining the ferrule brazes on two of the four specimens in the second group.

Section-A of the Appendix contains the latest revised mechanical, thermal and nuclear design data for the first core. The studies and calculations performed under Projects 2.0, 4.0, and 5.0 during the second quarter of 1958 were responsible for the changes made in these data from the data presented in Quarterly Progress Report YAEC-65, covering the first quarter of 1958.

Section-B, "Information Availability", reviews the accessibility to the Westinghouse Atomic Power Department and the Yankee Atomic Electric Company of documents and reports pertaining to nuclear power reactors. The quantities and types of reports and microfilms received by the Yankee Atomic Electric Company and Westinghouse Atomic Power Department during the second quarter of 1958 are also given in this section.

Section-C contains abstracts of trip reports written as a result of visits made by Westinghouse APD and YAEC personnel during the second quarter of 1958 to nuclear testing facilities, vendors locations, and installations of AEC contractors.

INTRODUCTION

The Yankee Atomic Electric Company has contracted with the Westinghouse Atomic Power Department under Yankee Contract No. 1 of Contract AT(30-3)-222 between the U. S. Atomic Energy Commission and the Yankee Atomic Electric Company to perform the research and development work required to build a 134 MW (net electrical output) pressurized light water nuclear reactor power plant using slightly enriched uranium dioxide (UO_2) fuel pellets contained in stainless steel tubes.

YAEC Development Program Report, YAEC-41, Rev. 2, outlines the Research and Development Program for the period from January 1 to June 30, 1958. Quarterly Progress Reports, YAEC-7, YAEC-13 (Revision 1), YAEC-20, YAEC-35, YAEC-44, YAEC-52, and YAEC-65 describe and evaluate the work accomplished from the beginning of the program, June 6, 1956 to March 31, 1958.

The basic objectives of the YAEC Research and Development Program are:

1. To carry out the development of stainless steel UO_2 fuel rods to a point where successful operation of experimental elements in a test reactor has been demonstrated.

2. To improve the process of manufacturing sintered UO_2 fuel pellets for the rod elements and, to reduce fuel preparation costs.

3. To select a proven material or to demonstrate the feasibility of a material suitable for use in control rods for a long-lifetime pressurized water reactor core.

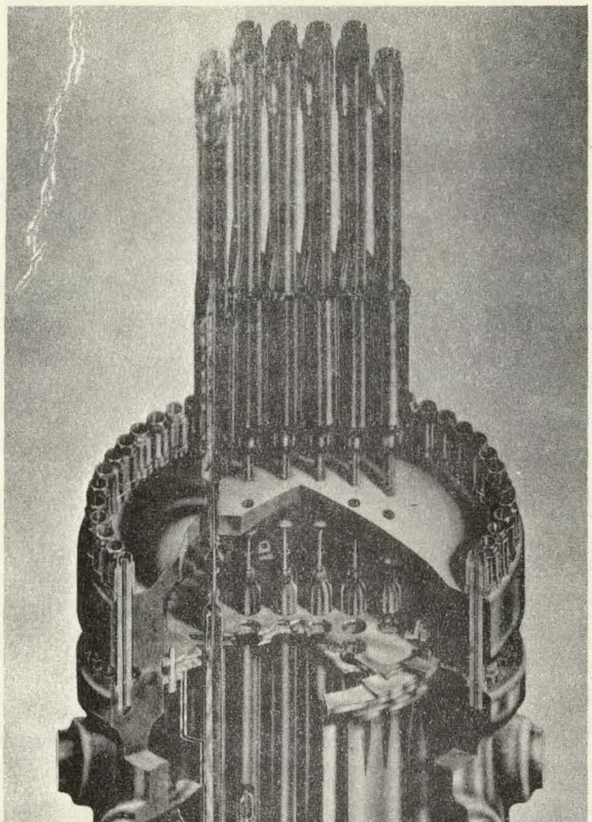
4. To obtain through analysis and experiments a thorough understanding of the nuclear physics aspects of the proposed stainless steel UO_2 pressurized water reactor.

5. To obtain an optimized preliminary design for a nuclear reactor and associated auxiliaries for a turbine electric power plant based upon the use of a stainless steel clad long-life core.

6. To determine the proper materials of construction, the required efficiency of the water purification system, the selection and use of a soluble neutron absorber and the minimizing of crud deposition through water chemistry analyses.

"Information Availability Report", YAEC-30 for the period from June 6, 1956 to September 30, 1957, and Quarterly Progress Reports, YAEC-44, YAEC-52, and YAEC-65 for the third and fourth quarters of 1957 and the first quarter of 1958, covered the availability of information to, and the difficulties encountered by the Westinghouse APD in their efforts to obtain documents and reports to further the success of the YAEC Research and Development Program. The four reports also included a section containing abstracts of trip reports made under the YAEC Research and Development Program.

A sectionalized illustration of a large Yankee-type pressurized water reactor is shown on the following page.



Large Pressurized Water Reactor

1.0 FUEL ELEMENT DEVELOPMENT

Chemistry and Ceramics Section: R. F. Sterling, Manager
Metallurgical Section: R. K. McGeary, Manager

The work under this project is directed toward the development of a satisfactory stainless steel clad UO_2 fuel element.

1.1 Uranium Dioxide Fuel Material Preparation

R. Winchell

Uranium Dioxide Press Feed Preparation Methods

A series of laboratory-scale trials to reduce the magnitude and complexity of preparation of ceramic fuels prior to pressing which were previously reported in YAEC-65, Quarterly Progress Report for the January 1 to March 31, 1958 period, were extended to a series of runs on production equipment in an effort to produce satisfactory uranium oxide fuel pellets using the simplified procedures. The operating parameters and results of a number of the trials using "as received" UO_2 powder are as follows:

Trial 1 - Press UO_2.

Forming Pressure, TSI	Pelletization Attempts	Result
Under 5	20	Green density 6.09 g/cc
Under 8	--	No whole pellets were produced, all fell apart from laminae.

Trial 2 - Press UO_2 after adding only 0.2% Stero-tex Z.

Forming Pressure, TSI	Pelletization Attempts	Green Density, g/cc
8	10	6.23
17.5	10	6.35
25	6	No whole pellets were produced, all fell apart from laminae.

Trial 3 - Press UO_2 after adding only 0.4% Stero-tex Z.

Forming Pressure, TSI	Pelletization Attempts	Green Density, g/cc
17	12	6.27
25	10	6.29
35	12	6.43
43	10	6.40
48	10	6.38

Trial 4 - Press UO_2 after adding only 1% Stero-tex.

Forming Pressure, TSI	Pelletization Attempts	Green Density, g/cc
8	20	6.11
17	20	6.25
26	20	6.51
35	20	7.19

Trial 5 - Wet ball mill UO_2, granulate when medium wet, press without drying.

No pellets were produced during ten attempts - all fell apart upon ejection from the die due to laminae.

Trial 6 - Wet ball mill UO_2, partially dry, add 0.5% Stero-tex, press.

Powder would not feed automatically to the press die during six attempts until the powder was thoroughly dried. Results were as follows:

Forming Pressure, TSI	Pelletization Attempts	Green Density, g/cc
44	30	7.06
52.5	30	7.29

Trial 7 - Wet ball mil UO_2 with 1% Polyvinylalcohol (PVA), dry, granulate, add 0.2% Stero-tex Z, and press.

No pellets survived ejection from the die during ten attempts due to laminae.

Trial 8 - Wet ball mill UO_2 with 1% PVA plus 2% Carbowax 20M, dry, granulate and press.

Control Ball mill UO_2 two hours, 50% water; dry, 80° ± 5°C.; granulate, 20 mesh; add PVA, 1%, 100 mesh; add water, 9%; partially dry; granulate, 8 mesh; thoroughly dry; granulate, 20 mesh; add 0.2% Stero-tex Z; press; presinter; sinter; grind.

		Densities, g/cc					
		Trial			Control		
Forming Pressure, TSI	Pelletization Attempts	Green	Fired, 3 hrs. 1500 °C	Fired, 3 hrs. 1600 °C	Green	Fired, 3 hrs. 1500 °C	Fired, 3 hrs. 1600 °C
17	20	6.92	9.02	9.71	7.58	9.01	9.80
35	20	7.25	9.27	9.90	7.58	9.77	9.77
61	20	7.74	9.49	10.09	7.82	8.95	9.92

The trial procedure produced good pellets from three lots of UO_2. A fourth lot remains to be tested.

Trial 9 - Wet ball mill UO_2 with 2% Carbowax, dry, press.

No satisfactory pellets were produced during twelve attempts using production pressing equipment.

Trial 10- Wet ball mill with 1% Carboxymethylcellulose (CMC), dry, granulate, add 0.2% Stero-tex, press.

Pressure, TSI	Pelletization Attempts	Densities, g/cc	
		Green	Sintered at 1600°C.
1	--	--	--
8	--	--	--
17	10	--	8.86
61	10	7.1	9.62
75	10	--	9.79
88	10	--	9.89

A topical report entitled, "Progress Toward Development of UO_2 Pellets for Yankee Core-I" is being prepared to cover the development of uranium oxide powder specifications; the present pelletization procedure and the resultant pellet specifications.

Dimensional Control of Uranium Oxide Pellets

A relationship regarding dimensional control of uranium oxide pellets was developed which closely approximates the observed behavior of fabricated uranium oxide pellets. However, use of the relationship requires the following two reservations: shrinkage is not always exactly isotropic, and the green density must be adjusted for the loss of weight of volatiles.

$$\frac{d_g}{d_f} = \left(\frac{D_f}{D_g}\right)^{1/3}$$

where d_g = green dimension

d_f = fired dimension

D_f = fired density

D_g = green density

The relationship may be expressed: % linear shrinkage, on the fired basis is equal to:

$$\left[\left(\frac{D_f}{D_g}\right)^{1/3} - 1\right] 100$$

The relationship indicates that control of green density and fired density enables a control to be made of the fired dimension. Control of green density has not proven difficult to obtain while fired density control is difficult to obtain apparently because of the variability of ranium oxide as received.

1.3.1 End Closure of Fuel Rods

G. G. Lessmann

Although the end plug welding device used to seal the fuel tubes for the Yankee CRX produced closures which are in excess of 99% acceptable, it was felt that further improvements in the integrities of the weld could be made only by major revisions in the equipment. Consequently, an electronically controlled AIRCO welding head has been purchased and installed. A Plexiglass hood has been placed over the arc and tube end. A pyramid shaped hood proved to be superior to a cylindrical hood with regard to distribution of the helium from the welding nozzle. The power lines for the D.C. current and high frequency starting current were rewired and the tube end support reconstructed to minimize tube wobble.

The AIRCO automatic welding head is especially designed to maintain a constant arc gap even though the tube wobbles. Although welds obtained with the improved equipment appear to be decidedly superior to those obtained with the old equipment, data from a large quantity of welded tubing are being compiled for evaluation. A topical report, "Heliarc End-Closure of Stainless Steel Fuel Element Tubing" is being prepared covering the investigations performed and the results obtained under this subproject.

1.3.2 Joining Fuel Bundles into Assemblies

P. P. King J. R. Dazen M. D'Amore

Brazed Joint Studies

A major effort during the period has been on improving the joint strength and ductility and determining the factors which will produce the desired result. Microstructures of the strongest and most ductile joints always show diffusion across the tube interface which produces a single phase alloy joint. Fit-up, brazing temperature and time, hydrogen impurity, and other factors appear to govern the obtaining of diffusion bonded joints. Of the various possible braze application methods and materials, there remains three methods which are now being investigated intensively to determine which is most satisfactory for a production brazing process. These are:

1. Plating the ferrules with 0.0005" to 0.001" of Ni-P brazing alloy and using a jig during brazing of the bundle.

2. Plating in the same manner as in (1) using a jig and then applying a small quantity of Nicrobraz 50 brazing slurry to each joint.

3. Using a jig and applying Nicrobraz 50 brazing slurry to each joint.

Each method appears to have certain advantages and disadvantages. With (1), the fit-up during jigging is important so that unbrazed joints are avoided. With (2), a double operation is necessary although this is not considered excessively expensive. With (3), experience indicates that the operator may omit applying the brazing alloy to every joint. The conditions for braze flow are not as satisfactory as is the case in (1) and (2). Figure-1 shows the tensile load required to fracture a single joint for the three methods discussed when subjected to various heat treatments. It may be noted that these data indicated no marked advantage of one method over another with regard to breaking load.

An additional 7-1/2 foot long, 36-tube, Yankee fuel sub-assembly was brazed at the Ferrotherm Company during the second quarter of 1958. There has been a steady improvement in the dimensional tolerances obtained due to the experience gained in proper jigging of the four half-length Yankee and four Belgium Thermal Reactor prototype assemblies which were brazed ~ one hour at 1850°F. at the Wall - Colmonoy Company. It was determined that additional shims are required between horizontally and vertically spaced tubes to meet the tube spacing requirements and that additional metal welded to the U-shaped clamping fixture produces a more uniform cooling of the bundling after brazing. However, in no case have the brazing conditions been optimum for obtaining a high strength bundle with a majority of diffusion bonded joints. Brazing work with vendors has now been suspended since the vertical pit brazing furnace at Westinghouse APD has been installed and is operating in a very satisfactory manner. At 1850°F., the pit furnace has a uniform hot zone of 108" and can be maintained within ± 20°F. The first 36-tube test bundle for the Belgium Thermal Reactor, 52 inches long, was brazed in this furnace late in June. The top three spacer rows were coated with 0.010" of Ni-P on the ferrules and the bottom three rows were held in a jig while Nicrobraz 50 brazing slurry was added to the ferrules. After obtaining dimensional data, the bundle was destructively tested for fracture loads for each of the joints. Results obtained were as follows:

1. Straightness: maximum bow approximately 0.015".

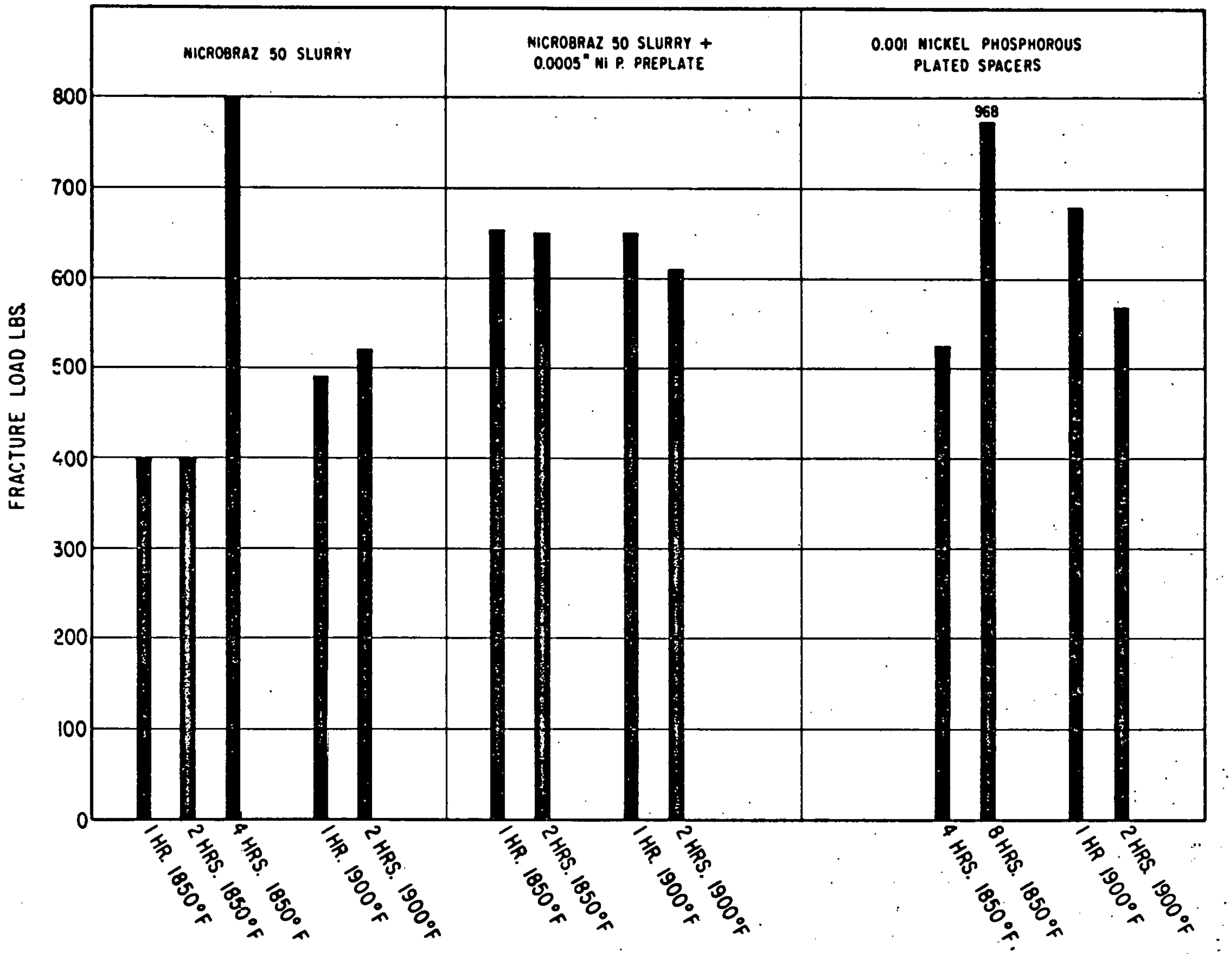
2. Spread on design width of 2.538": maximum 2.546", minimum 2.528".

3. Inter-tube spacing: All within ± 0.015" of design.

4. Braze efficiency: Five Ni-P plated joints were unbrazed; all Ni-P plus Nicrobraz 50 joints were brazed.

5. Joint strength: Average joint strength was equal to 320 lbs. after brazing the bundle at 1850°F. for one hour.

The next bundle will be brazed at 1900°F. for two hours. This should improve the joint strength but may result in poor dimensional control. Several additional bundles will be brazed in the coming period in which temperature, time, and braze application techniques will be altered in order to ascertain optimum brazing conditions.



SINGLE JOINT FRACTURE LOAD
FOR VARIOUS BRAZING MATERIALS

FIGURE 1

One of the 50-inch long, 36-tube, sub-assemblies brazed at the Wall-Colmonoy Corporation was thermal cycled 28 times from room temperature to 600°F. in an autoclave containing neutral degassed water. Each cycle had a duration of 24 hours. After thermal cycling, the sub-assembly was visibly unchanged and dimensional analysis indicated a maximum change in any cross sectional measurement to be + 0.003 inches, which is within the accuracy of the measurement. All of the joints visible from the outside of the bundle were visually inspected and no pitting or cracking was observed under a stereoscopic microscope. An inspection of 184 joints revealed only one joint which was not brazed. Each of the seven ferrule modes was then sectioned and checked for individual tube-to-tube spacing. Of the 340 measurements taken, 92.7% were within $\pm$ 0.010 inches of the design spacing. The minimum spacing was 0.087 inches and the maximum spacing was 0.118 inches. The next step was to destructively test all of the fuel tube joints in each cross section held by one brazed joint to determine joint fracture load. Of the 146 such joints tested, 6 fractured during handling and 19 had fracture loads below 100 pounds. The remaining 110 joints had fracture loads above 200 pounds. The average joint fracture load for the sub-assembly was 340 pounds. These results indicate that dimensional tolerances are being controlled closely, but that joint strengths are erratic because diffusion bonds are not being consistently obtained. Joint strength can be improved by brazing at increased temperature for a longer period of time. An important conclusion is that, although the bundle described had not received optimum brazing treatment, thermal cycling apparently produced no significant change in the dimensions and did not cause cracking of the brazed joints.

Stretch-Forming

A hand-operated tube-stretching machine is now in operation after being completely rebuilt and modified as follows:

a. New gripping heads were added.

b. A new ram system was developed.

c. The length of the bed has been increased to allow stretching of full-size Yankee fuel rods. Previously, tube lengths were limited to 60 inches.

d. The frame has been altered so that now it is a permanent, horizontal position tube stretching machine.

An experiment to determine the stretching characteristics of AISI 304 weldrawn stainless steel tubing has been completed. The rate of decrease of the I.D., O.D., and wall thickness of the tubing were determined with various elongations up to 5%. Ovality change and spring-back of the length on release of tension were also recorded. Data obtained from the study are summarized in Table I.

TABLE I

Stretching Characteristics of AISI 304 Stainless Steel Weldrawn Tubing

Nominal Tube Size: 0.3055" I.D. x 0.021" Wall Thickness

1. I.D. Reduction Rate = 1.4 mils decrease in diameter for each 1% elongation of tube.

2. O.D. Reduction Rate = 1.64 mils decrease in diameter for each 1% elongation tube.

3. Wall Thickness Reduction Rate = 0.11 mils decrease in thickness for each 1% elongation of tube.

4. Change in Ovality (Maximum) = I.D. = 0.3 mil increase
O.D. = 0.43 mil increase

5. Springback (Length) of Tube After Stretch

Elongation (10" Gage Length)		Springback	
%	inches	inches	% of Length Stretched
0.5	0.050	0.011	22
1.0	0.100	0.012	12
1.5	0.150	0.015	10
2.0	0.200	0.013	7
3.0	0.300	0.020	7
4.0	0.400	0.020	5
5.0	0.500	0.020	4

The results of the preceding study were obtained with tubes containing aluminum-oxide ceramic pellets. Present work has been extended to include depleted uranium dioxide pellets and compartmentation discs brazed in place in the tubes.

Spacer Disc Brazing Studies

Techniques for satisfactorily brazing discs in fuel tubes have been established. The parameters that were varied in obtaining optimum results were:

1. The thickness of the nickel-phosphorus brazing alloy (Ni - 90 wt. %, P - 8 to 9 wt. %), which was varied between one-quarter and one mil. The discs were ground to a 0.302" diameter from a plated diameter of 0.304".

2. Heat treatment to improve the ductility of the brazing alloy.

3. Surface grinding of the disc flat surfaces to eliminate the braze material that would be in contact with the UO_2 pellets.

Satisfactory results have been obtained by the use of a stainless steel disc having a one-half mil deposit of brazing alloy and subjecting it to a four hour heat treatment at 1400°F. It was determined that "starved" joints were produced when the flat surfaces of the discs were ground; apparently, during brazing, molten braze material flows to the fillet from the flat surfaces upon which it has been deposited. The brazing cycle for the disc-compartmented tube consisted of a one hour heat treatment in the hot zone of the brazing furnace at a temperature of 1875°F.

Reaction Between Spacer Discs and Fuel Pellets

The possibility of a reaction occurring during the brazing of compartmentation discs was investigated. Consideration was given to the possible reaction among UO_2, nickel-phosphorus brazing alloy and AISI 304 stainless steel cladding material during a 1900°F. brazing treatment. Microscopic examination of samples subjected to a brazing treatment of 45 minutes at 1900°F. indicated only a mechanically-bonded interface with no indication of a chemical reaction. Metallographic examination after a scaled-down corrosion test of four days at 680°F. in 40 ppm boronated water indicated no excessive corrosion or reaction at the bond interface.

1.4 Fabrication and Analysis of Fuel Elements for Critical Assembly and Irradiation Tests

J. J. Lombardo

During the second quarter of 1958, considerable work was performed on planning, scheduling, and coordination of the irradiation programs. An irradiation coordinator has been appointed to exercise overall responsibility for both the process water irradiation program and the in-pile loop irradiation program. Definite completion dates have been scheduled. A description of the complete Yankee irradiation program (with exception of WCAP-1 and WCAP-2 experiments) is outlined in Table II. The fabrication of samples for the in-pile loop experiment is proceeding; priority will be given to expediting the delivery of mock fuel specimens to the Lummus Company, (loop constructors) by August 1, 1958. Work is also proceeding on the fabrication of fuel specimens for the process water irradiation tests.

TABLE II

YANKEE R & D PROGRAM - PROJECT 11.2 - MTR PROCESS WATER AND IN-PILE LOOP IRRADIATION EXPERIMENTS

Tube	Annular Clearance, in. (Pellet to Tube)	Axial Clearance	Configuration	Clad Thickness	Bundle Description	No. of Capsules
Additional 6-1/2" Process Water Irradiation Capsules (five bundles)						
1	0.0000	0.055 in/compartment	Discs in center and ends	21 mils	Tubes	
2	0.0005 - 0.0015	0.055 in/compartment	Discs in center and ends	21 mils	brazed	3
3	0.0025 - 0.0030	0.055 in/compartment	Discs in center and ends	21 mils	together	
4	0.0005 - 0.0015	0.000	Disc at one end	21 mils	Tubes free to	
5	Banding occurs	0.012 in/pellet	No discs	15 mils	move	2
6	0.0005 - 0.0015	0.084 in/compartment	Discs at both ends	21 mils	independently	
30" Long Process Water Irradiation Experiment (One bundle)						
1	0.0005 - 0.0015	0.160 in/compartment	Compartmented	21 mils	Tubes brazed	
1	0.0005 - 0.0015	0.160 in/compartment	Compartmented	21 mils	together	1
2	Banding occurs	0.012 in/pellet	No compartment	15 mils	Free tube	
In-Pile Loop Experiment (Three bundles)						
Sample A - Three tubes (30" long)	0.0005 - 0.0015	0.160 in/compartment	Compartmented	21 mils	Tubes brazed together	1
Sample B - (8" long)	Control Rod Samples					1
Sample C - (24" long)	Banding occurs	0.012 in/pellet	No compartments	15 mils	Tubes brazed together	1

2.0 NUCLEAR DESIGN AND REACTOR PHYSICS

Nuclear Design Section: H. W. Graves, Jr., Manager
Computer Analysis Group: G. H. Minton, Supervisor
Reactor Physics Group: W. H. Arnold Jr., Supervisor

Included in this project are studies and calculation on the core deisgn optimization, kinetic and steady state reactor analysis, planning of criticality and irradiation experiments, shielding, and the reactor startup and operation.

2.1 Core Design Optimization

No work was performed under this subproject during the second quarter of 1958.

2.2 Core Steady State Analysis

R. E. Wolf G. G. Tirellis R. J. French

Multi-Region Loaded Core Studies

A study of multi-region loading in the Yankee first core, although not yet completed, has given some preliminary results. It appears possible to improve the radial hot channel factor by 30% over a uniform core by going to a two-enrichment non-cycled core. This would probably allow 145 net MwE to be drawn from a 7.5 feet high core. Such a core requires a slightly higher U-235 inventory than either the uniform or an outside-in cycled core of the same power output, but the latter type does not show a comparable improvement in hot channel factors without going to very large burnups.

The first phase of the study was to compare the average burnup and lifetime average radial heat flux for the uniform core, parameters for which appear in the Nuclear Design Data section of Appendix A, with those for two-region outside-in cycled cores of various burnups. The cycle times (i.e., the time between fuel shifting operations or one-half the total residence time) considered were 5,000; 7,000; and 10,000 hours. Table III demonstrates the fact that none of these cores shows a significant improvement in burnup and heat flux over those for the uniform core.

TABLE III

Comparison-Average Burnup and Lifetime Average Radial Heat Flux of Uniform and Two-Region Outside-In Cycled Cores

Core Lifetime in Hours		Average Burnup MWD/T at 392 Mw	Lifetime Average Radial Heat Flux (F_Q) (Including 1.2 Peaking Factor)
10,666 uniform		8,208	2.07
Outer Region	Inner Region		
5,000	5,000	8,150	2.12
7,000	7,000	11,400	2.05
10,000	10,000	16,300	1.93

The reason for this failure of the cycled cores to improve F_Q is due to insufficient burnup of the fuel in the outer core region before it is moved to the inner region. Power flattening is not effective unless the central region fuel is appreciably less reactive than that in the outer region.

On the other hand, cores which are initially loaded in two enrichments can begin life with radial factors as low as 1.7 (including the 1.2 peaking factor). Furthermore, information obtained from the Belgium Thermal Reactor two-region studies indicates that this factor improves or, at the very least, stays constant with life. Another advantage of the two-enrichment core over a cycled core is that, for a given average MWD/T burnup, the pressure vessel need be opened only half as often. Other advantages are that the two regions need not be of equal area and that a detailed record of the burnup of each fuel element, used to determine the configuration after cycling, need not be kept. It must be remembered, however, that the two-enrichment core does require a higher U-235 inventory and a greater control rod worth. The higher loading might prove desirable if fuel burnups could be attained which would allow a cycled core to have as low an F_Q as a two-enrichment core, and if the fuel inventory charge were above 4%.

As a first step in finding the optimum two-enrichment Yankee core, three different two-region configurations were chosen, as shown in Figure 2. Each of these three configurations was then studied on the WANDA one dimensional flux distribution code with four combinations of inner and outer enrichments in the ratios (outer/inner) of 1.25, 1.50, 1.75, and 2.00. Figure 2 shows the results of this study. The three vertical lines are the configurations A, B, and C of Figure 2. The ordinate is the ratio of the outer region enrichment to the inner, so that each of the points is one of the twelve cores studied. A thirteenth, the uniform core, lies on the abscissa. The pair of numbers to the left of each point are the two-region enrichments, while the number on the right is the radial F_Q

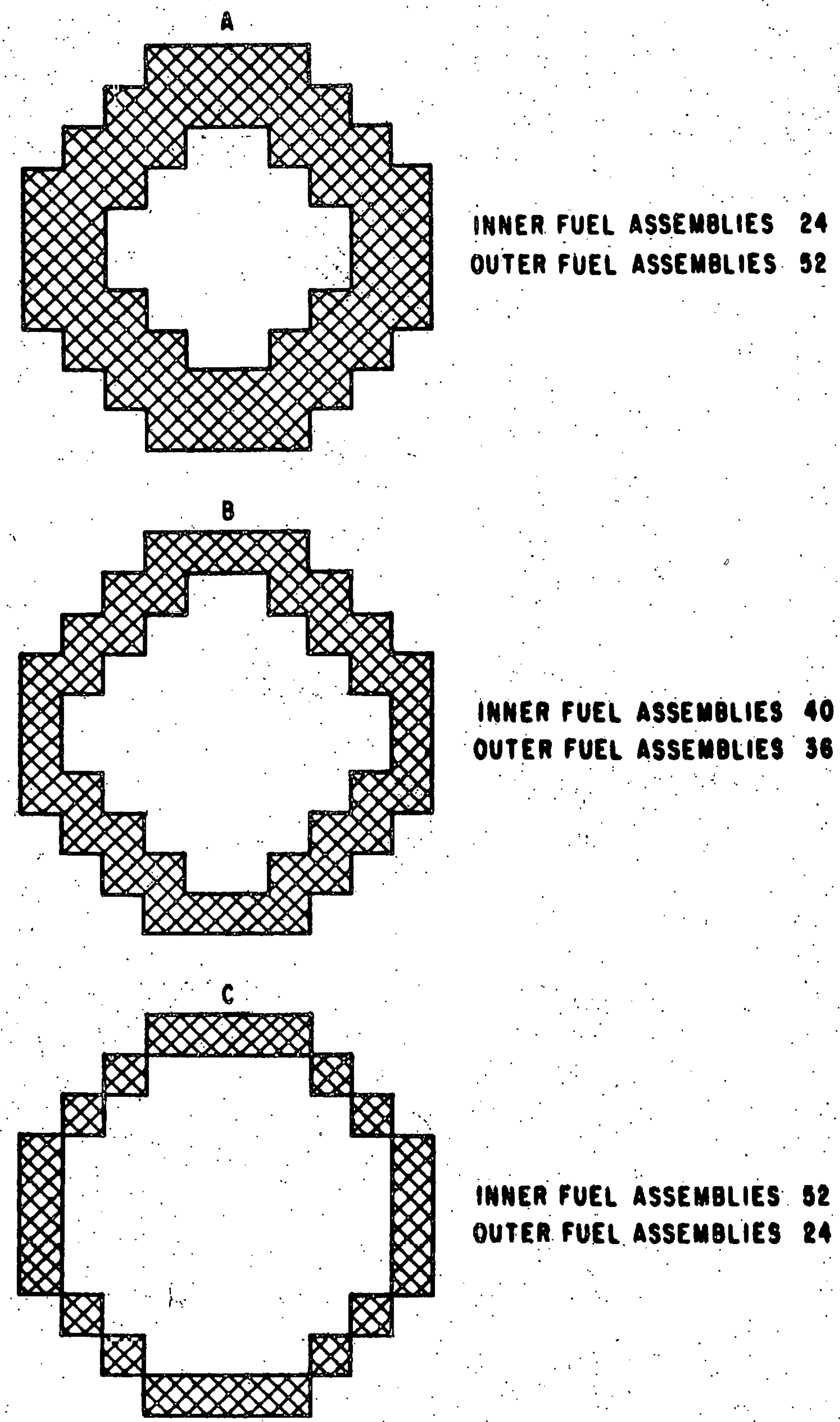


Figure 2 **Two Region Core Configurations for Two Enrichment Yankee Cores**

(without peaking factor). It can be seen that this number is as low as 1.28 for one of the cores. The three cores marked with an asterisk appeared to warrant further and more detailed study. In general, cores to the left of the diagram are more desirable because the outer enrichment required is less.

The next step in the study was to subject the three cores starred in Figure 3 to a two-dimensional power distribution analysis using the geometries shown in Figure 2. The two-dimensional study was performed using the PDQ multi-group multi-region x - y diffusion theory code. The results of this study, reproduced in Table IV for three of the four cases, show that an initial F_Q radial can be obtained which is 30% better than that of the uniform core in the design data sheets. Control rod programming for a two-region core must be considered, however, before a definite quantitative comparison can be made.

Presently, the CANDLE pointwise burnout code is being applied to the third core in Table IV to determine the level of region enrichments required for various desired lifetimes. Upon completion of this study, a fuel cost analysis can be made of the two-enrichment core.

TABLE IV

PDQ Results for Two Enrichment Cores

U-235 Enrichment (%)			
Outer Region	4.57	4.00	3.46
Inner Region	2.61	2.67	2.77
Inner/Outer Sub-assemblies	52/24	40/36	24/52
F_Q Radial (with 1.2 peaking factor)	1.70	1.74	1.71

A relationship for the determination of the resonance escape probability in heterogeneous systems has been found which appears to fit the Bettis TRX data more precisely than the Wigner equation which is in use at the present time. The values of the effective resonance integral, inferred from the Bettis data and the standard escape equation

$$P^{28} = \exp\left[-\frac{N_{28}RI}{\epsilon\Sigma_s}\right]$$

in which P^{28} is the resonance escape probability, N_{28} is the number density of U_{238} in the fuel region, RI is the resonance integral and $\epsilon\Sigma_s$ is the homogenized slowing down power; do not appear to have the simple correlation

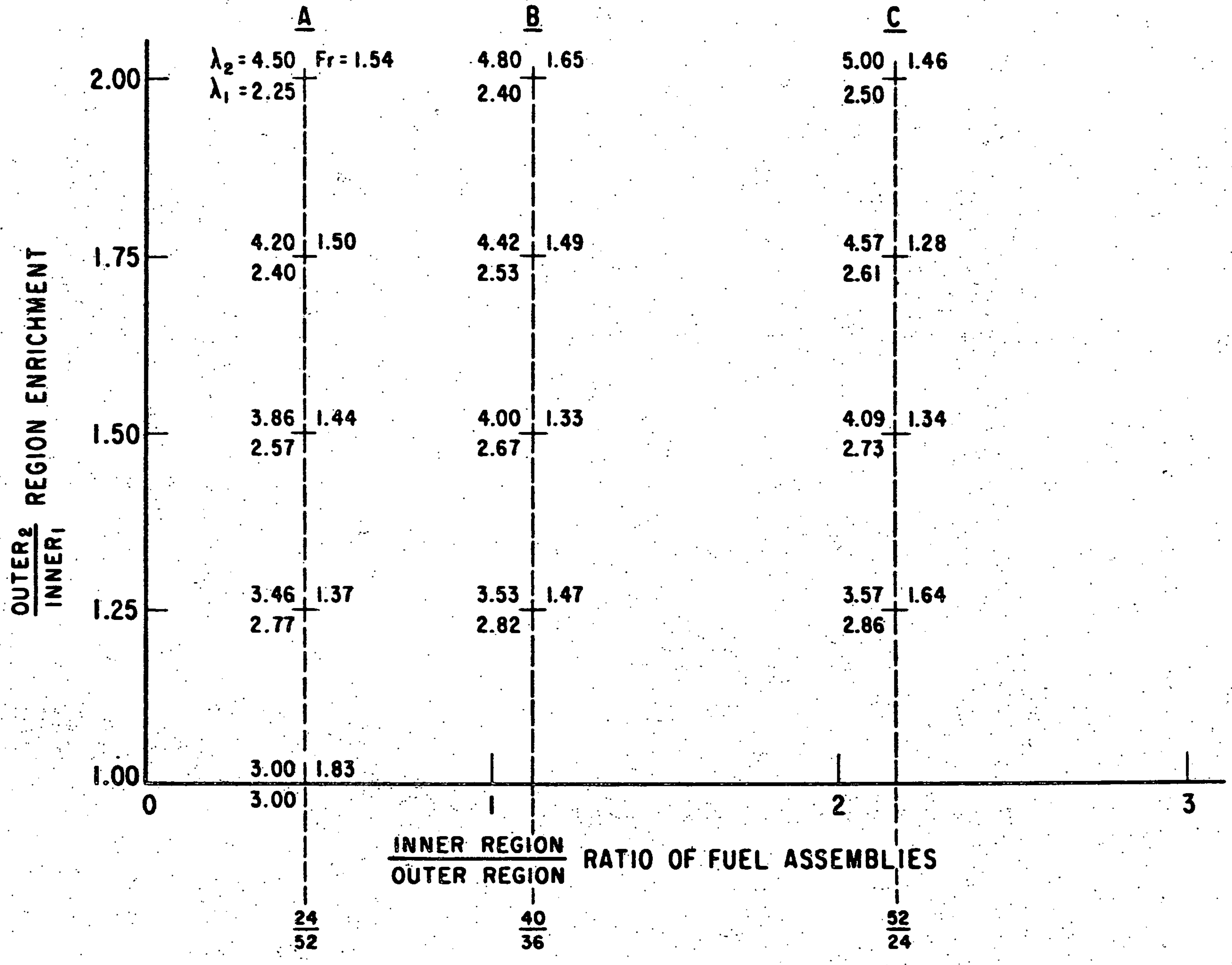


Figure 3 **Enrichments and Hot Channel Factors of Two Enrichment Yankee Cores**

with surface-to-mass ratio (corrected for shielding by neighboring elements) proposed by Wigner. This is in opposition to the fact that directly measured resonance integrals do fit the Wigner relationship. On the basis of this information it has been decided to alter the method used to calculate resonance escape probability.

Based on the assumption that the slowing down density in a material is proportional to the slowing down power in the material, all neutrons slowing down in the fuel regions will be absorbed, and the absorption of neutrons slowing down in the moderator will be proportional to the interface surface area between the fuel and the moderator; it was found that

$$(1 - P^{28}) = \mu \frac{(S/V)}{\xi \Sigma_s} (1 - C)$$

where (S/V) is the surface to volume ratio, (1 - C) is the standard Dancoff shielding correction and μ is a constant of proportionality for a given fuel material and fuel element geometry. A preliminary analysis of Bettis data has indicated that the value of μ may be represented by

$$\mu = 0.573 + 0.0987 \ln (N^{28} r)$$

$$\text{for } N^{28} r > 0.005$$

in which r is the radius of the cylindrical fuel element.

A more complete analysis of the problem requires an extension of the presently available range of the Dancoff correction. A program for the IBM-704 computer has been written to perform these calculations.

2.3 Core Kinetic Analysis

H. C. Hecker
D. Hunter
J. M. Gallagher
R. E. Wolf

Loss-of-Coolant Flow Studies

The contribution of control to the temperature coefficient has been recalculated using better values for the thermal constants, as obtained from the SOFOCATE code, and an improved method of obtaining control rod worths. As before, three combinations of boron concentration and control rods corresponding to different reactivity conditions and temperatures were considered. First, control rods were assumed withdrawn entirely and boron poison added to make $k_{eff} = 1.00$. In the second case, sufficient boron concentration for 5% cold shutdown (when control rods are fully inserted) and control rods inserted to make $k_{eff} = 1.00$ were assumed. Full insertion of 24 control rods was assumed in the third case plus sufficient boron to make $k_{eff} = 1.00$ at lower temperatures; however, at 255°F. no boron is needed to make $k_{eff} = 1.00$ and at higher temperatures it is necessary to partially remove control rods to achieve $k_{eff} = 1.00$.

The effect of dissolved poison and control rods for these three cases was added to the moderator coefficient for the clean core at beginning of life. Figure 4 shows the extreme range in the moderator coefficient under various possible types of control. However, even in the case where only boron is used for control (the upper curve of Figure 4), the total moderator coefficient is $-0.42 \times 10^{-4}/°F.$ at 300°F. Thus, above 300°F. the moderator temperature coefficient is always negative. In the case where a minimum boron concentration was used (control rods fully inserted, see lower curve on Figure 4, the total coefficient is somewhat more negative than that of the uncontrolled core. These assumptions represent the extreme limits, so any other combination of control should fall within the curves given on Figure 4. Uniform void coefficients have been obtained at two temperatures for various core conditions and are given in Table V.

TABLE V

Uniform Void Coefficients for the Yankee Core

	% Δk/k per % Void	
	368°F.	534°F.
Beginning of Life	-.14	-.16
End of Life	-.14	-.18
End of Life with 3 times expected Pu	-.19	-.22

The draft of a topical report has been written ("Complete Loss of Coolant Flow in the Yankee Reactor", YAEC-83) which describes the methods employed and the results obtained in the analog computer study of complete loss of coolant flow in the Yankee Reactor. An appendix in the report describes an approximate method for evaluating the consequences of a partial loss of coolant flow.

A study has been initiated comparing the moment method of approximation of the temperature distribution within fuel pellets during power transients with a procedure involving sectionalization of fuel pellets. The two methods will be applied to a study of transient conditions during startup of the Yankee reactor. The results will be compared in an effort to evaluate the relative merits of the two approximations.

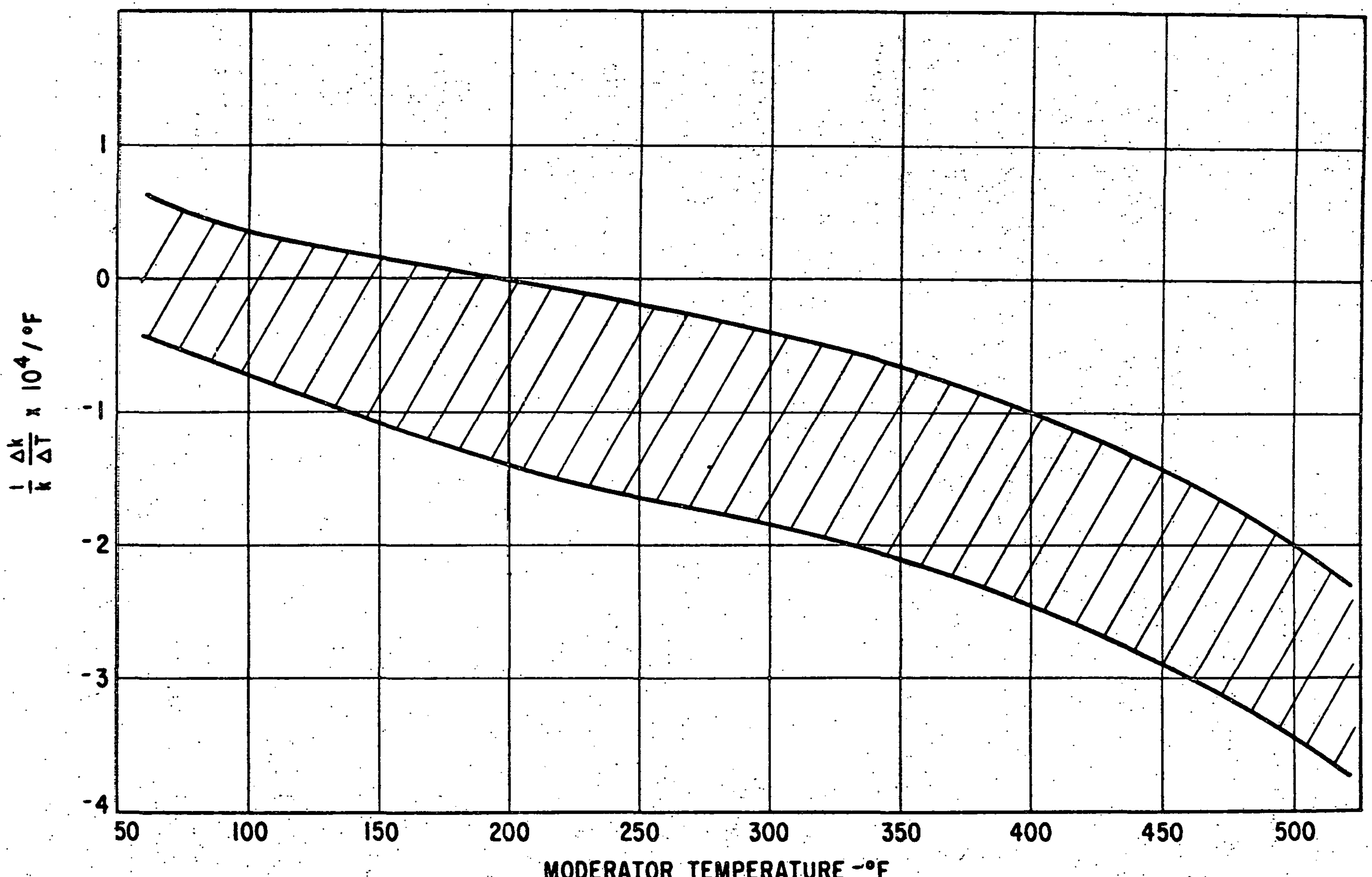


Figure 4 **Possible Range of Moderator Temperature Coefficients, First Yankee Core, Due to Various Methods of Control**

2.4 Control Rod and Chemical Poison Analysis

G. G. Tirellis W. J. Eich R. E. Wolf

Control Rod Worth Studies

A preliminary burnup analysis has been performed on the reference control rod program chosen for the Yankee core to study the effect of this program on the behavior of the nuclear hot channel factor throughout core life. The uniformly loaded core given in the design data sheets has been assumed.

The method employed in this analysis is a flux synthesis technique developed at the Westinghouse Bettis Plant. This technique makes use of the WANDA one-dimensional diffusion theory program for the IBM-704 computer to obtain flux distributions and overall core reactivity. This program is combined with the CAP-1 program developed at Westinghouse APD to obtain flux distributions as a function of fuel burnup for a specific control rod program which renders the reactor "just critical" throughout life.

The steps which are followed in the analysis are:

1. Divide the core into a finite number of radial and axial regions so that each region will be either completely occupied by, or completely devoid of a control rod group. The larger the number of regions selected, the more accurate the simulation of the physical case.

2. On the basis of unit cell calculations, determine the uniformly distributed poison cross section which will yield the same reactivity change in a region as a completely inserted control rod (or rods).

3. Divide the core lifetime into a series of time intervals and determine a control rod pattern which renders the reactor very nearly critical at the beginning of each interval i.e. a combination of controlled and uncontrolled regions for which $k_{eff} = 1.00$.

4. Calculate radial and axial flux distributions for the above control rod pattern, and find volume averages of power density for each of the regions of the core. This step is accomplished by first calculating a radial flux distribution for each of the axial regions considered. From these radial flux distributions flux weighted cross sections and radial leakages are obtained and used as input to an axial problem. The resulting axial and radial one-dimensional plots then are combined to yield the core power distribution.

5. Using the region volume averages of power density obtained in step 4, calculate the change in fuel composition for a finite time step using the CAP-1 program for fuel burnup. The new values of fuel composition are then used as initial conditions for the next interval in the burnup analysis. Steps 3 through 5 are then repeated for each interval until rods are completely removed from the core and k_{eff} falls below unity.

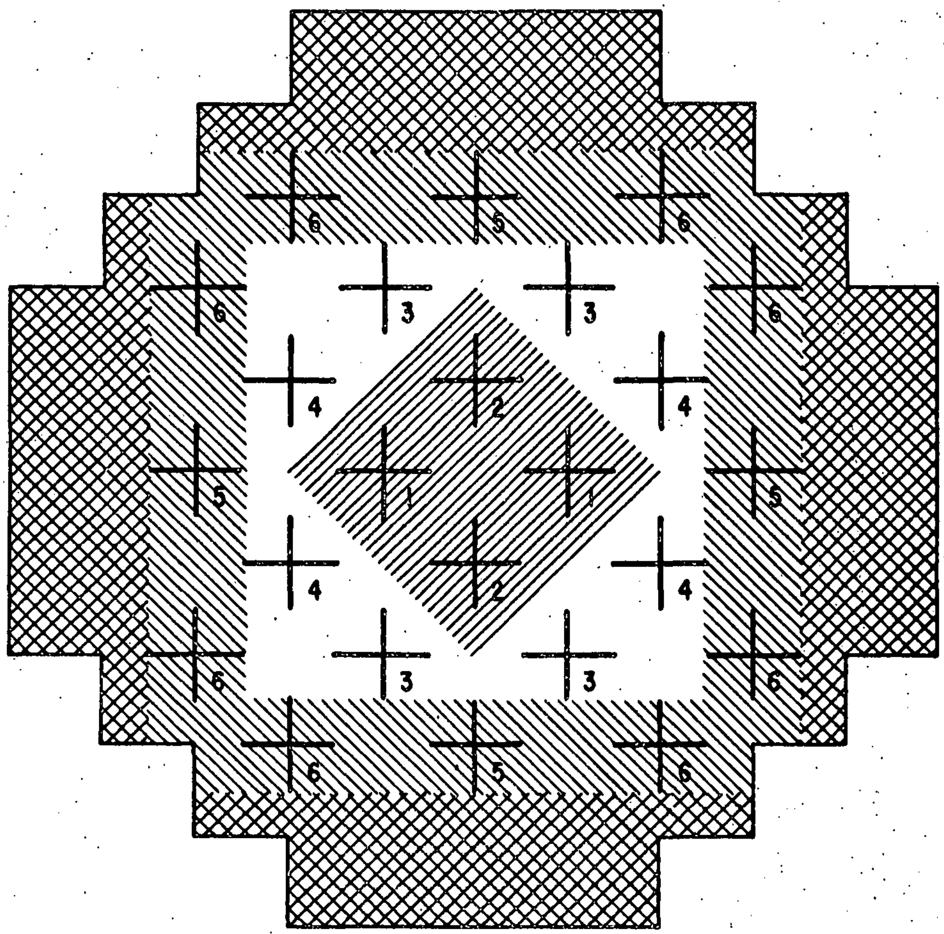
The core was divided into eight regions in this preliminary analysis, two axial regions each containing four concentric radial regions. The four radial regions were defined by homogenizing two fuel assemblies with each control rod and obtaining the equivalent locations of the core areas containing; (1) groups 1 and 2; (2) groups 3 and 4; (3) groups 5 and 6; and (4) the unrodded balance of the core. (The boundaries on Figure 5 show these regions).

The reference control rod program consists of progressive, stepwise, outside-in rod withdrawal such that each step consisted of control rod group movement over half the core length. Referring to the rod numbering and worth tables shown on Figure 5, the assumption was made that the core would be controlled initially by Groups 1 and 3 fully inserted and Groups 5 and 6 withdrawn halfway. In the following paragraphs, references made to Groups 1 and 3 apply in equivalent fashion to Groups 2 and 4, respectively; Groups 1 and 2 and Groups 3 and 4 would be alternated frequently to average the burnout effects.

The assumption was made that the rod configuration described above was maintained until the reactivity decreased to such a value that Groups 1 and 3 fully inserted would control the excess k_{eff}. Again, it was assumed that this position of the two rod groups was held until the excess reactivity declined to a value that Group 1 fully inserted and Group 3 halfway inserted could control the core. Rod groups inserted halfway were assumed to possess half their fully inserted worth. This stepwise method of withdrawal was continued until the end of life was reached. The times for each interval were obtained by normalizing the time-dependent k_{eff} obtained from a CAP-1 burnout study to a 10,000 hour lifetime. Figure 6 shows the 5 time intervals which were used in the analysis. The initial value of k_{eff} (1.057) given on Figure 6 is somewhat below the value of 1.065 given in the design data sheets because it has not been raised to allow for nonuniform fuel burnup.

Figures 7 and 8 show the axial power density in the hottest fuel assembly at the end of the first and last time intervals. The discontinuity at the boundary between the lower and upper regions is due to the use of a different average fission cross section in the two regions. This difference is a consequence of both a difference in fuel burnup and the technique used to obtain weighted cross sections.

The primary disadvantage of this technique is that it assumes that the radial flux shape within an axial region is not dependent upon axial position in that region. Experimental flux distributions from the



CORE REGIONS 1 2 3 4

GROUP	RELATIVE WORTH ALL INSERTED	ACTUAL WORTH (% $\frac{\Delta k}{k}$)
1	.162	2.43
2	.162	2.43
3	.207	3.11
4	.207	3.11
5	.124	1.86
6	.138	2.07

Figure 5 Yankee Control Rod Groups and Nominal Worths, First Core

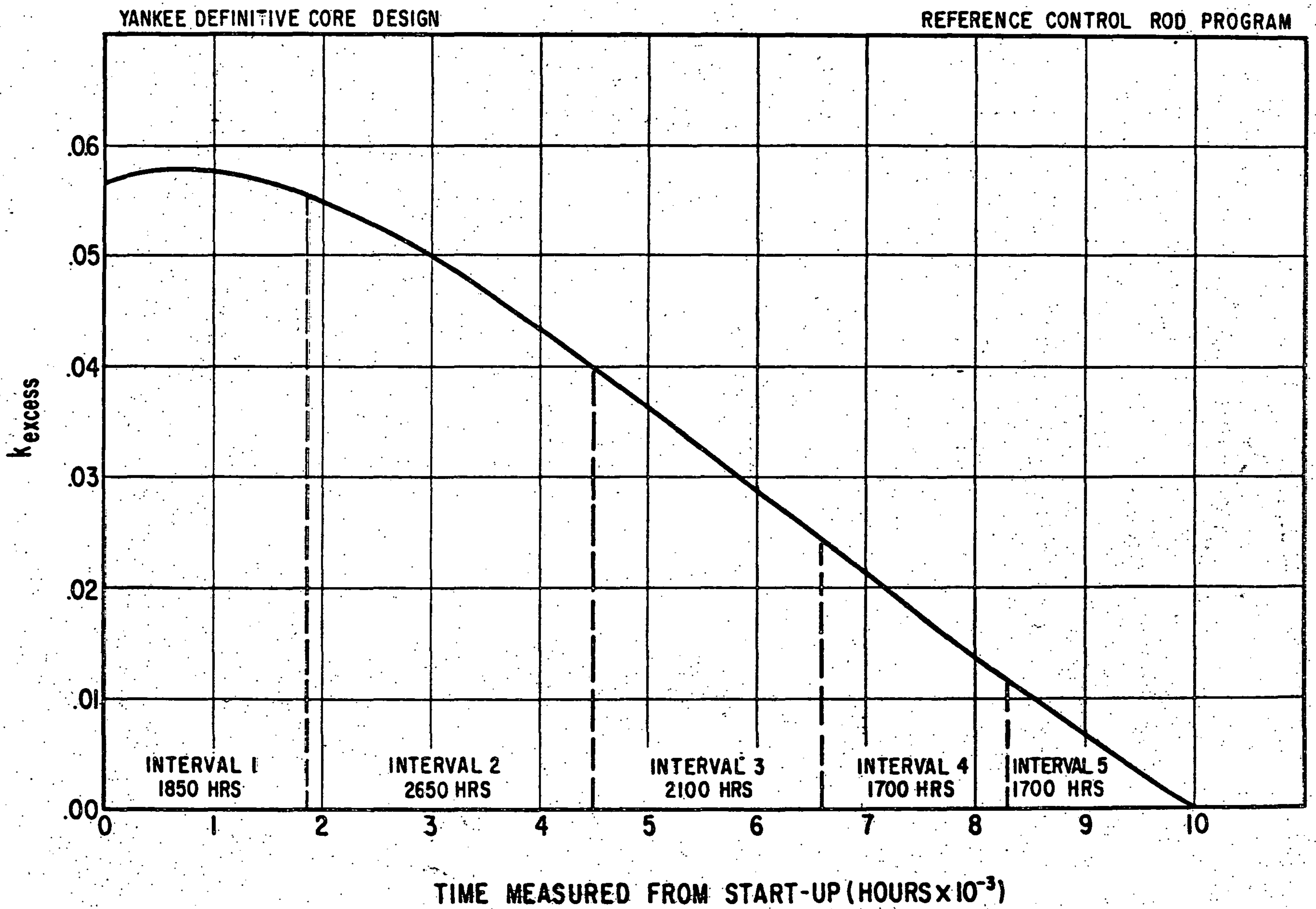


Figure 6 Time Intervals During Which Configuration of Yankee Control Rods Remains Constant

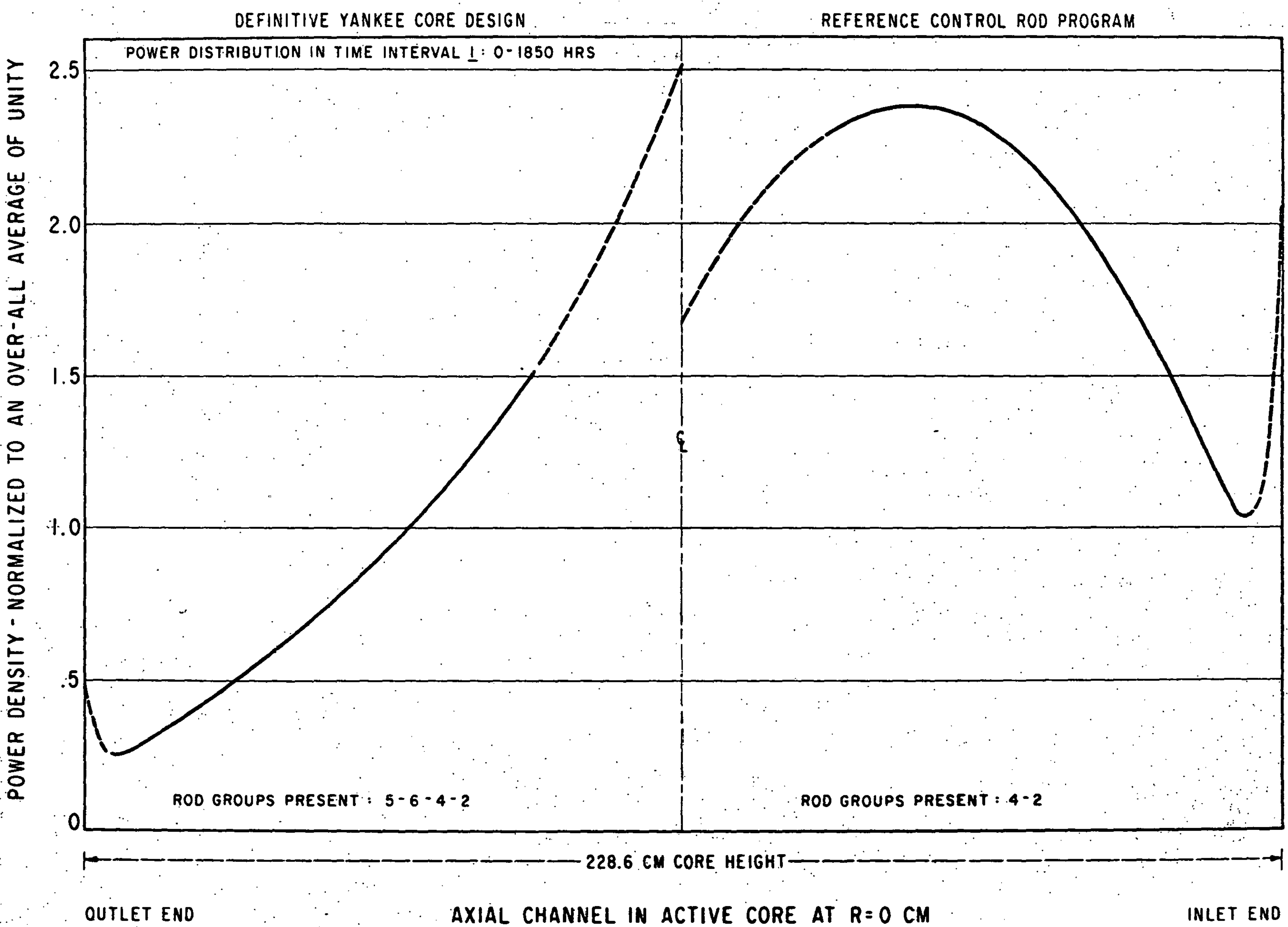


Figure 7 Ratio of Hot Channel to Average Power Density During First Interval of Core Life

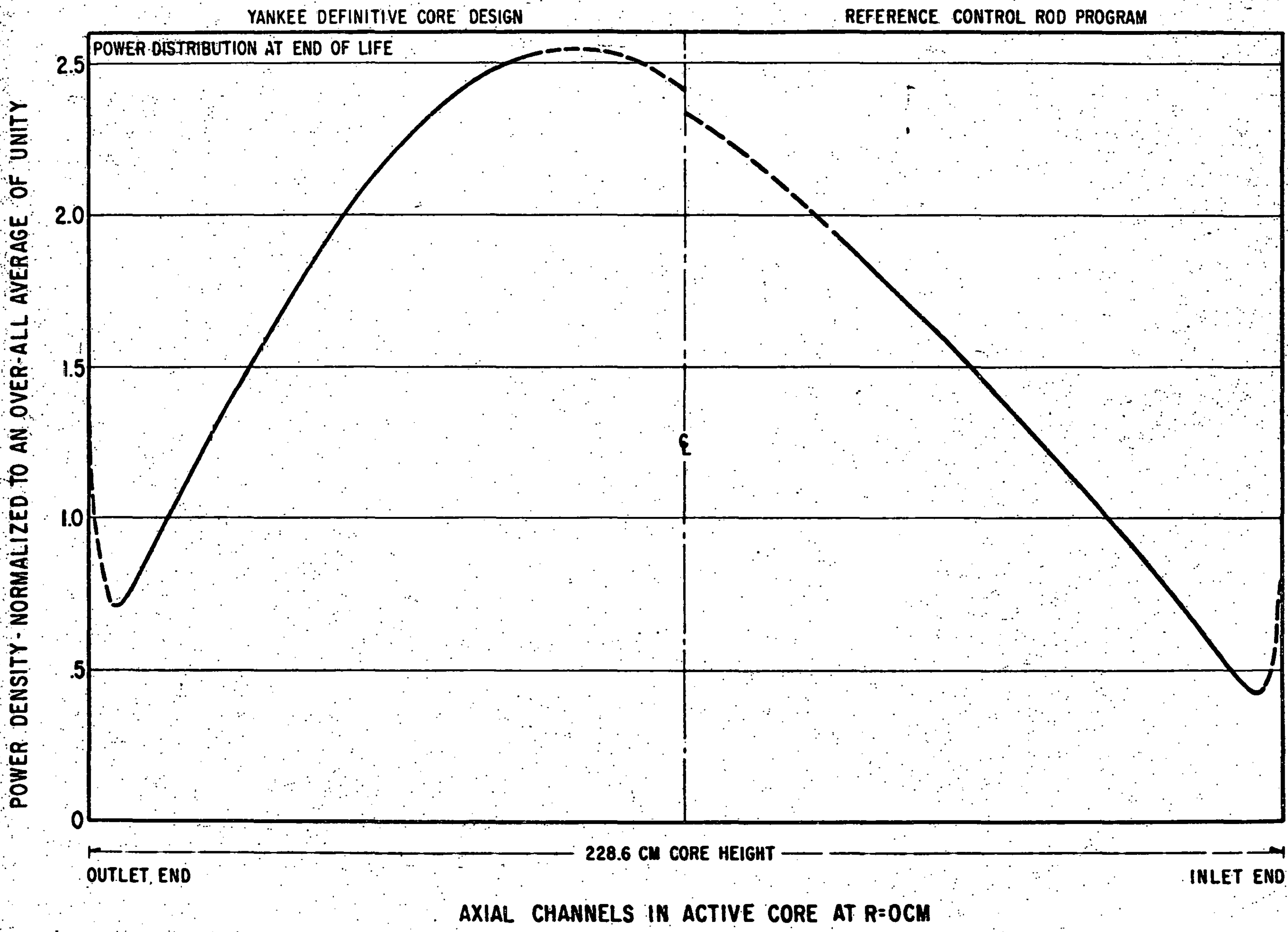


Figure 8 Hot Channel Power/Average Power in Yankee First Core (10,000 hours)

Yankee critical experiments and other experimental facilities indicate that this is not the case. This disadvantage can be offset by using the PDQ two-dimensional digital computer program which has recently been made available. Another disadvantage is that a large number of core regions are required to accurately simulate a given control rod program, since the method approximates continuous control rod movement with finite steps in control movement.

Based on the control rod program assumed, this preliminary study has revealed that:

1. The reference control program appears to be adequate to maintain the hot channel factor below the value used in the design to date.

2. During the core's lifetime, the magnitude of this overall nuclear F_Q varies between 1.9 and 2.5. Since the core was analyzed radially using the one-dimensional WANDA code, no account was taken of any local nuclear peaking factor. One should bear in mind that, because of the added inhomogenieties due to having combinations of rods and followers not previously considered, the peaking factor might be larger here than the 1.2 usually assumed.

3. Before half the core lifetime has been completed, the peak power density moves radially from the central position to an annular "channel" located approximately one-third of the core radius from the centerline. Although the location of the peak later returns to the centerline position, the power distribution along the radius is somewhat flat in the central part of the core as the end of core life is approached.

Effort is now being directed toward the use of the PDQ program for the solution of the same problem. A more detailed analysis of control rod programming in the uniformly loaded core will be necessary before a sound technical decision can be made regarding the final fuel configuration for the Yankee reactor.

2.5 Critical Experiment Planning and Analysis

J. Jedruch, V. E. Grob, D. Hunter, R. French, H. Hecker

Buckling and Reflector Savings Evaluation

Experiments on the 3:1 water/uranium volume ratio critical lattice have been made to obtain flux and power measurements along the axial and radial directions of the core. These measurements were performed using indium and gold foils as well as fission product decay activity of the fuel rods. The data thus obtained require reduction to a form acceptable for mathematical analysis. Such a reduction requires a correction of the raw data for counter deadtime, background, and efficiency as well as correcting for radioactive decay and foil weight (if foils are used). These corrections have been made by hand, and the data have been prepared for least squares fitting to some predetermined shape.

Attempts to hand-fit the data to the best argument of the cosine and Bessel J_0 functions using the least squares technique, have proven to be laborious and liable to arithmetical errors. To avoid these difficulties, COFIT, a least squares fitting code developed at Bettis for the IBM-704 computer has been obtained. The COFIT code produces a three parameter fit to the function $Y = A \cos B (X-C)$. The parameter B is the required axial buckling of the neutron flux. In addition to obtaining the peak value, A, and the center shift, C, the code calculates the extrapolated core height, H', and the standard deviation of these four parameters. The standard deviation of the experimental points provides a good measure of the scatter of the points.

In experiments involving an approach to criticality with the core completely covered with water, the actual core height, H, is a known quantity. Subtraction of this quantity from the extrapolated core height, H', produces directly the total (top plus bottom) reflector savings. Since the core height is accurately know, the error in the extrapolated core height can be assigned wholly to the reflector savings. The results in Table VI are for reflector savings on one end; they were obtained by dividing the total axial reflector savings, and its error, by two.

TABLE VI

Results of COFIT Calculations of Axial Reflector Savings - 3:1 W/U CRX Core

Run	Reflector Savings (One End)	Standard Deviation
64-III-7	4.16 cm	± 0.51 cm
64-III-10	3.18	± 0.63
164-III-10	8.30	± 0.75
161-IX-21	4.54	± 0.53
1, 162-IX-22	7.88	± 0.51
2, 162-IX-22	6.67	± 0.22
162-IX-22	6.16	± 0.26
163-IV-23	9.03	± 0.62
Weighted Average	6.30	± 0.46

Fitting of flux and power measurements taken along the radial direction of the core had been initially delayed because no Bessel J_o least squares fitting code was available for the IBM-704 computer. This difficulty has been surmounted by modifying the COFIT code to fit to the J_o Bessel function. The modified version of the code has been named JOFIT and is now operational.

The JOFIT code fits the experimental points to the best value of the arguments of the function $Y = A\ J_o\ [B\ (X-C)]$. Both the input and the output format are quite similar to those of COFIT except that the code calculates the extrapolated core radius, R^1, instead of the core diameter. The radial reflector savings given in Table VII were obtained by subtracting the actual core radius from the extrapolated radius and assigning all error to the reflector savings.

TABLE VII

Results of JOFIT Calculations for Radial Reflector Savings - 3:1 W/U CRX Core

Run	Reflector Savings (One End)	Standard Deviation
50-II-32	6.03 cm	± 0.30 cm
59-III-2	7.18	± 2.14
60-III-3	2.07	± 0.83
60-III-3	5.82	± 1.07
66-III-9	4.91	± 1.14
66-III-9	10.48	± 1.94
68-III-11	3.02	± 1.17
78-III-11	8.06	± 1.25
105-V-7	7.37	± 1.03
105-V-7	7.37	± 1.27
105-V-7	6.36	± 1.44
147-IX-7	7.73	± 0.56
147-IX-7	6.97	± 0.55
161-IX-21	8.17	± 0.89
162-IX-22	7.40	± 0.20
162-IX-22	7.31	± 0.16
Weighted Average	7.05	± 0.23

As mentioned previously, a major difficulty in the evaluation of the flux measurements has been the amount of time and effort required for the reduction of data to analyzable form. It was deemed advisable, therefore, to develop a code for the IBM-704 computer to assist in data reduction. The proposed code has been named DARED (data reduction). It is designed to reduce up to 100 points of data obtained from foil irradiations or fuel rod scanning to a form directly usable as input to the COFIT and JOFIT codes. The code will apply the deadtime, background, and decay corrections and will correct the results for variations in foil weight. The code will also calculate the space ordinate of the foil position in the core and find the statistical weight of the given point. The code has been written and is in the process of being operationally checked.

Flux Peaking in Water and Rod Follower Slots

A comparison of calculated values with flux peaking data measured at the Westinghouse Reactor Evaluation Center has indicated that a two-group diffusion theory calculation may be used to obtain a reasonable approximation (within a few percent) of the thermal flux peak in a water slot. Fast group constants were obtained using standard hand-calculation techniques, and thermal group constants were obtained by averaging over a Wigner-Wilkens spectrum (SOFOCATE) code. In the case of thermal constants, the spectrum used in the water slot was that for pure water. The close agreement indicates that the transition in the thermal neutron spectrum in traversing from region to region is rather abrupt.

Age Calculation for CRX

Review of available experimental data from other facilities has indicated the necessity of a revision of the currently used calculation of age. The change increases the effect of inelastic scattering in uranium. This lowers the calculated age in the Yankee critical core from 49 to 45 cm^2 and improves the agreement with experimental data.

Comparison of CANDLE 4-Group Calculation with CRX Experimental Data

As a check on the criticality calculation in the CANDLE four-group non-uniform burnup code for the IBM-704 computer, a calculation was made of an exactly critical core as assembled at the Westinghouse Reactor Evaluation Center. The burnup part of the machine calculation was, of course, ignored. The core considered was a circular cylinder with a loading of 1862 fuel rods and a 3:1 water/uranium volume ratio. The thermal group constants were determined both by averaging over a Maxwellian neutron spectrum and by averaging over a Wigner-Wilkens neutron spectrum as calculated by the IBM-704 computer SOFOCATE code. The resulting multiplication factor was 1.041 when Maxwellian thermal constants were used, and 1.030 when Wigner-Wilkens thermal constants were used.

A check of the CANDLE fast absorption cross sections for stainless steel indicated that they were low by a factor of two. Evaluation of the multiplication factor with the increased fast absorption in stainless steel resulted in a value of 1.021 when Wigner-Wilkens thermal constants were used.

A comparison of the effective age of neutrons in the critical core, as calculated by CANDLE, and the experimentally measured migration area (37.4 cm^2 and 42 - 47 cm^2 respectively) has indicated that an adjustment of CANDLE fast group constants is necessary to bring the two into closer agreement. It has been estimated that a 10% increase in effective age would result in a reduction in the multiplication factor, calculated by CANDLE, of 2%.

Hazards Evaluations for CRX Cores

Evaluation was completed of the hazards associated with the 4:1 W/U and 2.23:1 W/U ratio Yankee critical cores. The ramp reactivity insertions studied on the analog computer correspond either to water insertion at maximum pumping speed or to control rod removal at maximum design speed. From the power response and temperature responses that were obtained as computer outputs, the amount of fuel melting and hence the fission product dispersion were calculated. Figure 9 shows a typical set of responses for the 2.23:1 W/U ratio core. This figure shows the power and temperatures which result when water is added to the CRX tank when it contains the maximum dissolved boron concentration. The dilution of boron results in reactivity insertion at the rate of $1.43 \times 10^{-2} \Delta k/k$ per second. These studies have been written as YAEC-31 Supplement 1 for the 2.23:1 W/U ratio and YAEC-31 Supplement 2 for the 4:1 W/U ratios. In addition to these studies, the hazards associated with the 3:1 W/U ratio critical experiment were re-evaluated using the improved mathematical representation developed since the original studies were performed.

Because of the variety of W/U ratios and fuel element configurations analyzed for the various critical experiment cores, a large range of parameter values are encountered in critical hazards analysis. Therefore, a parameter study was undertaken in order to determine the sensitivity of a low enrichment, UO_2 fueled critical assembly to each parameter. The following parameters were examined:

1. Doppler Coefficient of Reactivity

2. Moderator Coefficient of Reactivity

3. Neutron Lifetime

4. Total Delayed Neutron Fraction

5. Heat Transfer Conductivity from Stainless Steel to Water

In addition to the preceding parameters, a comparison was made of the techniques used for solution of the problem, e.g. a comparison of the types of mathematical models used to describe the physical system.

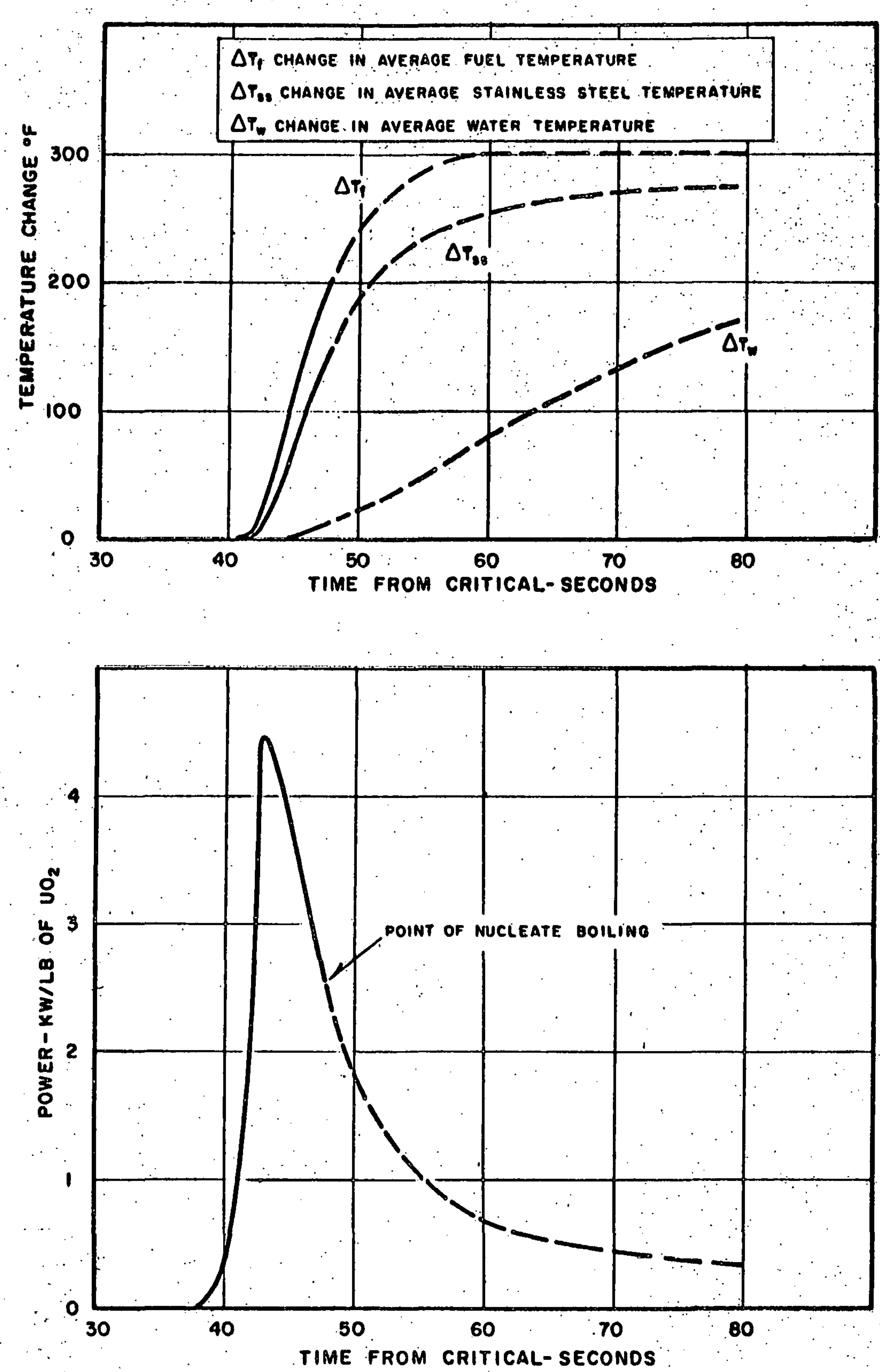


Figure 9 Kinetic Response of the 2.23:1 CRX Core to a Ramp of 1.43×10^{-2} %Δk/k per Second

The 3:1 W/U ratio Yankee critical core was used as the physical system upon which all changes were imposed. Extremes in ramp rates were examined for resulting parameter changes since the effects are dependent upon this variable. Hereafter, $4.6 \times 10^{-4} \Delta k/k$/sec will be referred to as a fast ramp and $1 \times 10^{-4} \Delta k/k$/sec as a slow ramp.

The fuel conductivity and moderator coefficient of reactivity were found to affect only the "tail" of the power response curve for fast ramps as was expected from the examination of the thermal time constants. That is, the propagation of heat to the moderator and subsequent feedback through the moderator coefficient takes place at a rate determined by the time constant associated with the fuel and stainless steel thermal characteristics. For slow ramps, an effect was observed when peak power was reached. Inasmuch as fast ramps are of primary importance in a hazards analysis, the effects associated with them are of primary significance.

The delayed neutron fraction varied from 0.0064 to 0.0075 and was found to have little effect on the determination of the power response curve. The only appreciable change was a shift of the time at which the peak occurred.

The Doppler coefficient of reactivity affects the magnitude of the transients more than any parameter examined, as shown in Figure 10. In this figure, K_w and K_f represent the negative temperature coefficient of reactivity for the moderator and the fuel and Υ is the ramp rate of inserted reactivity. The peak power, normalized to the peak for a coefficient of $0.6 \times 10^{-5} \Delta k/k\,^{\circ}F.$, was approximately proportional to the magnitude of the normalized coefficient. For a ramp input of reactivity, an increase in the value used for the neutron lifetime resulted in an increase in the peak power attained in the transient as shown in Figure 11. This effect appeared to contradict all physical "feeling" for the problem. Examination of the input $\Delta k/k$ function in terms of the neutron kinetic equations showed that a larger value of neutron lifetime would give a lower peak value, however, for a step increase in reactivity the larger value of neutron lifetime would give a larger peak value for a ramp with the reactor parameter considered in this study.

2.6 Irradiation Experiment Design and Analysis

W. J. Eich

MTR In-Pile Test Loop and Process Water Fuel Enrichment Studies

Calculations were performed of fuel enrichments for fuel rods containing 0.294 inch diameter pellets to be placed in the in-pile test loop and process water irradiation holes located in the MTR reflector. Results of calculations to determine the enrichments of rod samples containing 0.300 inch diameter pellets within a 0.021 inch clad were reported in YAEC-65, Quarterly Progress Report for the January 1 to March 31, 1958 period.

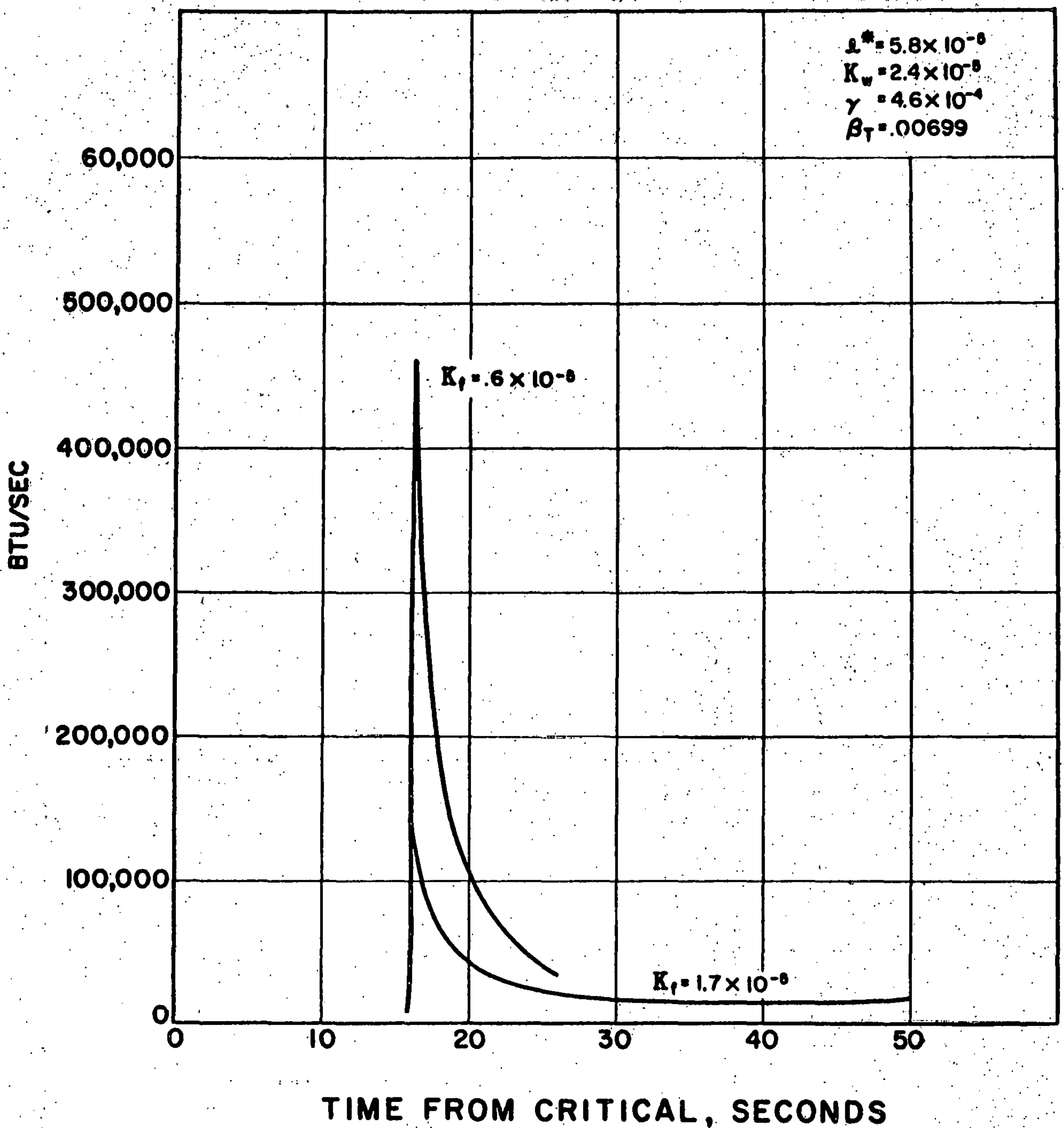


Figure 10 Response of the 3:1 W/U Yankee CRX Core to a Ramp Increase in Reactivity for Two Values of the Fuel Temperature Coefficient

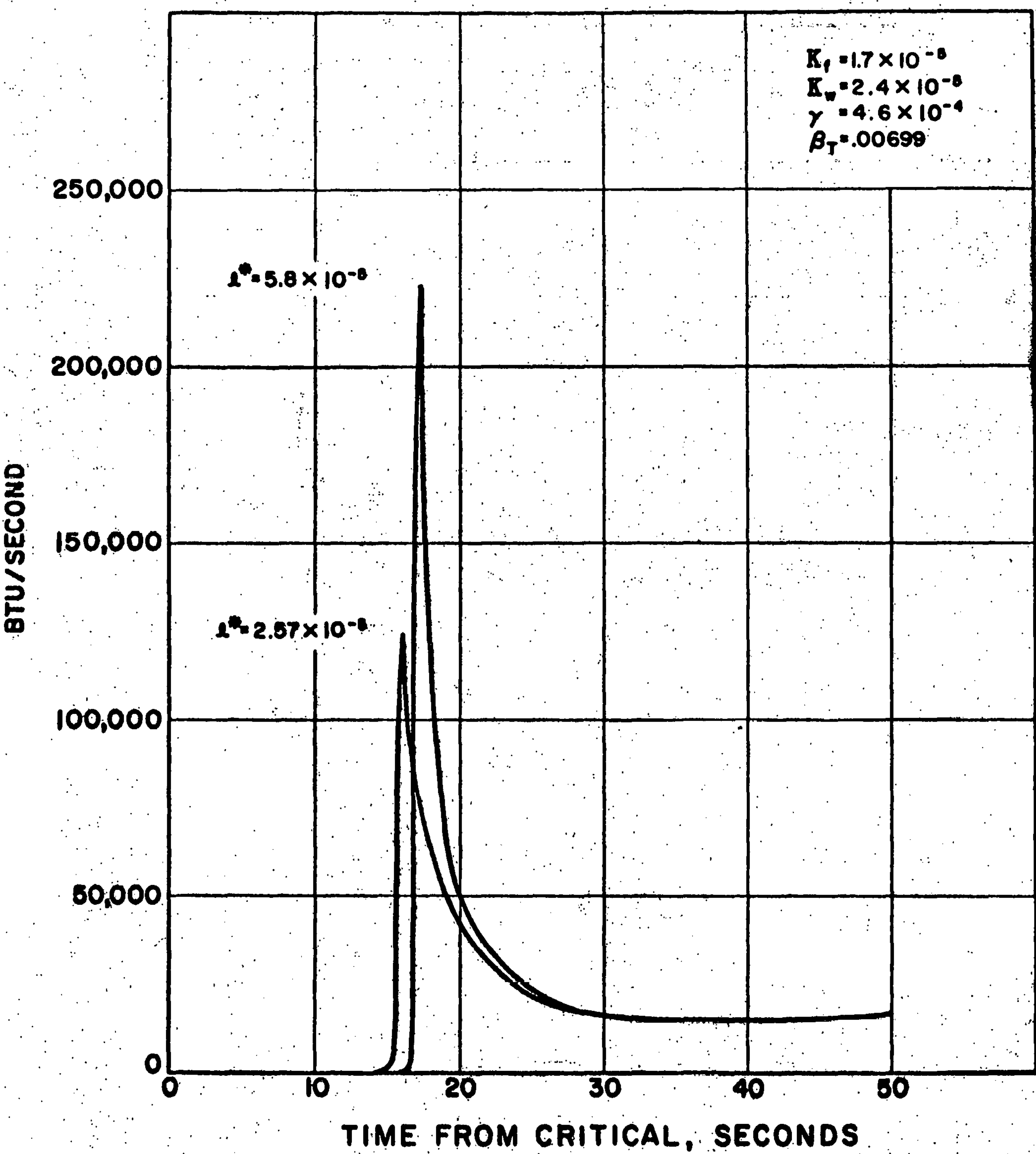


Figure 11 Response of the 3:1 W/U Yankee CRX Core to a Ramp Increase in Reactivity for Two Values of the Prompt Neutron Lifetime

Calculations have since been performed for rod samples of a somewhat different design; these rods would contain pellets 0.294 inch in diameter, enriched to 5.4%, and clad in either 15 or 21 mil stainless steel. Semi-empirical in nature, these calculations determined that an "in-process water" irradiation hole characterized by an unperturbed thermal neutron flux of $6.2 \times 10^{13}/cm^2/sec$ would be required to produce a heat flux of 6×10^5 Btu/ft^2 hr at the surface of the 0.015 inch clad and that a characteristic flux of $6.4 \times 10^{13}/cm^2/sec$ would be required for the 0.021 inch clad samples to produce the same surface heat flux.

In YAEC-65, the pellet enrichments required to fulfill the in-pile test specifications were reported. The enrichments specified were 10% for pellets in the low flux leg and 27% for pellets in the high flux leg. The effects of substituting rod samples containing 0.294 inch pellets enriched to 10% and 27% was also determined: in both legs, the design value of maximum heat flux - 4.5×10^5 Btu/ft^2 hr - would be exceeded by 3% at the surface of the 0.021 inch clad and 7% at the surface of the 0.015 inch clad. This is not a significant difference.

2.7 Shielding Analysis

No work was performed under this subproject during the second quarter of 1958.

2.8 Startup Experiment Assistance

No work was performed under this subproject during the second quarter of 1958 in accordance with the Research and Development Program for the period from January 1 to June 30, 1958 as described in YAEC-41.

3.0 CHEMISTRY

Chemistry and Ceramics Section: R. F. Sterling, Manager

Studies of stability, corrosion effects, removal of nuclear poisons, crud problems, fuel element rupture problems and the specification of the primary coolant are included under this project.

3.1 Properties and Removal of Chemical Neutron Absorbers

N. Michael

Ion Exchange of Radioisotopes

Mixed Fission Products

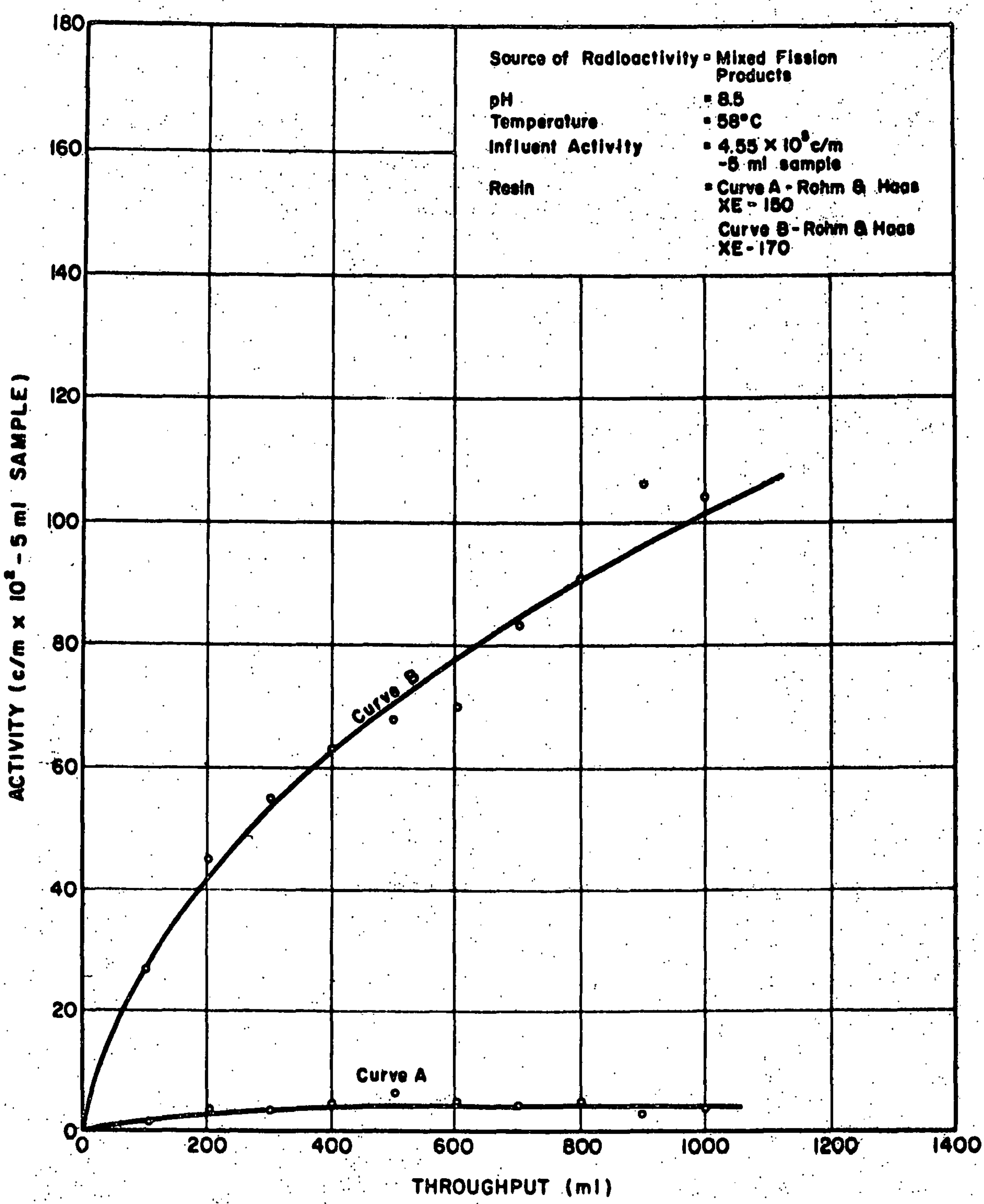
Rohm and Haas mixed-bed ion exchange resins XE-150 (H^+, OH^-) and XE-170 (NH_4^+ OH^-) were tested with a fission product solution (Catalog item FP-P, Mixed Fission Products, available from ORNL).

The pH of the test solution was 8.5, obtained by adjustment with NH_4OH, a temperature of 58 °C., a specific activity of 4.55×10^5 c/m in a 5 ml sample. The XE-150 gave a mean D. F. (decontamination factor) of 1154, with a standard deviation of 255 and a relative deviation of 22%. With the XE-170 the decontamination factors ranged from 103, after a 200 ml throughput, to 44, after a 1000 ml throughput. Both resins were tested with 1000 ml volumes of test solution. Figure 12 illustrates the relative decontaminating efficiencies of the two resins. Figure 13 shows the activity distribution in each column at the conclusion of the run.

The mean D.F. for XE-150 resin obtained at a pH 1.0 was 1.8×10^5, at a pH 2.1, 1.8×10^4. Therefore, it may be concluded that the decontaminating ability of XE-150 resin is some inverse function of the pH. One explanation for this phenomenon is that the fission product ion concentration diminishes while the radiocolloid concentration increases with a rise in pH.

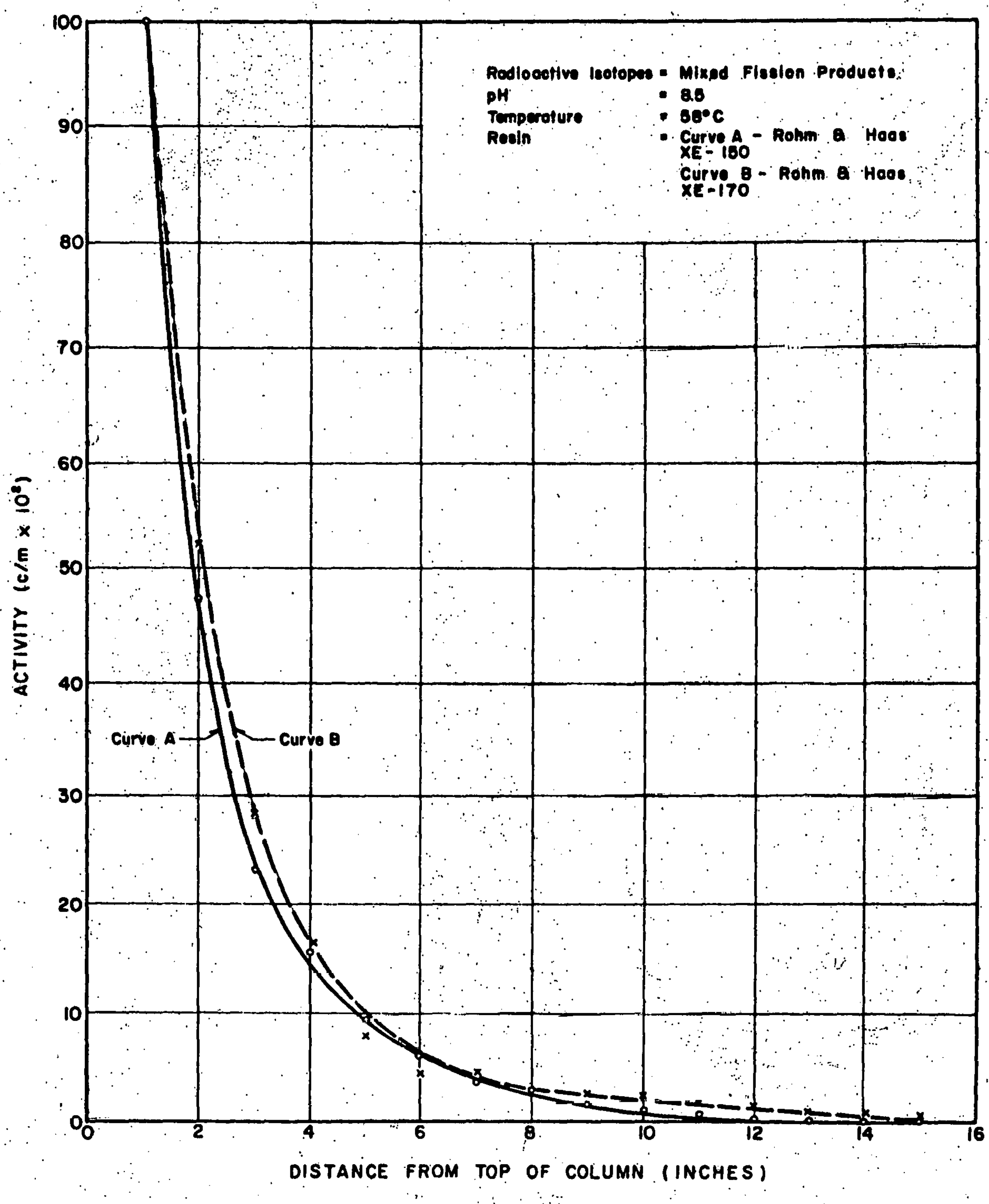
Corrosion Products

Radioactive decontamination factors were determined using a synthetic corrosion product mixture and Rohm and Haas XE-150 and XE-170 ion exchange resins. A 1500 ml volume of solution was passed through 100 ml of each type of resin. The pH of the solution was 8.6 obtained by the use of NH_4OH and the concentration in ppm of tagged synthetic corrosion products was Fe, 0.06; Cr, 0.007; Co, 0.012. Decontamination factors obtained for XE-150 remained fairly constant throughout the run. The mean of eleven separate determinations was 335.3; the standard deviation, 52.5; and the relative deviation 15.6%. With the XE-170 resin, successive



ACTIVITY OF COLUMN EFFLUENT
DURING FISSION PRODUCT RUN IR-6

FIGURE 12



DISTRIBUTION OF ACTIVITY AT CONCLUSION OF RUN IR-6

FIGURE 13

samples showed increasingly lower decontamination factors. The D.F. values ranged from 12 for the first sample to 6 for the last. The mean value of nine determinations was 7.7; the standard deviation, 1.8; and the relative deviation, 23.7%. Figure 14 graphically illustrates the continuing increase in activity leakage through the XE-170 resin. In the same figure the curve for XE-150 is shown. Here a plateau region is reached after a throughput of only 100 ml, indicating leakage had reached equilibrium. Figure 15 shows the activity distribution on the columns at the conclusion of the corrosion product run. With the XE-150 resin, more activity is concentrated at the top of the column than with the XE-170. The above results indicate that XE-150 has superior decontaminating qualities in the vicinity of pH 8.6 than does XE-170.

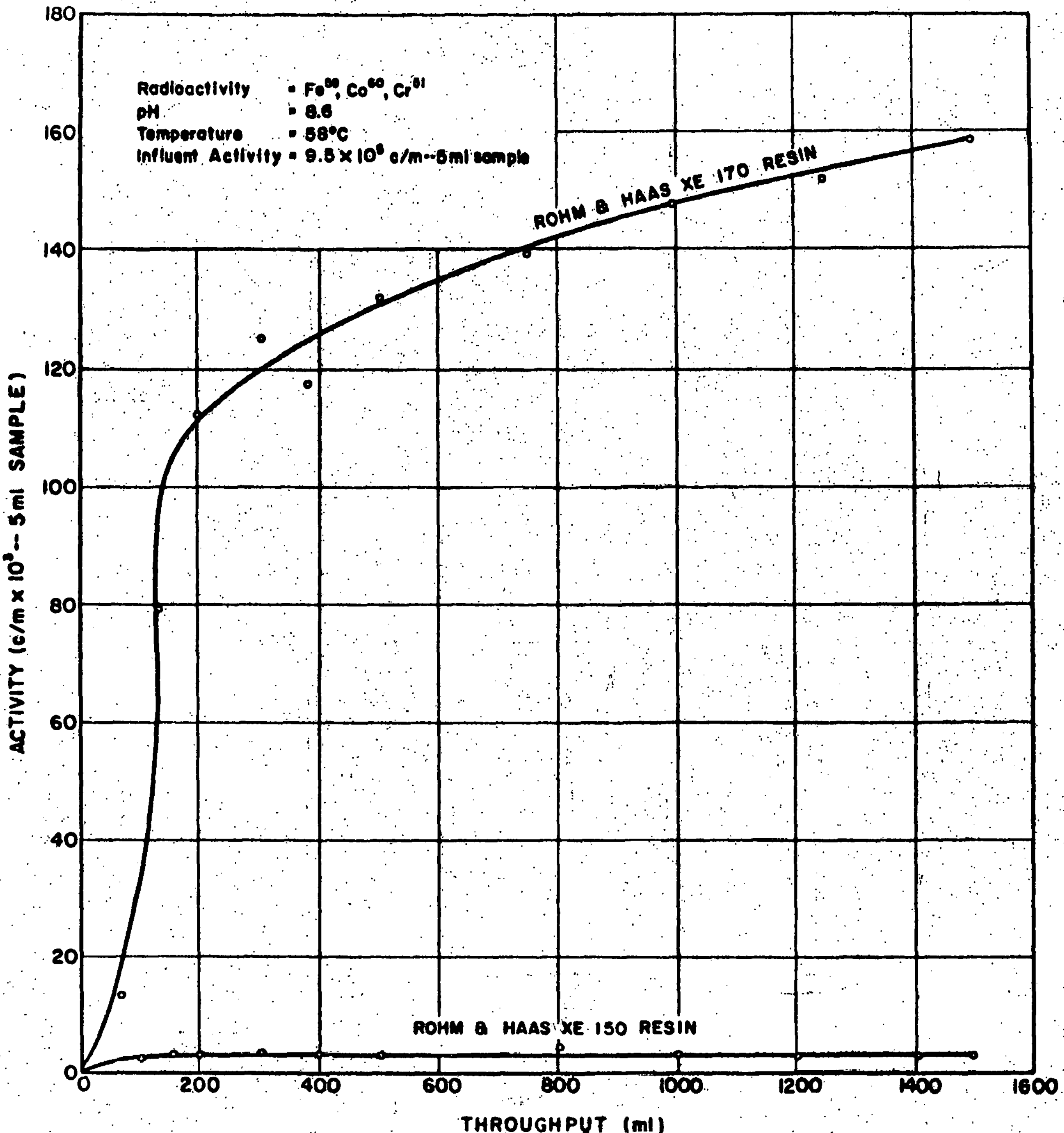
In order to determine whether there is selective pick-up of individual corrosion products, runs are planned with XE-150, XE-170, and XE-154 (Li^+, OH^- mixed bed) and the separate corrosion products, Fe, Cr, and Co, in the same concentrations and pH that were used for the corrosion product mixture. If substantially different decontamination factors are obtained for the separate isotopes, as compared with the mixture, the effect of pH and concentration in the operating efficiency of the resin for each specific element will be studied. This will enable predictions to be made for resin performance with any possible mixture. In line with this planned work, runs were completed with 0.012 ppm Co and the three previously mentioned mixed-bed resins. Results are not yet available.

Iodine Retention on Ion Exchange Resins

The retention of iodine on ion exchange resin is of interest, since it has found use as a sensitive indication of fission product release to reactor coolant. Decontamination factors for I^{131} were determined at a pH 8.5 obtained by the addition of NH_4OH with Rohm and Haas XE-150 resin. The average D.F. was 4.6×10^3. The standard deviation was 379 and the relative deviation, 8.0%. The D.F. previously reported for I^{131} at pH 7.0 was 2.6×10^4. The distribution of activity as shown in Figure 16 is essentially the same as that reported in the pH 7.0 experiment. Here, as with the mixed fission products, increasing pH has an adverse effect on the D.F.

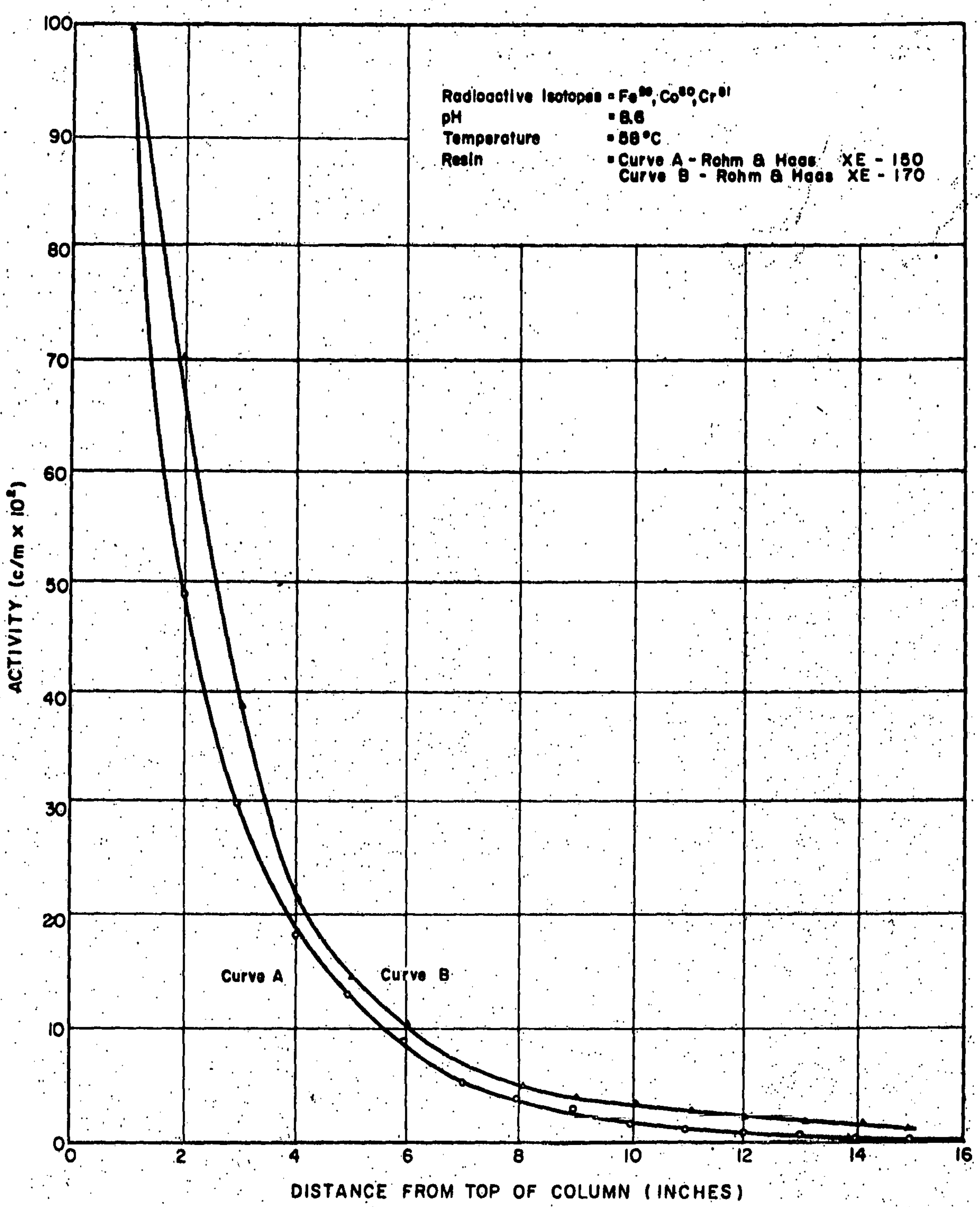
Properties of Ion Exchange Resins

It was found that Rohm and Haas XE-170 releases ammonium hydroxide on standing. A 100 ml column of resin was washed with demineralized water until the effluent had a pH of 7.5 and a conductance of 1.5 micromhos. After being stoppered, the column was put aside for 24 hours. At the end of this period, on passing additional demineralized water through the column, the effluent showed an increase in pH, to 9.0, and in conductance, to 3.5 micromhos. With further rinsing, the pH dropped to 7.0 and the conductance to 1.0 micromho.



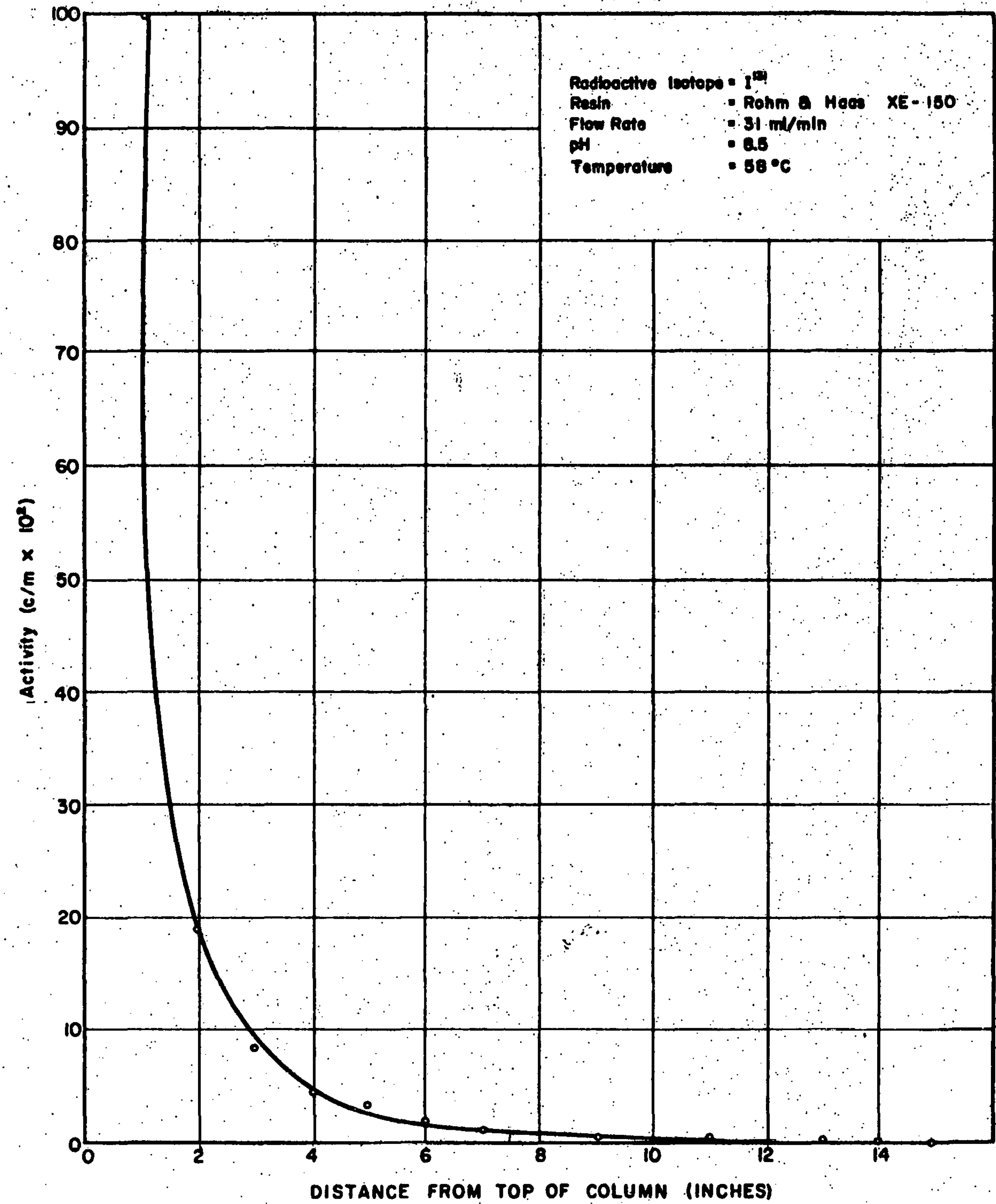
ACTIVITY OF COLUMN EFFLUENT
DURING CORROSION PRODUCT RUN

FIGURE 14



DISTRIBUTION OF ACTIVITY ON COLUMN
AT CONCLUSION OF CORROSION PRODUCT RUN

FIGURE 15



DISTRIBUTION OF ACTIVITY ON COLUMN AT CONCLUSION OF IODINE RETENTION RUN

FIGURE 16

A similar experiment was carried out with Rohm and Haas XE-154. The resin was rinsed with 2.8 liters of high purity demineralized water. Conductance of the effluent was observed to drop during the rinsing from 16 to 4.8 micromhos and the pH from 9.9 to 8.6. The column was stoppered for 16 hours and then rinsed again with demineralized water. With the second rinsing, the effluent showed an initial conductance of 7.7 micromhos and a pH of 9.9. Effluent quality gradually improved with additional rinsing. After using approximately three more liters of demineralized water in a second flushing, the pH dropped to 7.1 and the conductance to 4.3 micromhos.

Boric Acid Removal

Boric acid capacity tests were completed for Nalcite SBR and Rohm and Haas XE-78 resins under "high pH" water conditions. Water pH was adjusted to 10 with LiOH and boric acid was added to give a concentration of 94.5 ppm. Flow through the columns was maintained at 25 ml/min (1.9 GPM/ft^3), and 130°F. At the conclusion of the test, the boric acid was eluted from the resin with NaOH. Analysis of the eluate was used to indicate the resin's total capacity. XE-78 showed a capacity for boric acid of 5.6 lbs/ft^3 and the SBR, 3.7 lbs/ft^3. This experiment is now being repeated with 40 ppm boric acid solution, however, it now appears that the small amount of LiOH added has little effect on the total capacity of the resins for boric acid exchange.

Conditions for Nuclear Poison Storage

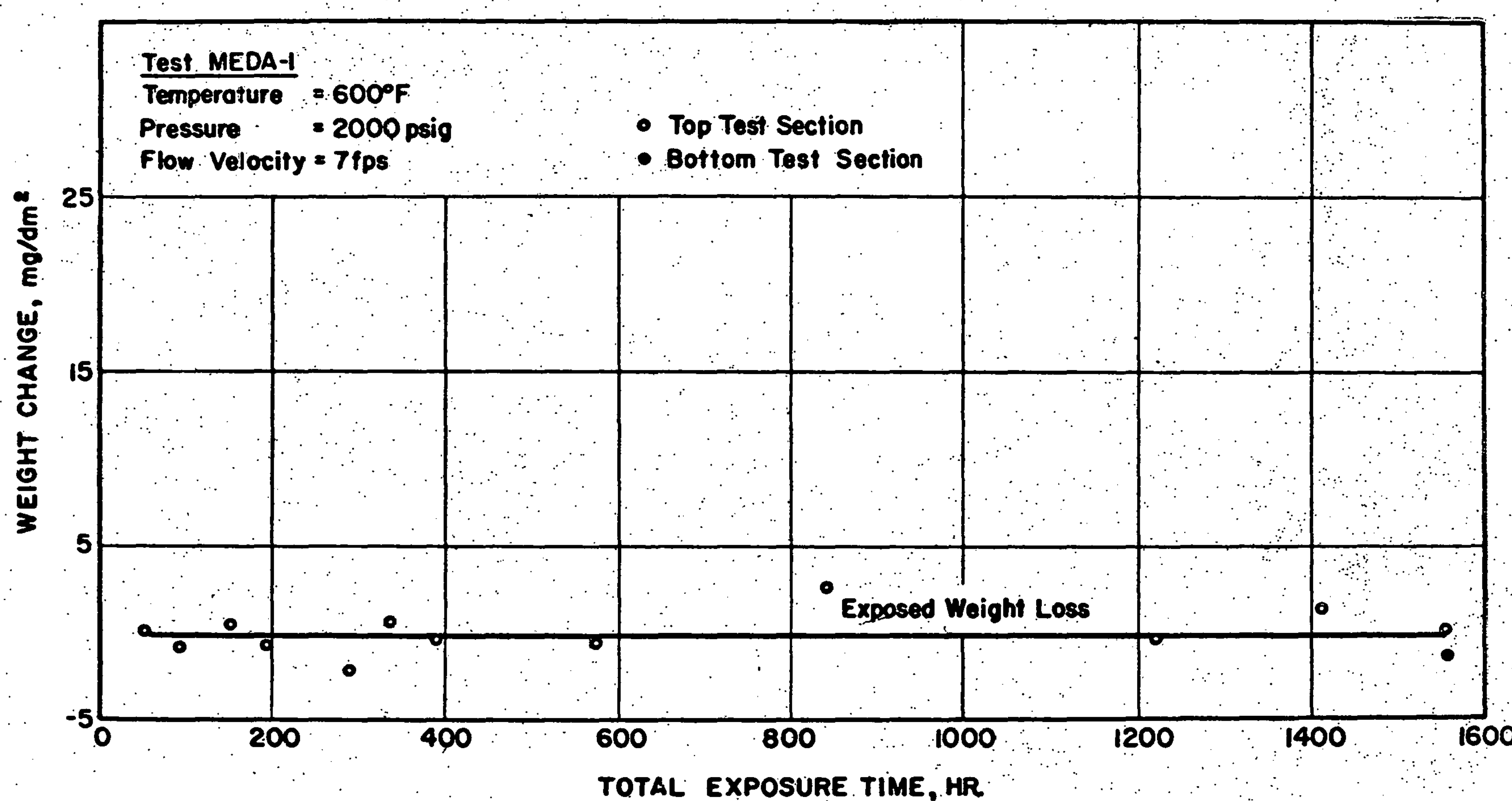
The series of nuclear poison storage tests has been completed. The draft of a topical report has been written, entitled "Studies of the Corrosion of AISI 304 Stainless Steel and AISI 4135 Carbon Steel Exposed to Saturated Solutions of Boric Acid". The results indicate that 18% chromium - 8% nickel stainless steel has adequate corrosion resistance for containing saturated boric acid solution at the contemplated storage temperature (150°F.). Carbon steel possibly could be used for storage of saturated boric acid solution at room temperature if the solution were neutralized, however, its value as a material for this application is marginal.

3.3 Corrosion of Materials of Construction

D. D. Whyte A. Krieg

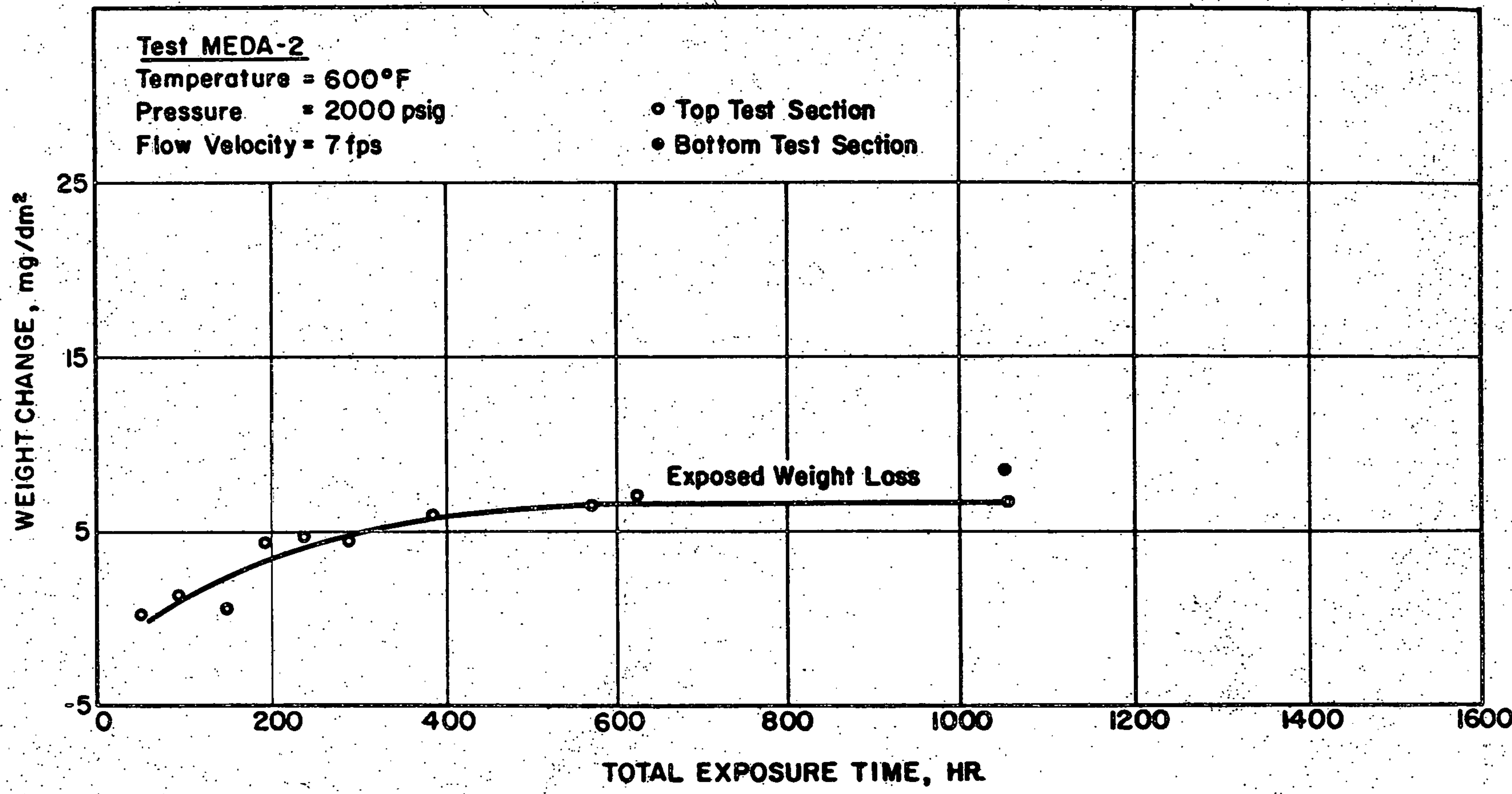
Corrosion of AISI 304 Stainless Steel in High Temperature Boronated Water

Results of corrosion tests of AISI 304 stainless steel in high temperature boronated water are shown in Figures 17, 18, 19, and 20. Exposed weight changes only (difference between original weight and weight after exposure) are shown. Descaled weight changes (difference between the original weight and the weight after scale removal) are being re-evaluated due to failure of the electrolytic descaling procedure to remove completely the type of scale produced in these tests.



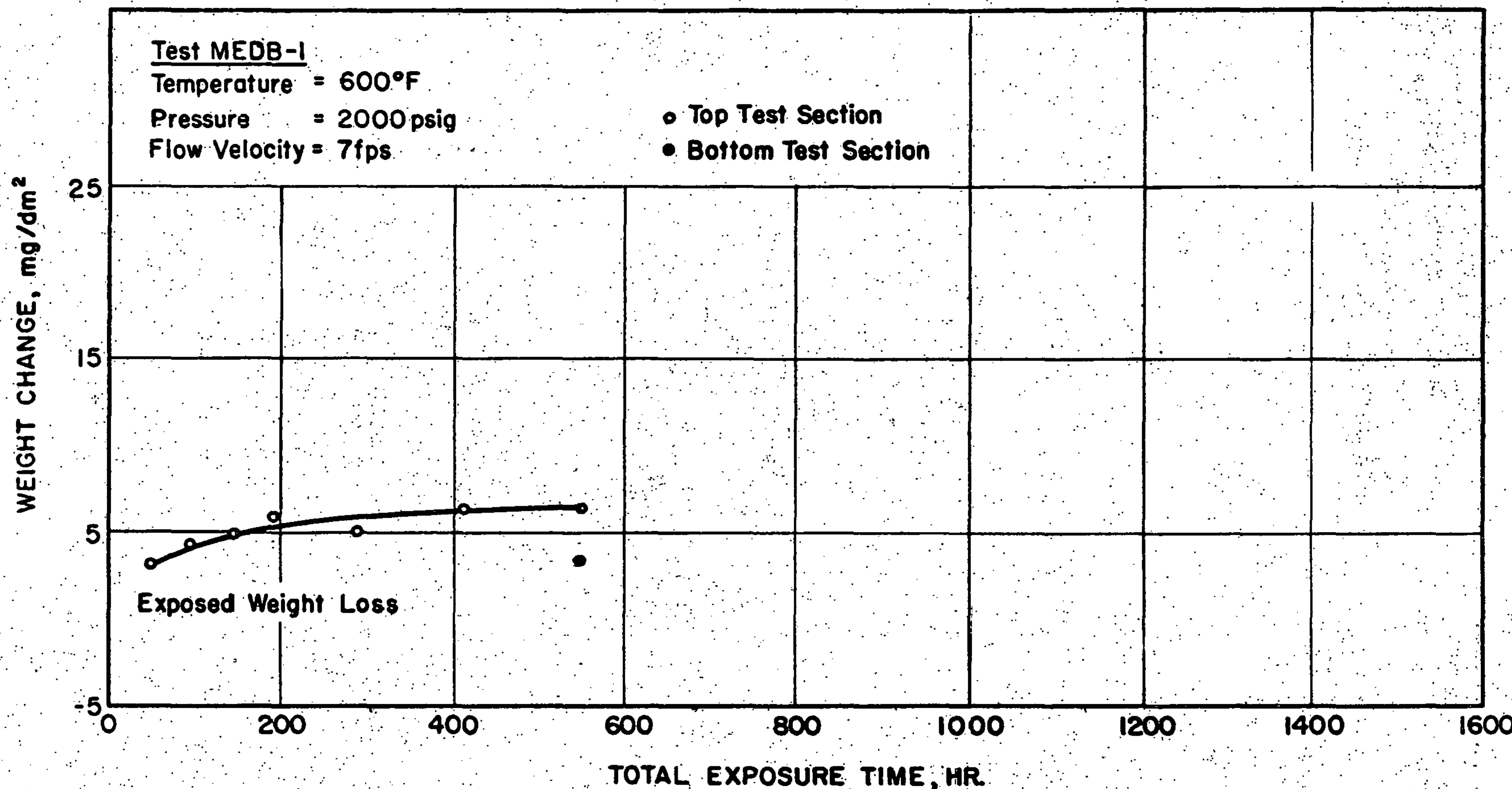
EXPOSED WEIGHT LOSS AS A FUNCTION OF EXPOSURE TIME OF 304 STAINLESS STEEL IN WATER CONTAINING 39 PARTS PER MILLION BORON AT pH = 9.8

FIGURE 17



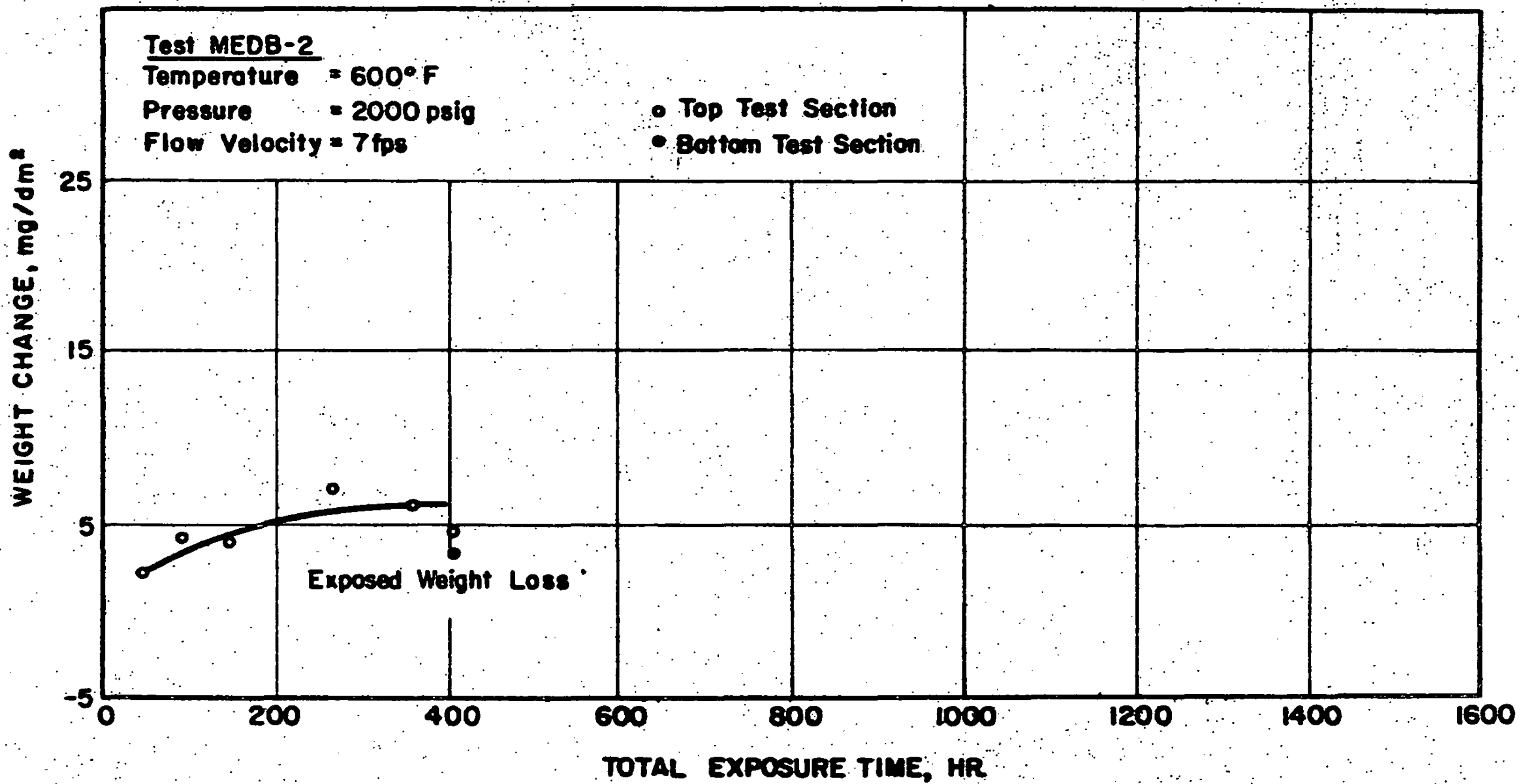
EXPOSED WEIGHT LOSS AS A FUNCTION OF EXPOSURE TIME OF 304 STAINLESS STEEL IN WATER CONTAINING 41 PARTS PER MILLION BORON AT pH = 6.9

FIGURE 18



EXPOSED WEIGHT LOSS AS A FUNCTION OF EXPOSURE TIME OF 304 STAINLESS STEEL IN WATER CONTAINING 1587 PARTS PER MILLION BORON AT ph= 9.7

FIGURE 19



EXPOSED WEIGHT LOSS AS A FUNCTION OF EXPOSURE TIME OF 304 STAINLESS STEEL IN WATER CONTAINING 1660 PARTS PER MILLION BORON AT pH = 9.9

FIGURE 20

The effect of the following aqueous environments on corrosion behavior of AISI 304 stainless steel were investigated at 600°F. and at a flow velocity of 7 feet per second:

Aqueous Environment

B, ppm	pH	Test No.
39	9.8 (with LiOH)	MEDA-1
41	6.9 (no additive)	MEDA-2
1587	9.7 (with LiOH)	MEDB-1
1660	9.9 (with LiOH)	MEDB-2

The following conclusions were obtained:

1. The exposed weight change at low boron concentration with high pH (Test MEDA-1) appeared to be independent of time after the first two days. The data are being re-evaluated.

2. The exposed weight changes as a function of time at low boron concentration with no pH adjustment (Test MEDA-2) and at high boron concentration with high pH (Tests MEDB-1 and MEDB-2) are expressed by a regression type equation of the form $Y = A + B \log (t + 1)$ where Y is the weight loss per unit area (mg/dm^2), t is the exposure time in hours and A and B are constants determined from the data by a least squares procedure. Based on this equation, the following rates of exposed weight change are derived for the periods from two days (48 hours) to 32 days (768 hours) of exposure. In the case of Tests MEDB-1 and MEDB-2, this involves cautionary extrapolation beyond the duration of the test.

Test No.	Duration, hr.	Rate of Exposed Weight Change mg/dm^2/30 days
MEDA-2	1056	0.23
MEDB-1	554	0.14
MEDB-2	418	0.17

3. No tendency towards stress corrosion cracking exists under any of the conditions tested as determined in U-bend specimens exposed during the test.

Qualitative Evaluation of Stainless Steel Tube Bundles Exposed in High Temperature Boronated Water

Two tube bundles (Figure 21) constructed of sensitized stainless steel (one bundle of AISI 316 and the other of AISI 304) with welded end-plugs and Nicrobraz 50 brazing alloy brazed tube-ferrule joints were exposed to water containing boric acid equivalent to approximately 40 parts per

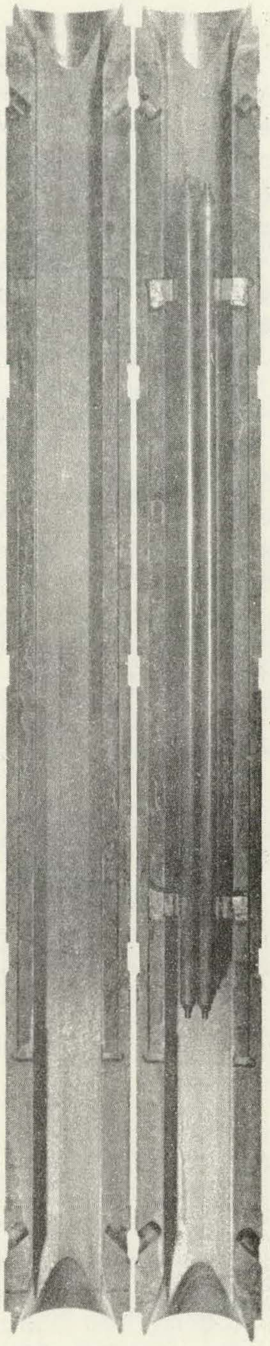


Figure 21 Specimen Holder with Tube Bundle

million of boron at 600 °F., 1850 psig and at a flow velocity of 36-42 feet per second. The 316 bundle was exposed for 321 hours. The AISI 304 stainless steel bundle was exposed for 708 hours. All parts of the two bundles showed good resistance to corrosion except for the occurrence of pitting at one tube end in the 316 bundle which had cracked and bulged during heat treatment prior to exposure. No tendency for accelerated corrosive attack was found at unbrazed tube-ferrule contact surfaces. Fine cracks in the braze fillets, produced during furnace cooling prior to exposure, did not propagate into diffusion bonded regions. Tube bundle integrity was maintained by the joints having diffusion bonds despite cracking in the braze fillets.

Dynamic Screening Tests of Yankee Materials of Construction

A series of three tests of proposed materials of construction has been completed in water containing various levels of boron concentration, (as boric acid) without pH adjustment. In test No. 4, Yankee materials of construction were screened for a total of 759 hours in which the boron concentration was periodically cycled (180 hour periods) from 1600 ppm boron to < 5 ppm boron. These tests were performed under the following average conditions:

	Test No. 3	Test No. 4	Test No. 5
Test duration, hr.	715	759	707
Temp., °F.	598	600	600
Pressure, psig	1792	1823	1831
Flow velocity, fps	39.4	38.7	37.7
H_2, ml (STP) per kg. solution	27	25-35	31.2
pH (maintained by boric acid)	5.0-5.8	cycled	6.1
B (as boric acid), ppm	1590	2 cycles	3
Rate of cycling held for approximately 180 hours at upper and lower limits		(1600 to < 5)	

Fresh specimens were exposed in each test except for two stainless steel specimens which are described below. The corrosion test results are summarized in Table VIII. Within the limits of accuracy of the test, there appears to be a slight decrease in the extent of attack on AISI 304 stainless steel during the cyclic test (Test No. 4) with essentially no difference (except in the case of the heat sensitized material) between the high boron concentration (Test No. 3) and the low boron concentration (Test No. 5). The Ag-In-Cd alloy shows an anomalous increase in weight at the low boron concentration (Test No. 5). Inconel shows a decrease in extent of corrosion with decreasing boron concentration. AISI 410 stainless steel shows a slight increase in attack in going from the high boron concentration to the cyclic boron condition with a subsequent decrease at the low boron concentration. Zircaloy-2 behaves in an inverse manner compared to AISI 410 stainless steel.

TABLE VIII

RESULTS OF SCREENING CORROSION TESTS OF YANKEE MATERIALS FOR ONE MONTH IN HIGH TEMPERATURE BORONATED WATER WITHOUT pH ADJUSTMENT

Material	Test No. 3				Test No. 4				Test No. 5			
		Aver. Wt. Change, mg/dm^2				Aver. Wt. Change, mg/dm^2				Aver. Wt. Change, mg/dm^2		
	No. of Specimens	Exposed	Descaled Electrolytic	Descaled Chemical*	No. of Specimens	Exposed	Descaled Electrolytic	Descaled Chemical*	No. of Specimens	Exposed	Descaled Electrolytic	Descaled Chemical*
304 S/S												
quenched	2	-21.4	-28.6	-46.7	1	-13.1	-22.6	-33.3	1	-16.9	-23.0	-46.7
heat sensitized	3	-17.3	-24.7	-34.3	2	-12.5	-20.8	-33.3	2	-20.1	-23.8	-52.3
as-received	0	-	-	-	2	-13.7	-20.2	-32.7	1	-21.7	-32.8	-62.2
as-received and polished		-	-	-	2	-11.3	-18.4	-33.6	2	-13.8	-20.0	-42.2
weld coated	1	-14.9	-19.6	-40.2	1	-10.1	-16.1	-27.0	1	-14.9	-19.6	-40.2
Nicrobraz 50	1	-31.5	-39.9	-55.9	0	-	-	-	1	-14.9	-21.8	?
Ag-In-Cd	3	+56.5	-	-	3	+22.6	-	-	3	+154.5	-	-
410 S/S	2	-65.5	-100.3	-121.4	2	-75.0	-160.1?	-142.7	2	-31.8	-83.7	-84.6
Inconel	2	-77.4	-79.8	-140.9	2	-41.1	-47.6	-80.9	2	-13.4	-16.1	-42.6
Zr-2	2	+16.7	+15.8	+15.2	2	+13.7	+13.1	-	2	+24.2	+24.2	-
Hafnium	2	+ 4.2	+ 2.8	+ 2.4	0	-	-	-	0	-	-	-

* By alkaline permangante-citrate method. A blank correction of 3.3 mg/dm^2 has been made for attack on the base metal for stainless steel 304. The Nicrobraz-50 blank correction is 56 mg/dm^2 and for Inconel is 1.6 mg/dm^2. For 410 stainless steel the correction is 34.3 mg/dm^2.

Two AISI 304 stainless steel specimens were re-exposed after Test No. 3 in Tests No. 4 and 5 with exposed weight change measurements made between tests and final exposed and descaled measurements made after Test No. 5. The following results were obtained:

	AISI 304 SS, quenched		AISI 304 SS, Heat Sensitized	
	Exposed Wt. Change mg/dm^2	Chemically Descaled Wt. Change mg/dm^2	Exposed Wt. Change mg/dm^2	Chemically Descaled Wt. Change mg/dm^2
After Test No. 3	- 22.6	--	- 16.8	--
After Test No. 4	- 1.2	--	- 1.2	--
After Test No. 5	+ 0.2	- 62.3	+ 0.1	- 52.3

These results indicate essentially no change in exposed weight after the first month of exposure. The descaled weight change (corrected for attack of the descaling solution on the base metal) is considerably lower than the summation of the attack suffered during each of the individual tests on similar specimens and is, in fact, only slightly greater than the extent of attack experienced in Test No. 5 for quenched material and with no change for the heat sensitized material.

Stainless steel AISI 304 foils (3 mils thick) were exposed in Test No. 3, 4, and 5 under stressed conditions varying from 25% to 100% of the yield point at operating temperature (calculated values) in the fixture shown in Figure 22. The stressed specimens were exposed in a stagnant location at the inlet to the loop pressurizer. The foils exposed in Tests No. 2 and 4 had been heat sensitized prior to test exposure (1850°F. for one hour followed by furnace cooling). The foil exposed in Test No. 5 showed no sensitization when examined after test exposure. No evidence of stress corrosion cracking was observed in any case.

A series of tests similar to Test No. 3, 4, and 5 has been initiated in which 1 ppm lithium (as lithium hydroxide) will be added to increase the pH value. Test No. 6, which corresponds to Test No. 3, has been completed under the following average conditions:

Test Duration, hr.	714
Temperature, °F.	600
Pressure, psig	1828
Flow velocity, fps	38.8
H_2, ml (STP) per kg solution	27.1
B (as boric acid), ppm	1561
Li (as lithium hydroxide), ppm	0.98
pH	6.3

The corrosion test results for Test No. 6 are summarized in Table IX. Comparison of these results with those of Test No. 3 shows a marked reduction in extent of corrosion in hot boric acid solutions for all materials when lithium hydroxide is present, even at concentrations

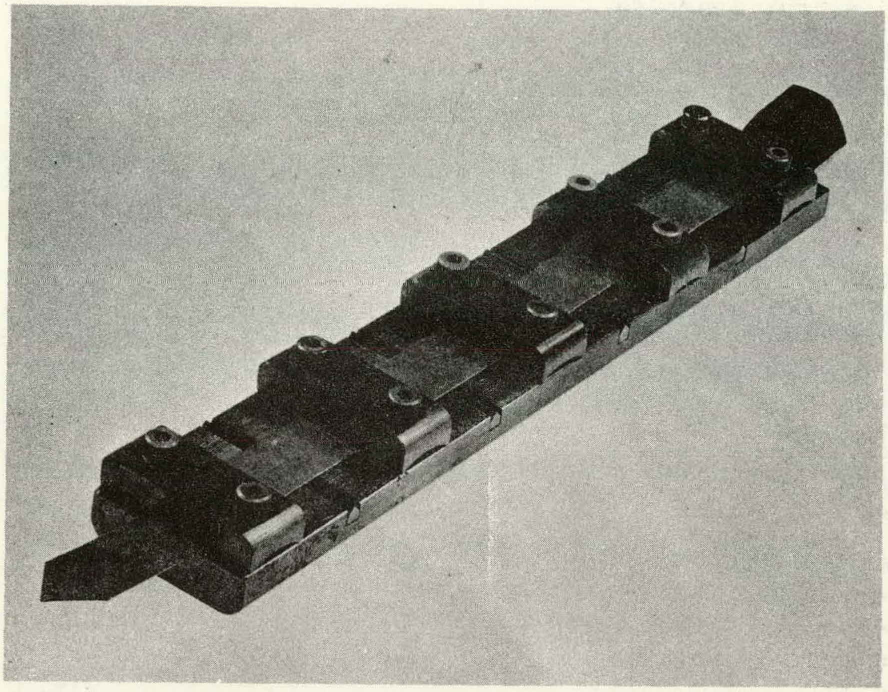


Figure 22 Stressed Corrosion Specimen Fixture

A single 3 mil thick specimen is distorted to levels varying from 9 mils to 36 mils in four stages across a span, at each stage, of one inch.

of lithium of less than 1 ppm. This is true in all cases except Zircaloy-2 which is essentially unchanged and Ag-In-Cd alloy which shows a slight increase.

TABLE IX

RESULTS OF SCREENING CORROSION TEST OF YANKEE MATERIALS FOR ONE MONTH IN HIGH TEMPERATURE BORONATED WATER WITH pH ADJUSTMENT

Test No. 6

Material	No. of Specimens	Aver. Wt. Change mg/dm^2 Exposed	Descaled Electrolytic	Chemical*
AISI 304 S/S				
Quenched	1	-10.4	-15.0	-23.6
Heat sensitized	2	- 8.4	-14.4	-23.4
As-received	0	-	-	-
As-received and polished	1	- 7.1	-15.4	-22.7
Weld coated	1	- 8.9	-15.3	-26.0
With Nicrobraz 50	1	-13.4	-18.3	?
Ag-In-Cd	3	+70.8	-	-
AISI 410 S/S	2	-33.4	-63.8	-62.6
Inconel	2	-27.2	-31.1	-70.6
Inconel-X	2	-24.5	-29.3	-71.4
Zr-2	2	+19.1	+19.1	-

* The blank corrections are shown in the footnote to Table VIII.

Testing of pH Control Agents in Static Autoclaves

Static corrosion studies on proposed materials of construction undertaken to determine a pH control agent have progressed to 600 hours. Two tests are being conducted simultaneously under the following conditions:

Flow	Static
Pressure	1525 psig
Temperature	600°F.
Water Chemistry	1600 ppm B (added as H_3BO_3) pH - 5.8 ± 0.2 (controlled with LiOH and KOH) 30 cc H_2/Kg H_2O (STP)

The test results to date are summarized in Table X.

TABLE X

RESULTS OF SCREENING pH CONTROL AGENTS

Material	No. of Spec.	LiOH Ave. Exposed Wt. Change (mg/dm^2) 24 hr.	96 hr.	264 hr.	600 hr.	KOH Avg. Exposed Wt. Change (mg/dm^2) 24 hr.	96 hr.	264 hr.	600 hr.
AISI 304 S/S									
Quenched	2	-0.4	0	0	-0.8	0	-0.6	-1.0	-0.6
Heat Sensitized	3	-0.8	0	-0.4	-0.5	0	-2.2	-2.6	-2.0
Weld coated	1	0	-0.4	0	-0.4	-0.4	0.4	0	0.8
With Nicrobraz-50	1	0.8	1.6	2.0	2.4	0	-1.2	-0.4	-0.4
As-received	0	-	-	-	-	0			
Ag-In-Cd	3	3.2	6.3	10.1	12.1	16.2 *	44.4 *	73.4 *	86.0 *
AISI 410 S/S	2	-4.8	-5.4	-5.8	-7.2	-0.6	0	-1.0	-2.0
Inconel	2	-1.4	-0.4	-0.2	1.6	0	0	0.6	0.8
Zircaloy-2	2	6.4	9.2	11.8	14.0	6.4	9.4	11.2	14.2

* Two specimens only.

The corrosion rate of Ag-In-Cd alloy appears to be greater in potassium hydroxide than in lithium hydroxide. Corrosion of all other materials was slight and little difference was observed for corrosion in LiOH or KOH. These tests however, are still in progress.

Dynamic Tests of Ag-In-Cd Alloy Material

The effect of lithium hydroxide and potassium hydroxide on corrosion behavior of Ag-In-Cd alloy control rod material is being investigated at 600°F. under semidynamic conditions (0.1 fps):

AQUEOUS ENVIRONMENT

Test No.	B, ppm	pH	H_2
DA-1	1600	5.8 with LiOH	30 cc/Kg H_2O (STP)
DA-2	1600	5.8 with KOH	30 cc/Kg H_2O (STP)

To date, results indicate that the exposed weight gain obtained in either test (DA-1, DA-2) at the end of 524 hours was approximately 35 mg/dm^2 and that the exposed weight changes as a straight line function of time after the initial three-day tests at a corrosion rate of 1.2 to 1.7 mg/dm^2/day. Further tests of this material are proceeding.

Static Autoclave Tests on Ag-In-Cd Control Rod Material

A three-week test was made to confirm previous data on sloughing of Ag-In-Cd alloy. The conditions of this test were as follows:

Fluid	neutral degassed water
Temperature	650°F.
Pressure	saturated steam pressure (2200 psi)
Flow	static

The results are listed below:

Total Time (hours)	Total Wt. Gain (mg/dm^2)
168	79
322	90
490	78

From these results, it appears that some sloughing apparently occurs after approximately two weeks of testing under the stated conditions.

Brazed Fuel Assembly Cyclic Test

A brazed fuel assembly, minus fuel, was cycled for 14 days with eight hour on - 16 hour off cycling and an additional 14 days with 16 hour on - 8 hour off cycling in a dynamic autoclave at 600°F. from room temperature, saturation pressure, in neutral degassed water.

No adverse effects were observed from the above treatment either on the brazed joints or the stainless steel tubes themselves other than normal corrosion (< 5 mg/dm^2/mo).

3.4 Interactions Between Chemical Absorbers, Corrosion Products and Fission Products

R. F. Sterling C. C. Thomas

During the second quarter of 1958, the Van de Graaff crud deposition loop was placed into operation. An exploratory run using AISI 410 stainless steel wool was completed. The run was performed at a neutral pH, less than 0.1 ppm dissolved oxygen, 601°F., with an AISI 304 test specimen and without radiation for a period of 99.61 hours. The total amount of iron released to the system by 190.6 dm^2 of stainless steel wool was less than 2 mgs. The descaled specimen weight loss was of the order expected from corrosion data. This data indicated that the stainless steel crud source did not release sufficient crud to permit accurate analysis. On this basis, a change to fine carbon steel wool as a crud source was made.

A series of runs have been completed and are as follows:

1. Control runs with AISI 304 test specimens at pH 7 and 10.

2. Irradiation runs with 304 test specimens at pH 7 and 10.

Complete analytical data are available for the irradiation run at pH 7 and is presented in Table XI. Samples from the other runs have been submitted for analysis. Sufficient data are not yet available to permit any valid conclusions to be drawn.

Figures 23a and 23b show the neutral pH irradiation and control run specimens before electrolytic descaling.

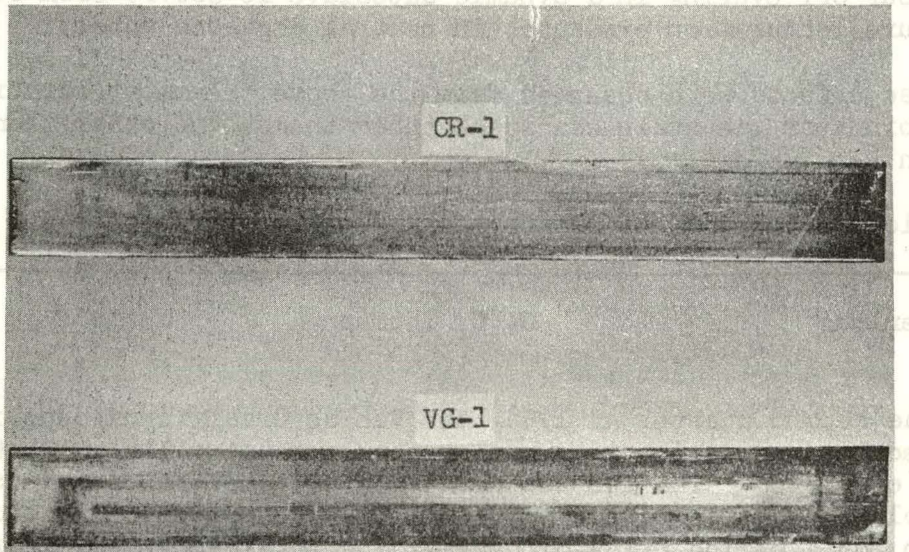


Figure 23a Stainless Steel Crud Deposition Specimens - Water Inlet Side
Top - Control Run - Inlet to Left
Bottom - Irradiation Run - Inlet to Right

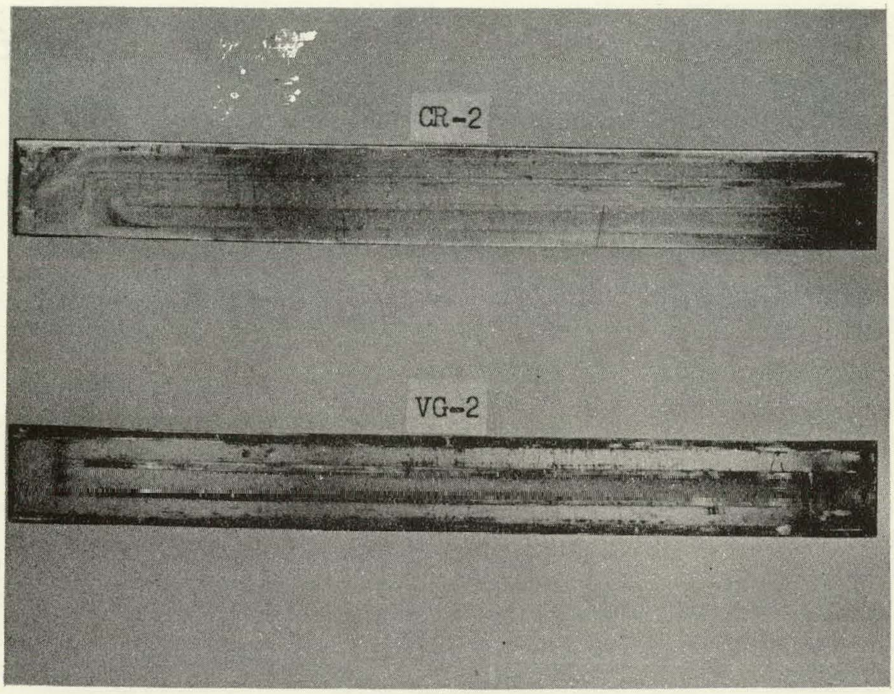


Figure 23b Stainless Steel Crud Deposition
Specimens - Water Outlet Side

Top - Control Run - Outlet to Left
Bottom - Irradiation Run - Outlet
to Right

TABLE XI

CARBON STEEL CRUD DEPOSITION ON AISI 304 STAINLESS STEEL AT NEUTRAL pH[a]

	VG-1 Test	CR-2 Test
Operating Time, hr.,		
a. Beam Time	100.5	--
b. Time at Temperature	111	111.5
Beam Current Microamperes	125.1	--
Average Temperature °F.	613	612
Loop Water Chemistry		
a. Maximum pH	7.1	7.2
b. Oxygen Content	0.1 ppm	0.1 ppm
c. Maximum Conductivity, micromhos	0.1	0.1
Average Test Section Velocity, fps	7.5	7.8
Crud Source Area, dm^2	508	498.6
Analytical Data		
Iron Found on Filter, mg	4.236	-- *
Iron Found on Anion Resin, mg	0.229	-- *
Iron Found on Cation Resin, mg	3.023	-- *
Iron Found on Window, mg	.098	negligible
Weight Before Descaling, gm	2.7710	2.7862
Descaled Weight Loss of Specimen, mg	1.000[b]	0.300[b]
Total Iron Found Attributible to Crud, mg	7.586	--
Percentage Deposition		
a. On Specimen	negligible	negligible
b. On Window	1.29	negligible

a. Loop clean up carried out with H-OH resin.

b. Descaled weight losses of 5 mg/dm^2 would be expected for corrosion alone for these test conditions. The test specimen area was 0.28 dm2 or an expected descaled weight loss of 1.4 mgs. Thus the crud deposition was deemed negligible.

* Analysis was not completed.

3.5 Decontamination and Waste Disposal Studies

R. M. Watkins

The draft of a topical report entitled, "A Study of Decontamination Agents for Use in the Yankee Reactor" was completed. This covers the work done during the first phase of the Yankee decontamination project leading up to the selection of the basic permanganate-citrate procedure as the method offering the most promise for successful solution of Yankee primary loop contamination problems. This report includes a discussion of the necessity for chemical decontamination, the properties sought in an ideal cleaning solution, a survey of available decontamination information and a description of the experimental test program carried out at the Westinghouse Atomic Power Department in an effort to develop new and improved methods for the removal of oxide scale and radioactive contamination from primary coolant system components.

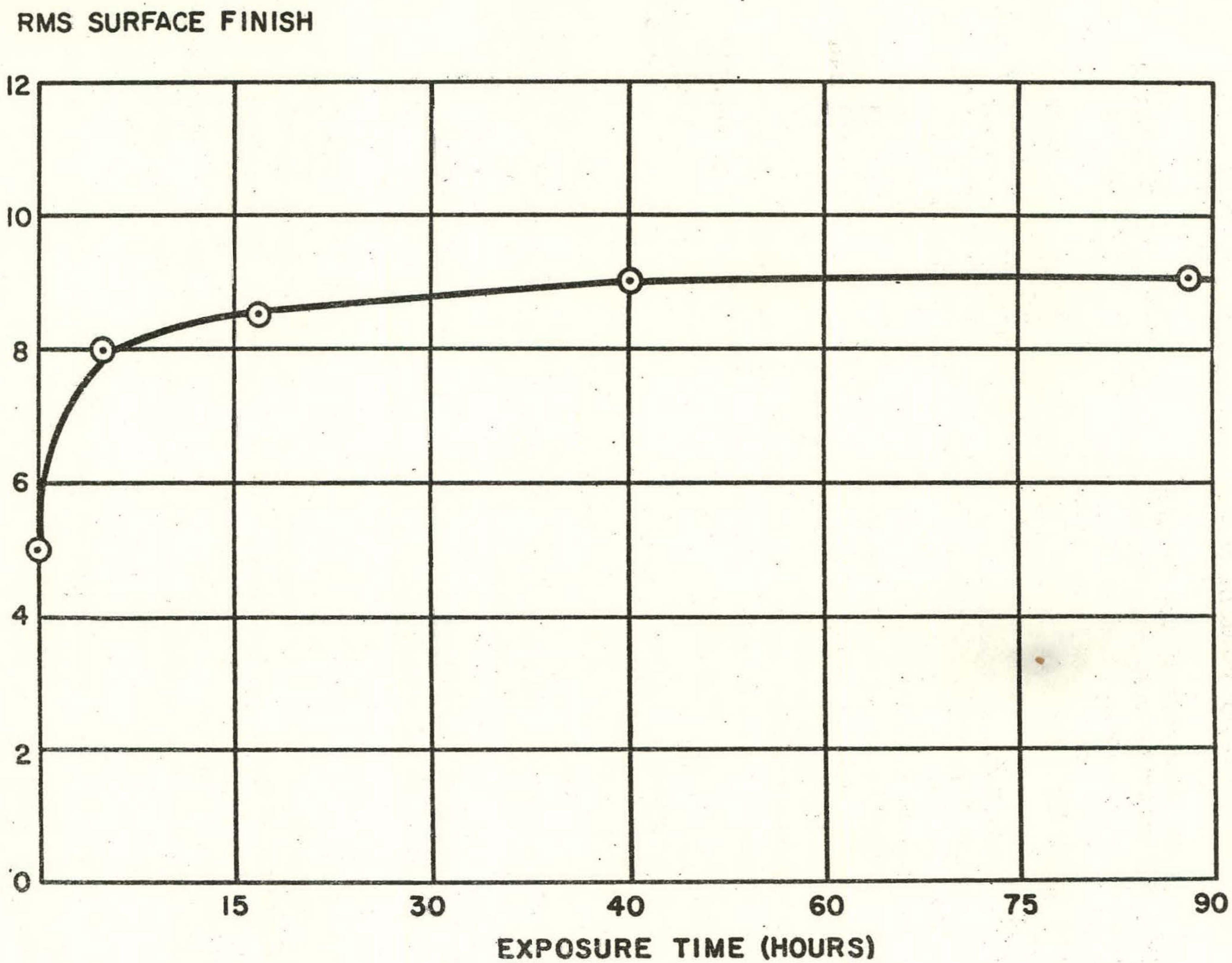
Data are presented concerning the attack rates produced by the basic permanganate and citrate reagents on various reactor materials of construction as is an evaluation of the possibility of caustic embrittlement of stainless steel resulting from the use of the basic permanganate solution. It was concluded that when properly applied, this procedure could provide a safe and efficient method of descaling the stainless steel primary coolant piping. The deposited crud film should actually be dissolved by the treatment, thus bringing the activity within it into solution or suspension. On the basis of the results of the bench scale tests it is anticipated that an overall decontamination factor of at least 10 (i.e., 90% activity removal) can be achieved through the use of this procedure.

The results of fatigue tests, cold bend tests, and metallographic examinations indicate that no embrittlement problem exists as a result of the use of the basic permanganate reagent and that it may be safely employed in cleaning AISI 300 series stainless steel without significant danger of caustic cracking of the metal.

In an attempt to gain experience with the application of the basic permanganate-citrate decontamination procedure in situations more closely approaching actual reactor application, a dynamic loop clean-up trial was conducted using the Westinghouse MED Loop "B". Seven gallon batches of the reagents were handled without difficulty and employed efficiently in the clean-up of this circulating system. A detailed report on the results obtained is being prepared.

The relation of exposure time to change in surface finish during basic permanganate-citrate decontamination of AISI 304 stainless steel was determined. After an initial increase in surface roughness, there is very little change during a prolonged exposure as shown in Figure 24. One of the stainless steel coupons from the surface finish test after 88 hours exposure to the basic permanganate reagent is shown in Figure 25. Note the reflection of the caption on the coupon's surface which has retained its original polished appearance.

A study was made of the relation of exposure time to weight loss in the decontamination of AISI 304 stainless steel using a 2% hydrofluoric acid - 5% ammonium persulfate solution at 140°F. Weight loss was found to rise linearly with time, and was considerably greater than in the case of the basic permanganate-citrate decontamination procedure. After two hours of contact with this reagent, AISI 304 stainless steel sample specimens had suffered weight losses averaging 0.83 mg/cm^2 as shown in Figure 26.



RELATION OF EXPOSURE TIME TO CHANGE IN SURFACE FINISH DURING BASIC PERMANGANATE-CITRATE DECONTAMINATION OF AISI 304 STAINLESS STEEL

FIGURE 24

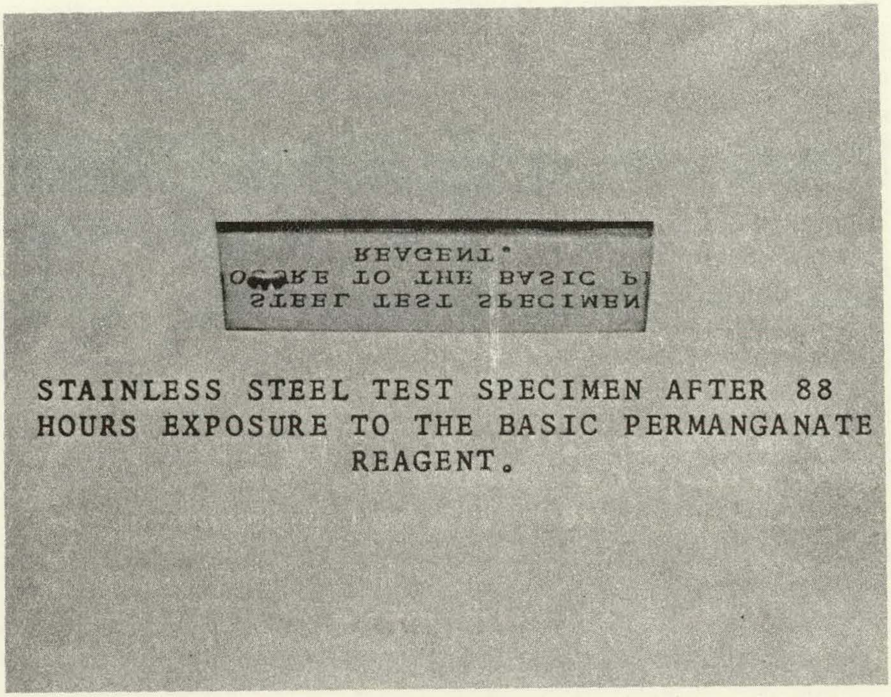
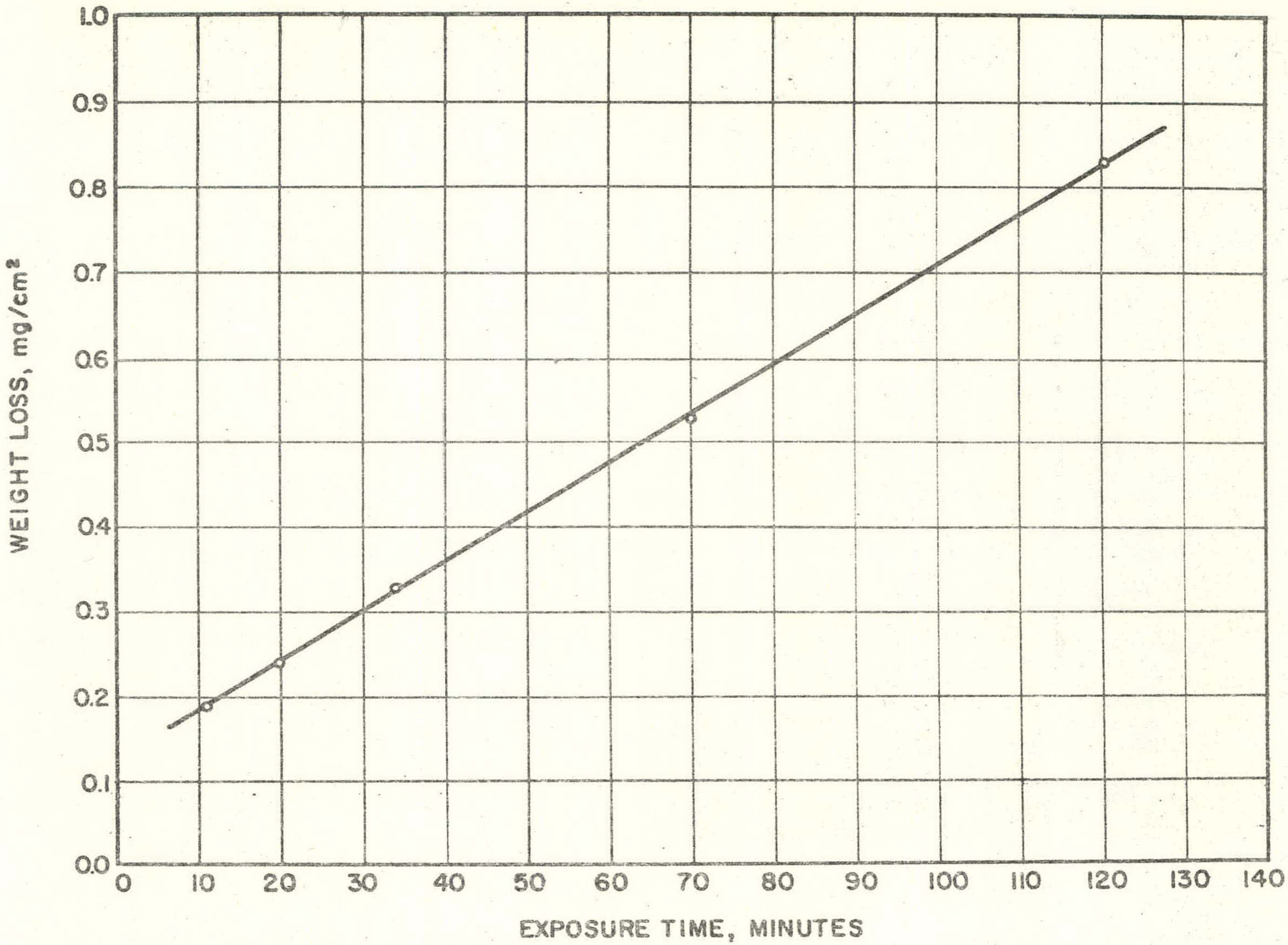

STAINLESS STEEL TEST SPECIMEN AFTER 88
HOURS EXPOSURE TO THE BASIC PERMANGANATE
REAGENT.

Figure 25



RELATION OF EXPOSURE TIME TO WEIGHT LOSS IN PERSULFATE-HYDROFLUORIC DECONTAMINATION OF AISI 304 STAINLESS STEEL

FIGURE 26

Specimens of AISI 304 stainless steel were exposed under the following conditions:

Flow	Static
Temperature	600°F.
Time	14 days
Pressure	1525 psig
Water Chemistry	O_2 - 425 cc/O_2/Kg H_2O
	pH - 8.9 with LiOH

The appearance of the specimens after exposure showed typical interference films. The exposed weight change averaged 40 mg/dm^2/14 days. This work was performed to prepare crudded specimens for subsequent decontamination studies.

In a study of a decontamination procedure suggested by Los Alamos personnel, a group of the crudded specimens prepared in the test referred to above were cycled for 20 hour periods under conditions of high hydrogen followed by high oxygen concentration (200 cc gas per liter H_2O) in a static autoclave at 600°F. and saturated steam pressure. The results indicated that the specimens gained up to 3.2 mg per dm^2 as a result of the treatment. Though this is an extremely simple method, it shows little promise as a feasible decontamination procedure.

3.6 Crud Inhibition, Suspension and Removal

D. G. Sammarone

Work was begun on the design of an experimental assembly which will be used on a high velocity dynamic pressurized loop to study the effects of crud deposition on AISI 304 stainless steel where a heat flux of 30,000 - 50,000 BTU/Hr ft^2 is present. This is being performed to provide a condition to simulate that found in the Yankee steam generator.

The possibility of using forced air as a coolant for crud deposition studies in a pressurized water heat exchanger has been investigated and found to be not feasible because of the large quantities of air which must be circulated. Design calculations are now in progress for a cooling water to pressurized water heat exchanger. Preliminary equipment sketches and flow sheets have been prepared. Work is now proceeding on finalizing the design of the heat exchanger as well as the associated crud generator and purification system. Detailed drawings of the experimental assembly will be issued upon finalizing the design. This assembly is scheduled for installation on the loop in September, 1958.

4.0 MECHANICAL DESIGN

Mechanical and Thermal Section: A. G. Thorp, Manager

This program includes the design and development of mechanical features of fuel assemblies, control rods, support structures, the reactor vessel closure, and fuel handling tools.

4.1 Fuel Assemblies and Control Rod Design

J. R. Reavis G. H. Eng R. Berringer

Control Rod Design

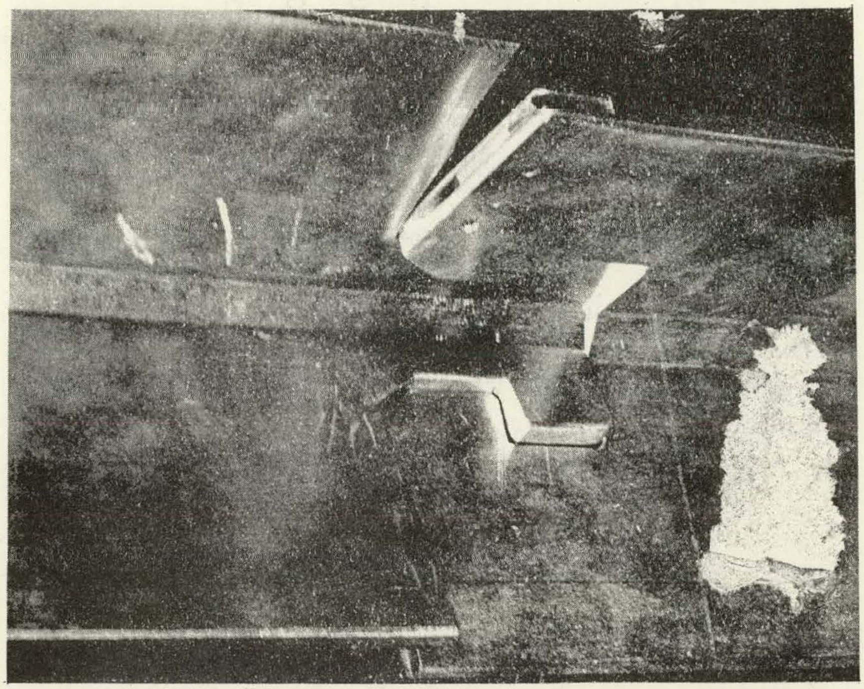
The Yankee control rod adapter and coupling section were redesigned. The Westinghouse APD Model Shop is now making the necessary modifications.

A prototype of a newly designed spring-loaded control rod joint was completed (Figure 27). The locking torque of this model was measured to be 50 foot pounds. Minor improvements are contemplated such as pre-loading the spring and increasing the rigidity of the spring assembly, however, the general design appears to be very satisfactory.

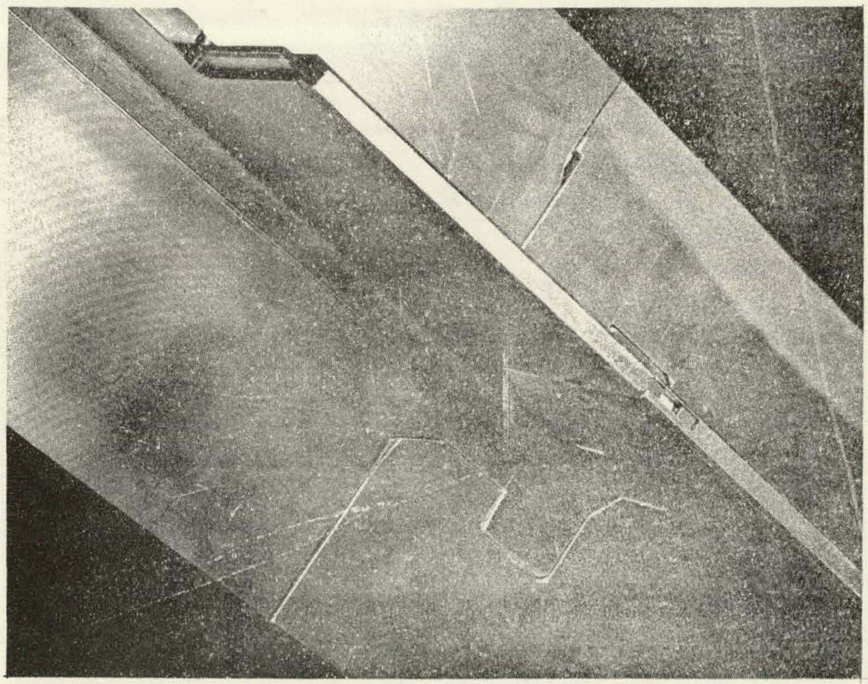
Fuel Assembly Design

During reactor operation a row of fuel tubes on one side of a fuel subassembly may be located in an average temperature coolant channel. The opposite row of tubes in the same subassembly may, at the same time, be located in a hot channel. The hot channel factors impose the limiting conditions on the thermal design and heat transfer analysis of the Yankee reactor. It is reasonable that the mechanical design should also be subject to the same qualifications and limitations. Accordingly, a study was made to determine the effect of a temperature difference from one side to the other side of the fuel subassembly. As shown in Table XII, the calculations indicate that excessive bowing of the fuel assembly might occur during normal operation. Since the nominal clearance between a fuel assembly and adjacent control rod is 0.120 inch, bowing of the assembly could impede movement of the control rod. This condition becomes even more severe when the build-up of mechanical tolerances is considered as shown in Table XIII.

Calculations indicate that, if thermal bowing of a fuel assembly should occur, it would be of such magnitude as to result in binding of the control rods. Redesign is now in process with the objective of preventing such a condition. An End Movement Deflection Test was assembled (Figure 28) in order to determine the flexural rigidity of a fuel subassembly. Information from this test will be used in the analysis of the thermal bowing.



Joint Open



Joint Closed

Figure 27 Yankee Control Rod Joint

TABLE XII

SUMMARY OF BOWING COMPUTATIONS
YANKEE DEFINITIVE FIRST CORE

Analysis No.	Core Type	Type of Traverse	Inner Tubes Located at Radius of	$F_{\Delta T}$ *	F_{θ} *	$F_{F.P.}$ *	Outer Tubes Located at Radius of	ΔT_s - °F	Axial Stress Psi	Load on End lbs.	Avg. Bowing, Inch	Max. Bowing, Inch
Prelim.	Uniform	Diagonal	22.4"	1.0	1.0	1.0	27.4"	8.7	2410	50.2	-	-
1	Uniform	Diagonal	22.4	1.392	1.532	1.0	27.4	29.3	7250	151	-	
2	Uniform	Diagonal	0.5	1.392	1.532	1.0	5.50	28.2	7000	146	-	-
3	Uniform	Center	11.5	1.392	1.532	1.20	15.25	45.9	11400	237	0.123	0.154
4	Two-Reg.	Center	31.5	1.392	1.532	1.20	33.5	38.0	-	-	0.114	0.143
5a	Two-Reg.	Diagonal	18.25	1.392	1.532	1.38	22.50	33.7	-	-	0.072	0.090
5b	Two-Reg.	Diagonal	31.0	1.392	1.532	1.38	33.5	47.4	-	-	0.101	0.127

* Applied to inner tubes at first equivalent radius

TABLE XIII

MECHANICAL TOLERANCE BUILD-UP IN YANKEE REACTOR
FOR REDUCTION OF CLEARANCE BETWEEN
CONTROL ROD AND FUEL ASSEMBLY

1.	Misalignment of Fuel Assembly Bundles with Centerline of End Nozzles	0.034
2.	Bowing or Permissable Distortion in Fuel Tubes Throughout Their Length	0.040
3.	Clearance on Diameter of End Nozzle and Hole in Support Plate	0.008
4.	Center to Center Location of Fuel Assembly and Control Rod	0.006
5.	Radial Tolerance on Control Rod Vane	0.017
	Total	0.105 inches

Nominal Clearance Between Control Rod and Fuel Assembly	0.119
Minus Sum of Mechanical Tolerance Build-Up	-0.105
Clearance Available for Bowing	0.014 inches

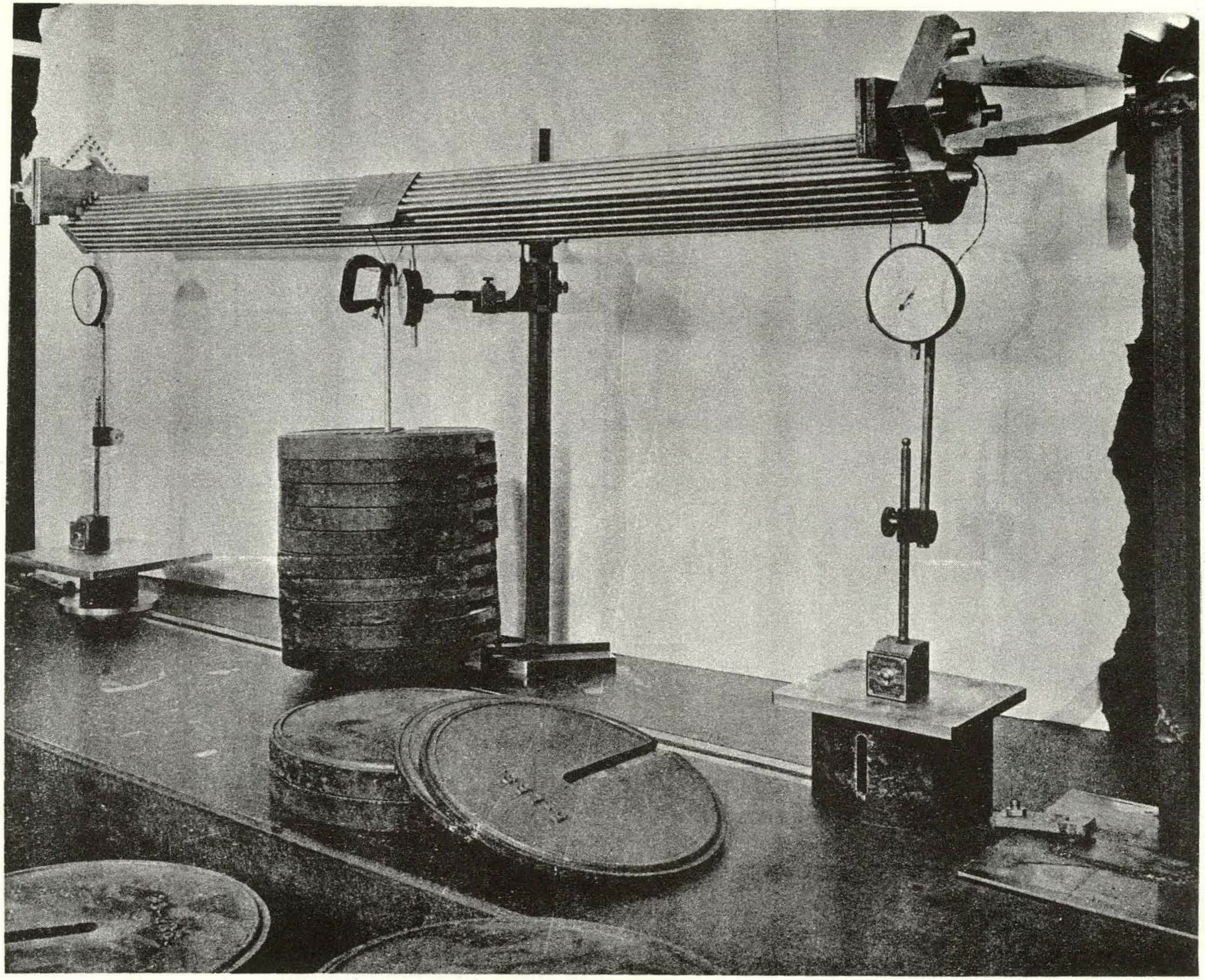


Figure 28 End Moment Deflection Test Assembly

4.2 Control Rod Drive Mechanisms

Cooling Requirement Calculations

The cooling requirements for a Yankee control rod drive mechanism were calculated based upon a reactor coolant water and mechanism coil temperature of 300°F; the reactor vessel temperature was assumed to be 516°F. When the control rod is not moved, the heat generated in the coils is 0.5 KW and the heat due to convection and conduction from the reactor vessel is 0.3 KW. This gives a total cooling requirement of 0.8 KW for a static control rod drive mechanism.

An operating mechanism generates 1.0 KW in the coils and 0.3 KW arises from convection and conduction heat from the reactor vessel. Thus, the cooling requirement for an operating control rod drive mechanism is 1.3 KW.

The total cooling requirements for the 24 control rod mechanisms will be determined by the method of control rod programming. However, for example, if 12 mechanisms are holding the rods and 12 mechanisms are moving the rods, the total cooling load would be $12 \times 0.8 + 12 \times 1.3 = 25.2$ KW.

All fabrication drawings of a positive grip control rod drive mechanism have been released for the purchase of material.

4.3 Design of Core Support Structure and Fuel Handling Tools

Core Support Structure

Several sandwich beam specimens of a geometry similar to an element of the core support structure were tested under various loading and boundary conditions. The main purpose of the tests was to select an optimum design of this type of sandwich structure based on the variable parameters of plate thickness and the thickness and height of the cylindrical spacers. The tests also produced data regarding the flexural and shear rigidity of the structure. The experimental data were compared with the calculated deflection based upon theoretical deflection formulae.

As a result of this study the following parameters for the Yankee core support structure are recommended:

Overall height	8.0 inches
Top plate, thickness	1.25 inches
Bottom plate, thickness	1.5 inches
Cylindrical spacer, thickness	0.5 inches
Circular reinforcing rib, thickness	1.0 inch

Figure 29 shows the measured deflection and the computed slope of the Yankee core support structure model under a total load of 100 pounds. The load was applied by using 76 columns of steel washers to simulate the fuel assemblies in the reactor. The maximum deflection of the model was 0.0081 inch from which the corresponding maximum prototype deflection is predicted to be 0.0365". The corresponding maximum-slope in the prototype is predicted to be 1.504×10^{-3} in/in. This gives a maximum horizontal movement of 0.153 inch at the upper end of the fuel assembly near the upper core support structure.

Electrical resistance strain gage tests on the plastic model indicated a maximum tensile stress of approximately 3600 psi on one side of the circular holes in the support plate at the outer region; a total design static load of 100,000 pounds was assumed.

Tests in the hydraulic loop indicated that the size of the coolant water flow holes in the lower core support structure had a larger effect on the flow distribution in the core than did the internal geometry of the core. It was recommended that the diameter of one water hole located at the center and four holes located 90° apart at a 10.83 inch radial distance from the center be increased from 1.625 to 3.0 inches. Model tests indicated that this modification increased the deflection of the core structure by approximately 10%. The maximum deflection of the prototype was calculated to be 0.0415 inch.

A topical report "Deflection and Stress Analysis of the Yankee Core Support Structure", YAEC-77, is being published.

Fuel Handling Tool

The handling socket in the fuel assembly, the control rod, and the guide tube were redesigned so they might be manipulated by a universal handling tool. These design changes, together with other modifications in existing equipment, are now being incorporated in the control rod channel mock-up being fabricated by the Model Shop. The control rod channel mock-up (Figure 30) will be located in the water filled deep pit. A view of the upper core support plate mock-up is shown in Figure 31.

The design of the universal handling tool (Figure 32) was made, and the detailed drawings are now being completed.

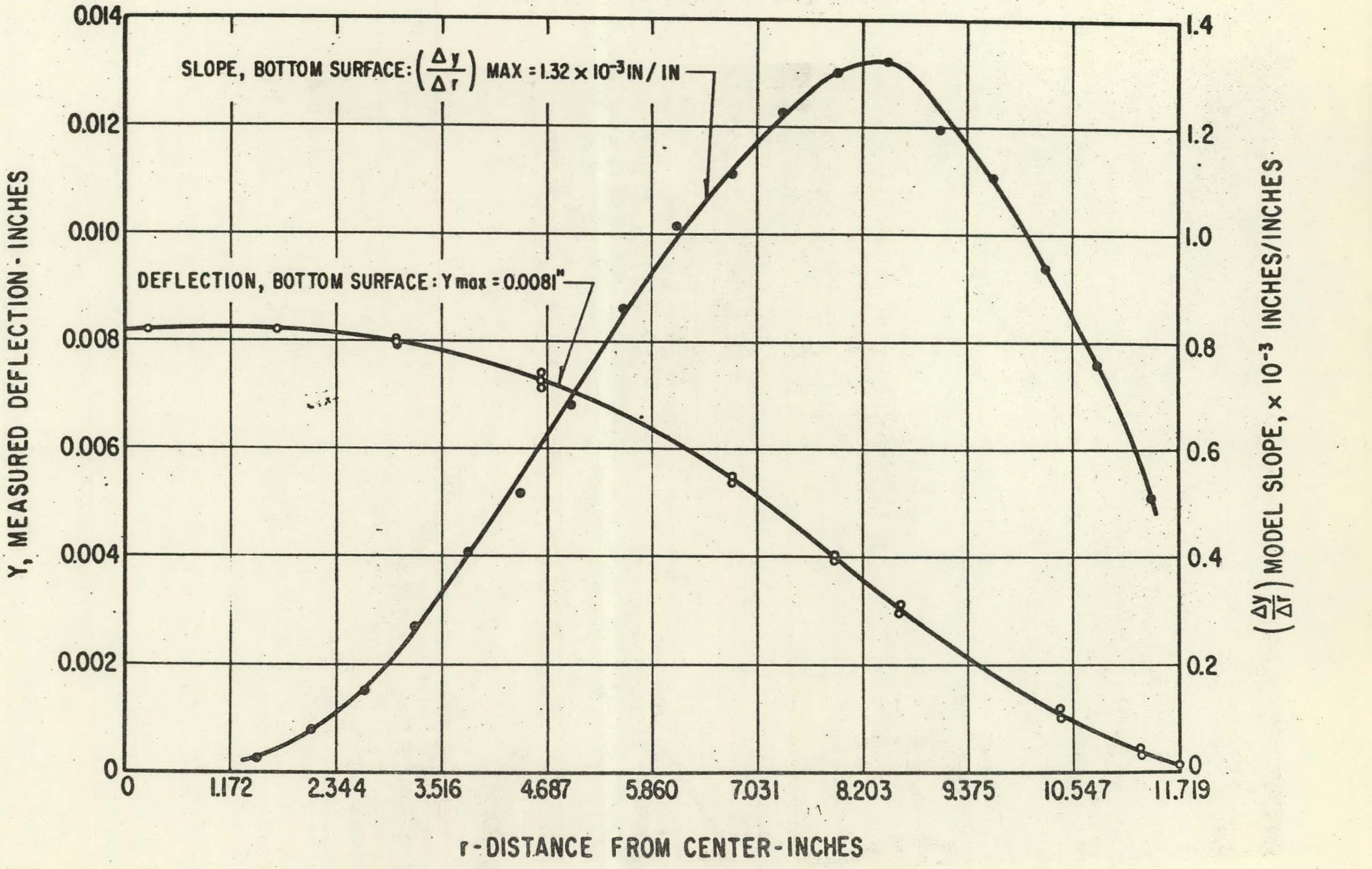


Figure 29 Deflection and Slope Variation with Radial Location for the Yankee Core Support Structure Model

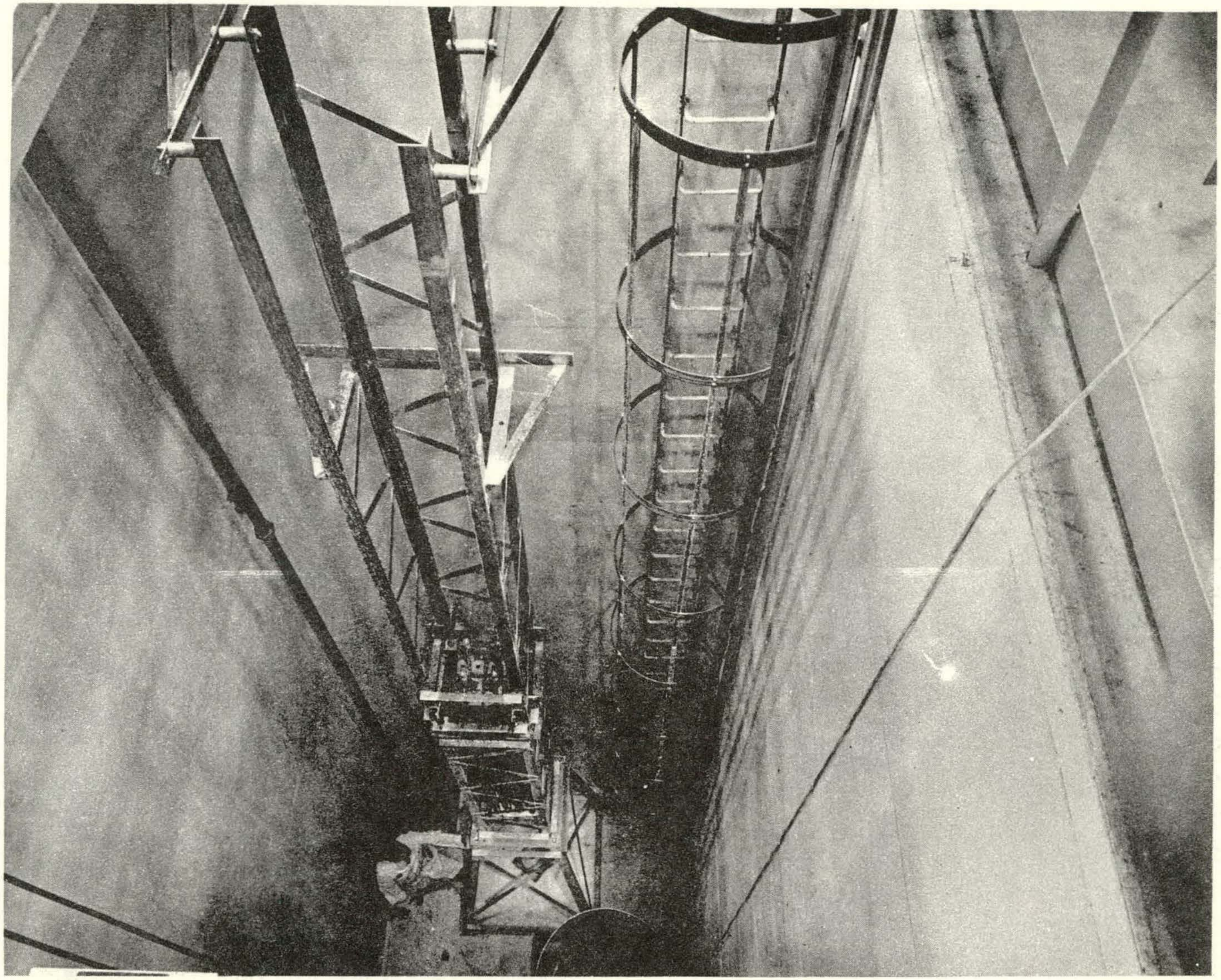


Figure 30 Control Rod Channel Mock-Up Test Assembly

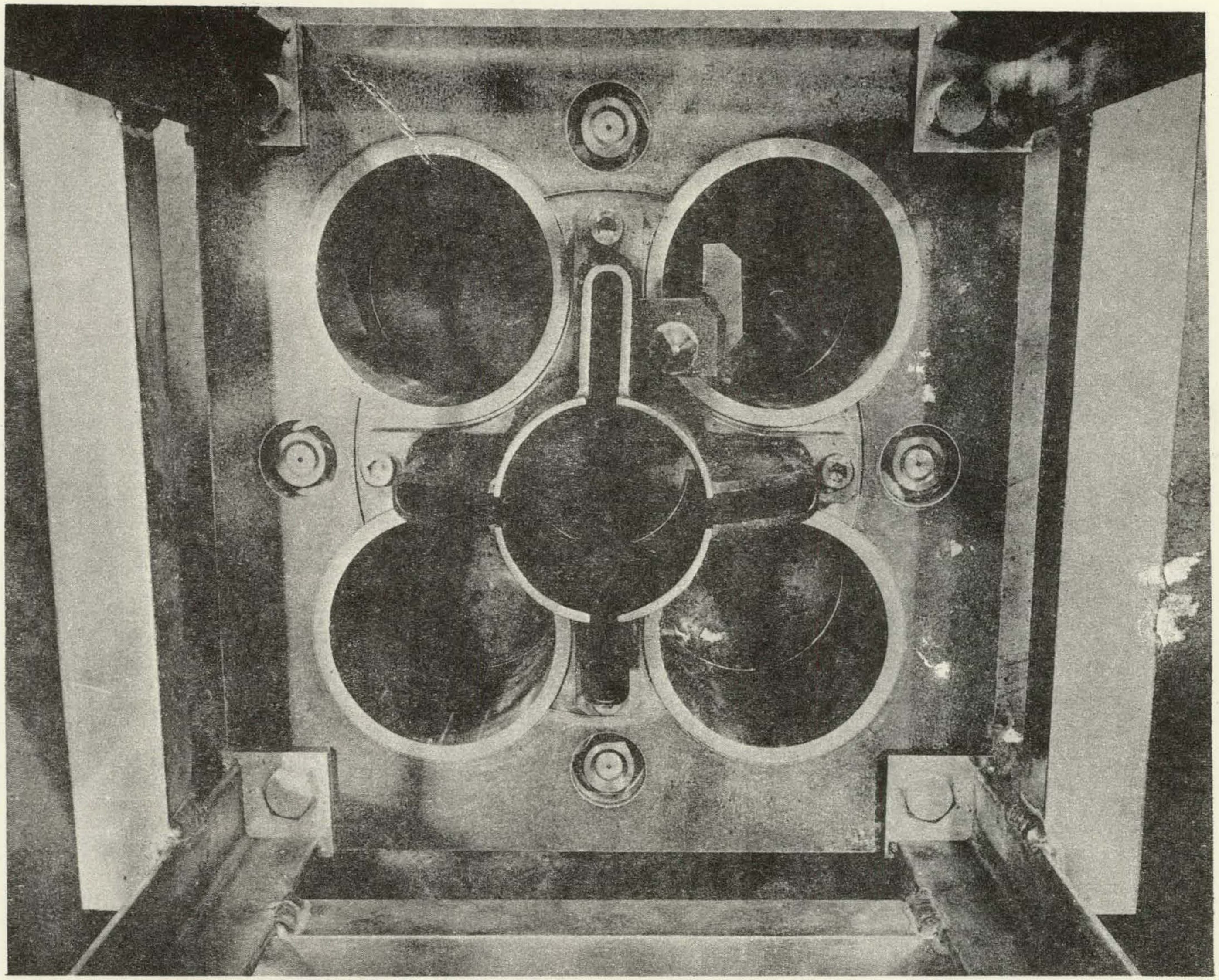


Figure 31 Upper Core Support Plate Mock-Up

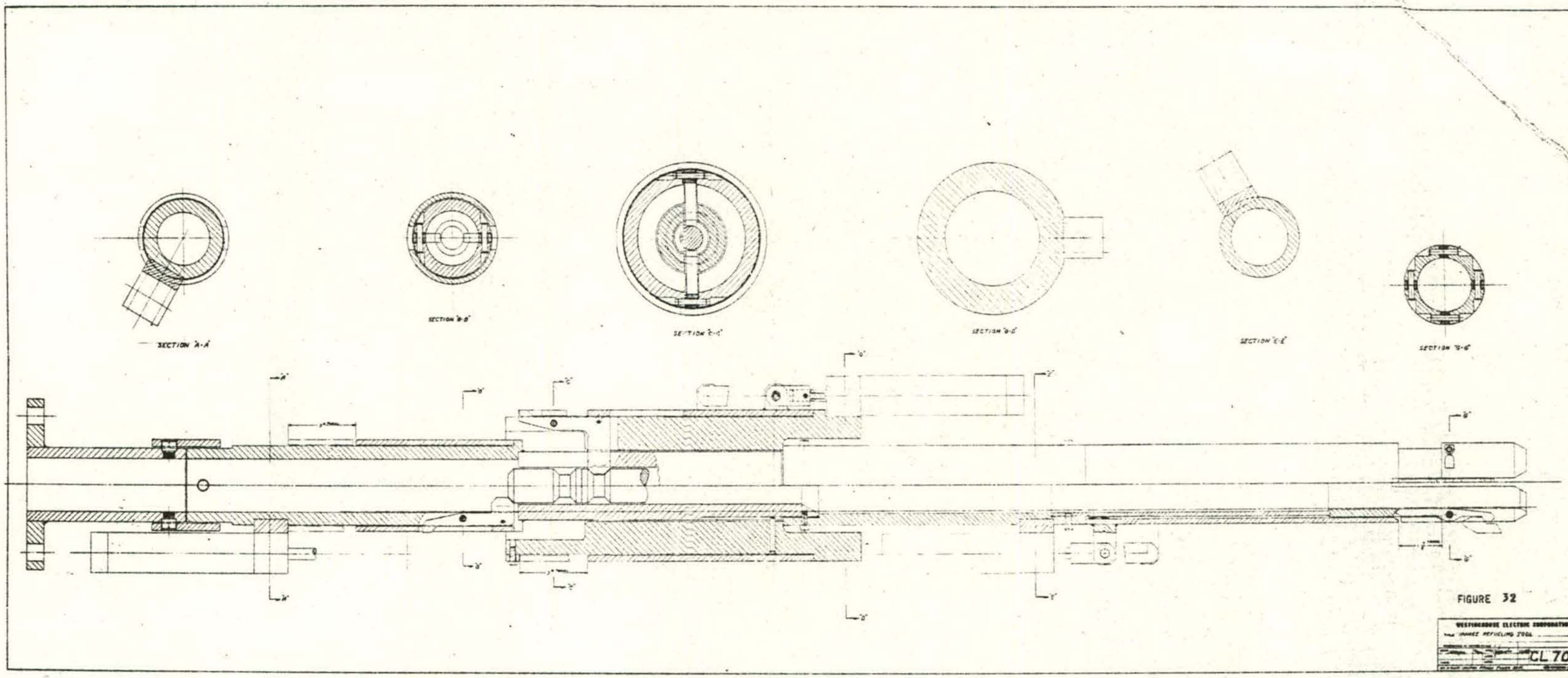


FIGURE 32

Yankee Core Baffle

A one-quarter scale model of the Yankee Core Baffle was completed in the APD Model Shop. The model (Figure 33) was subjected to pneumatic tests, and a few leaks were repaired. The model will now be used for stress and deflection tests.

4.4 Design for Critical Experiment and Irradiation Tests

R. D. Edsell G. H. Eng

CRX Core Internals

The Yankee CRX reactor core structure consisting of a core barrel, aluminum upper and lower core plates, and a set of lucite spacer plates have been in use since December. The core plates in use have a 0.435 inch center to center spacing and give a 3:1 water/uranium ratio.

The core plates and lucite spacer plates having a 0.405 inch spacing (2.23:1 water/uranium ratio) have been received.

The plates having a 0.470 inch spacing (4:1 water/uranium ratio) were received and are ready for installation in the core when required.

Eight sets of cruciform supports for holding fuel rods in vacant control rod slots are being fabricated by the Westinghouse APD Model Shop. The material and labor required for all of these items are charged to Project 10.0.

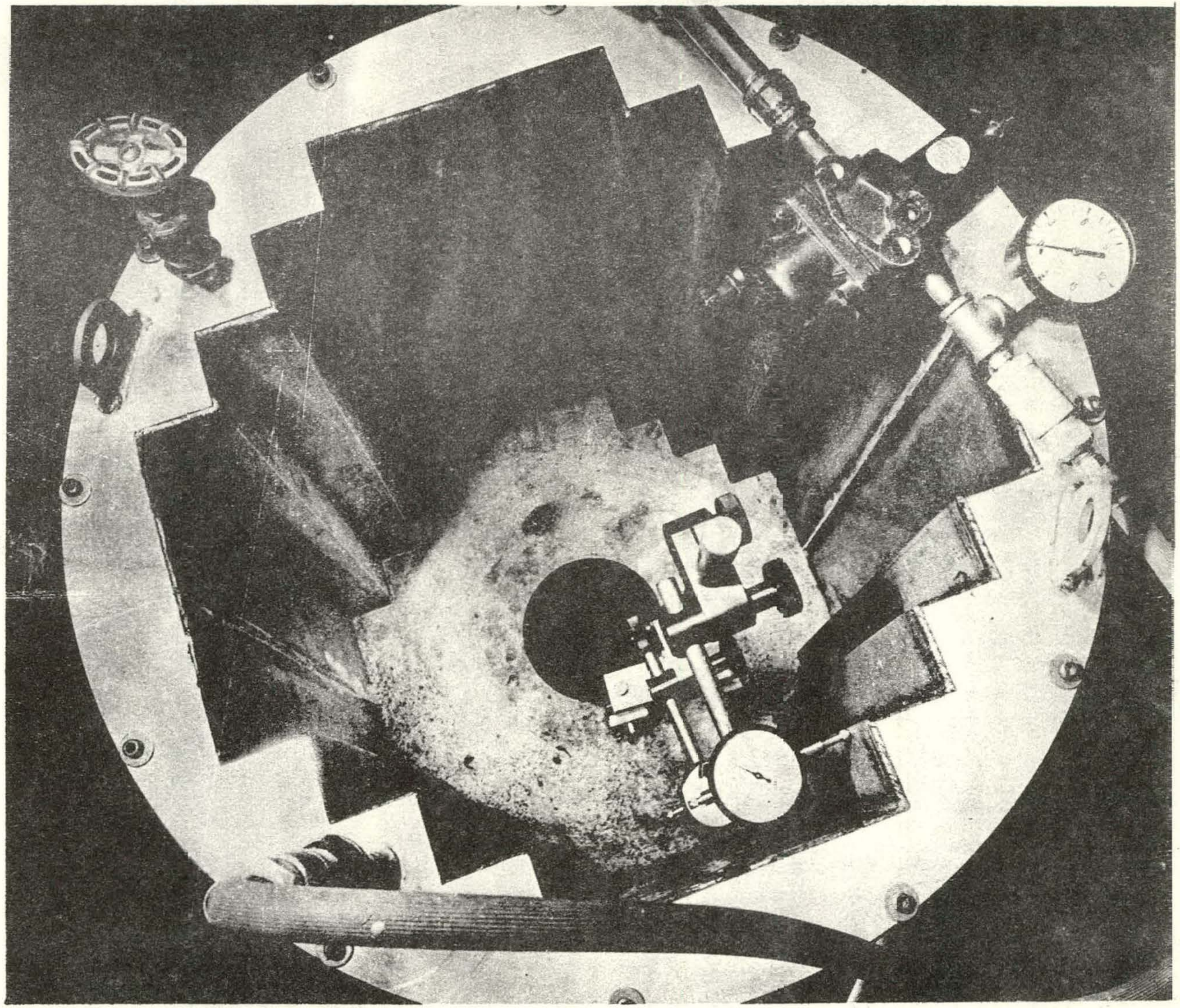


Figure 33 Yankee Core Baffle - One • Quarter Scale

5.0 THERMAL AND HYDRAULIC DESIGN

Thermal and Hydraulic Design Section: A. G. Thorp, Manager

The work performed under this project is directed toward the development of a design which will have satisfactory thermal and hydraulic characteristics under conditions of steady state, transient, and emergency operating conditions.

5.1 Thermal Design

A. Bournia E. McCabe A. Petrovas

Loss of Flow Accident

A loss of coolant flow accident in the Yankee reactor was studied assuming a power loss to two pumps; this is considered to be the most serious accident. Two types of flow coast down curves were considered in the analysis; (1) the loss of power to two pumps only; and (2) the loss of power to two pumps with a momentary loss of power to the remaining two pumps. For these calculations, the analysis continued beyond the saturation temperature with flow distribution, assuming that steady state conditions prevailed at each instant. In order to account for the flow distribution, the pressure drop in the hot channel was equated to that of the average channel which was considered to be located in an infinite reservoir. The pressure drop across the channels was determined by the following equation:

$$\Delta P = \frac{fLG^2 v_{av}}{2g\ De} + \frac{G^2}{g}(v_o - v_1) + \frac{L_o - L_1}{v_{av}} + \frac{1}{g}\int_0^L \frac{\partial G}{\partial t}\, dL + K_c \frac{G^2}{2g} v_{av}$$

The flow and power output for average conditions at any instant during the accident were determined. The results of the analysis are shown in Figures 34 and 35. The hot channel heat flux curve in Figure 35 shows the heat flux distribution that exists at point "A" in Figure 34 (4.5% quality steam). The burnout heat flux is also plotted in Figure 35 to indicate that no detrimental temperature condition is reached. A detrimental condition would exist if the two curves became tangent or intersected.

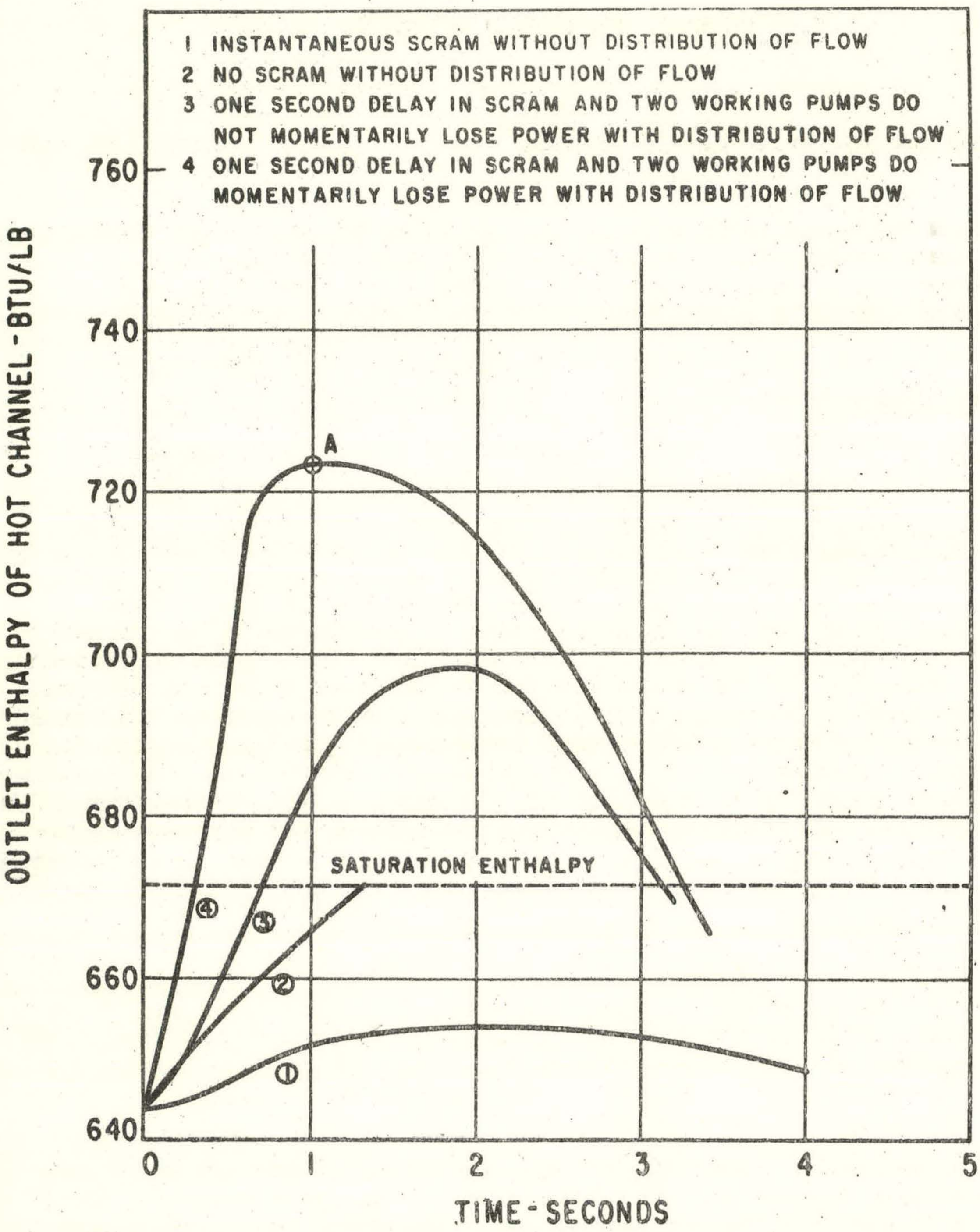


Figure 34 Transient Analysis of a Two Pump Loss-of-Flow Accident in the 482 MW Core

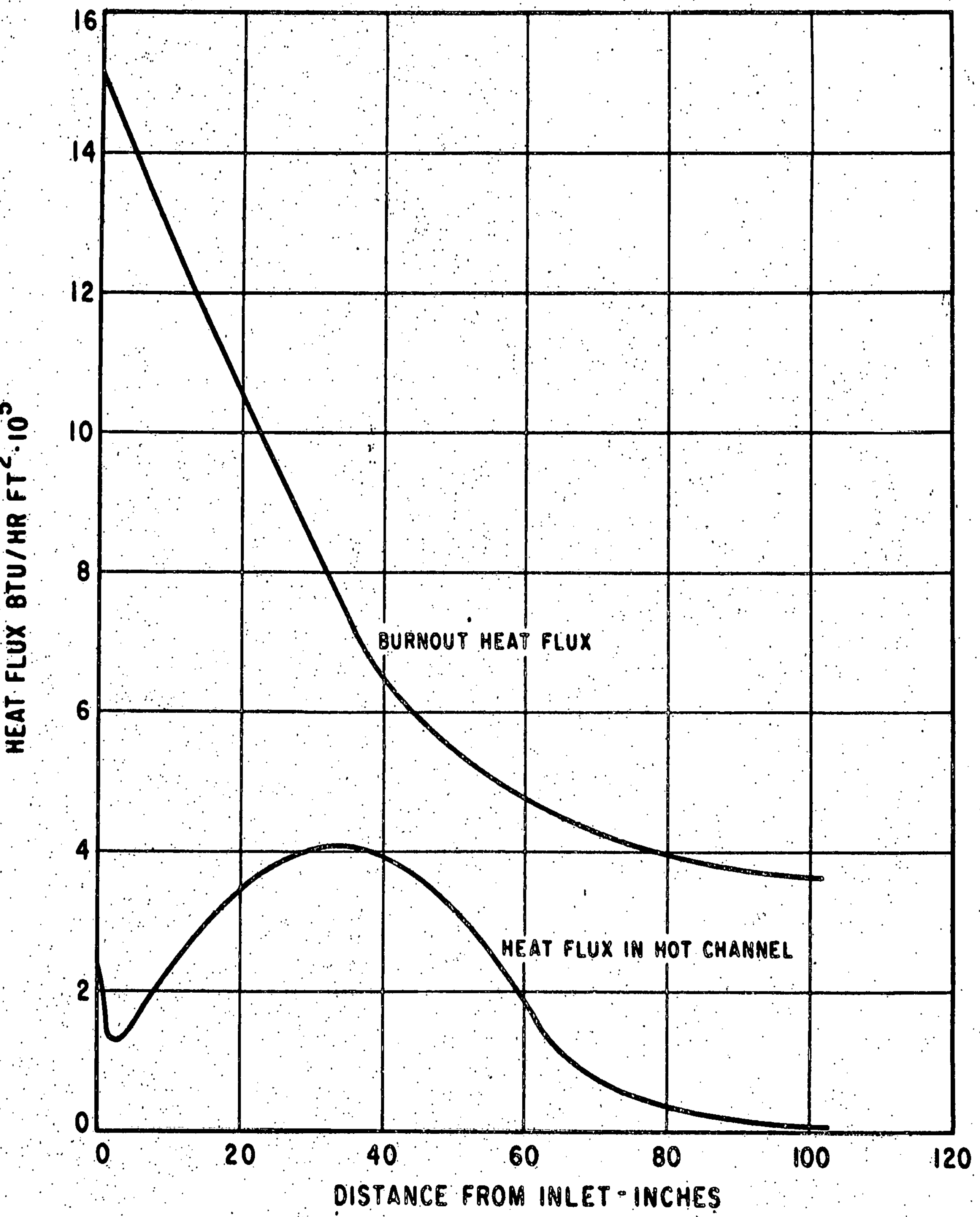


Figure 35 Burnout and Local Heat Flux vs. Distance from Coolant Inlet of the 482 MW Core

Pressure Inside Fuel Rod

Fission gases will be released from the UO_2 pellets during reactor operation. Using the following "cold" dimensions of the Yankee first core fuel rod, the internal pressure at the operating temperature of 1000°F. was calculated:

Pellet diameter	0.294 inch
Diametral gap	0.001 inch
Tube O.D.	0.337 inch
Tube I.D.	0.294 inch
Tube length	91.375 inches
Total end void	0.0418 cubic inch

Assuming a total of 351×10^{-4} gm-moles of fission gases were released to the end of core life, and the fuel rod contained no spacer discs, the pressure was calculated to be 4000 psi. Additional calculations are being performed using more realistic dimensions in order to obtain a better estimate of the pressure in a hot fuel rod. A topical report is being prepared including the calculations pertaining to this subject.

5.2 Hydraulic Design

A. A. Bishop and R. Berringer

Fibre Stress due to Water Hammer

When power to one main coolant pump is cut off, a water hammer is developed at the check valve by the reverse flow of water. This water hammer generates a pressure wave which causes an increase in the fibre stress of the reactor vessel components. If the maximum pressure excursion is 400 psig at the check valve, the additional fibre stress in the lower core barrel is 1000 psi. Because of the relatively small stress increase and the short duration of the applied forces, the effect of a water hammer excursion is not serious so far as the reactor vessel components are concerned.

Local Boiling Effects

The effect of local boiling on pressure drop and coolant flow distribution was investigated. When local boiling occurs in a semi-open heterogeneous reactor core consisting of parallel flow channels, a flow redistribution results. This is caused by a difference in the pressure drop in the local boiling regions and in the parallel adjacent regions. A semi-open heterogeneous reactor core is one in which some of the coolant water may flow in a radial direction from the core axis while traveling from the bottom to the top of the fuel assembly.

Assuming a straight line relationship between the enthalpy of the coolant and the friction factor ratio, the ratio of local boiling to single phase friction factor has a maximum value of 1.42 in the Yankee core (Figure 36).

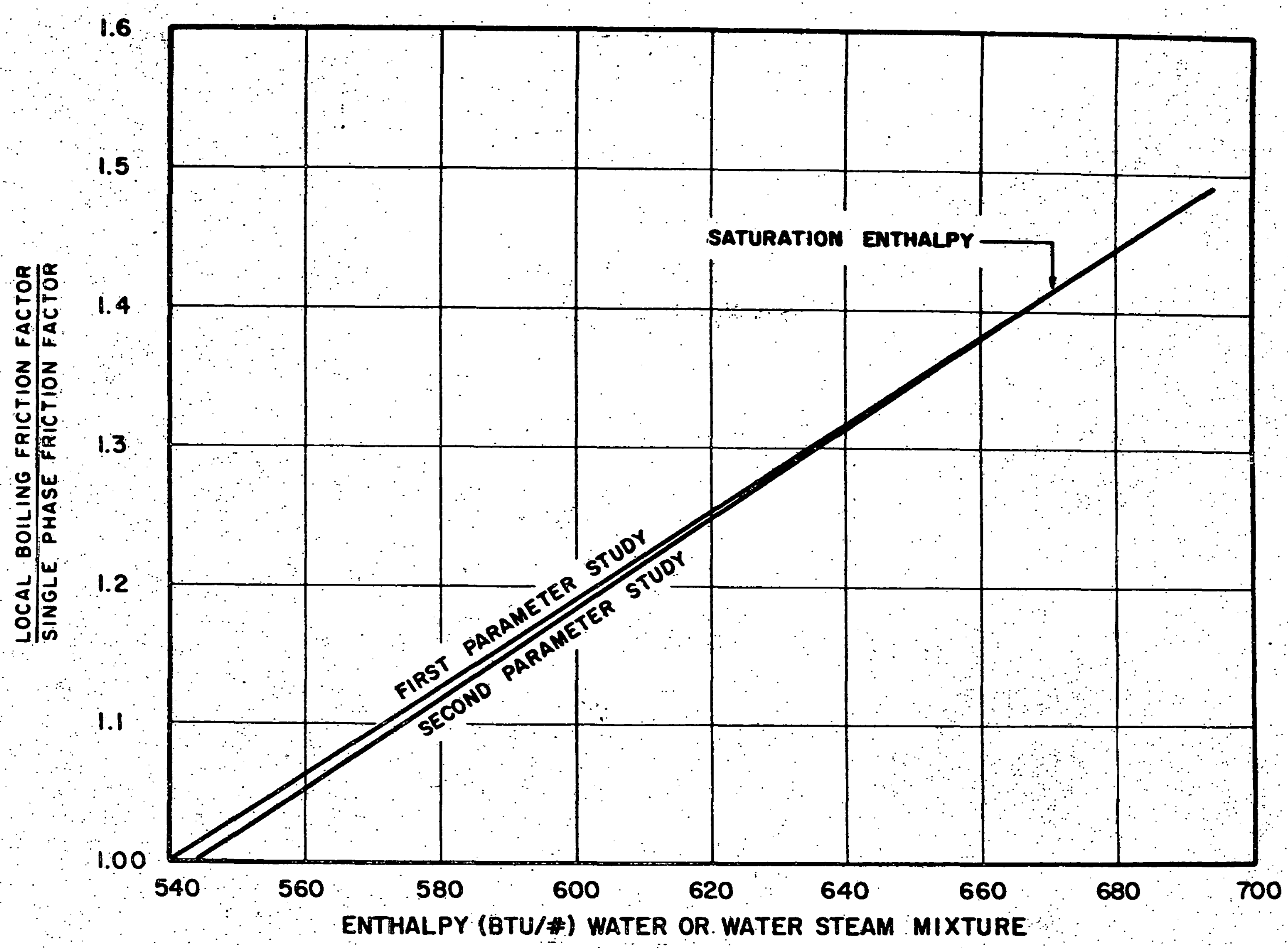


Figure 36 Local Boiling Friction Factor

Both analytical and graphical methods were developed to determine the region of the core in which local boiling may exist. Figures 37 and 38 indicate the application of the graphical method to the extremities of the local boiling region. For a cylindrical reactor core, the region in which local boiling might exist is an ellipsoid whose major axis is the locus of points of maximum surface temperature.

The maximum possible volume of local boiling was calculated to be 10.4 cubic feet (4.5% of the total core volume) in the first parameter study. The second parameter study gave a volume of 14.4 cubic feet which is equivalent to 5% of the core volume.

The flow redistribution in the reactor was calculated by a two channel analysis using the pressure gradient obtained from an average channel and applying to the hot channel. The loss of flow in the hot channel was calculated to be 12.0% for the first parameter study, and the effect of flow redistribution is shown in Figure 39. A second parameter study gave a loss of flow in the hot channel of 16.2%.

A topical report "The Effect of Local Boiling on Pressure Drop and Flow Distribution in the Yankee Reactor", YAEC-75, is being published.

Pressure Drop Studies

Both visual flow studies and pressure drop measurements were made on a 1/12 size plastic reactor vessel model. Ceramic spheres and Raschig rings were placed in the vessel to simulate the fuel assemblies. The experimental pressure drop data was within 5% of the theoretical calculations. The measured pressure drop across the inlet nozzle and the associated direction turns was considerably lower than the calculated value; the calculations assumed a one velocity head loss.

The pressure drop across the reactor vessel model was proportional to the 1.81-2.04 power of the flow rate for rates between 50 and 300 gpm. Various types of packings were used to simulate the fuel assemblies. The pressure drop across regions external to the core was virtually independent of both the pressure drop across the core and the core geometry.

Experiments to observe the coolant flow in the reactor vessel model using a dye tracer technique revealed no unusual hold-up time in the vessel. Flow appeared to be stable everywhere in the vessel. The control rod guide tubes exhibited a large influence on the flow pattern in the region where the water enters the core through the lower core support plate. The method of supporting the control rod shroud tubes also had an effect on the flow pattern in the lower plenum. A strap type mechanical brace was recommended for the support.

A study was made with a reactor vessel model to determine the pressure drop relationships in a Yankee fuel assembly. The experimental pressure drop across the assembly model was within 15% of a value calculated by equations for flow inside pipes. It was concluded that the equivalent diameter concept can be used to calculate flow parallel to a square

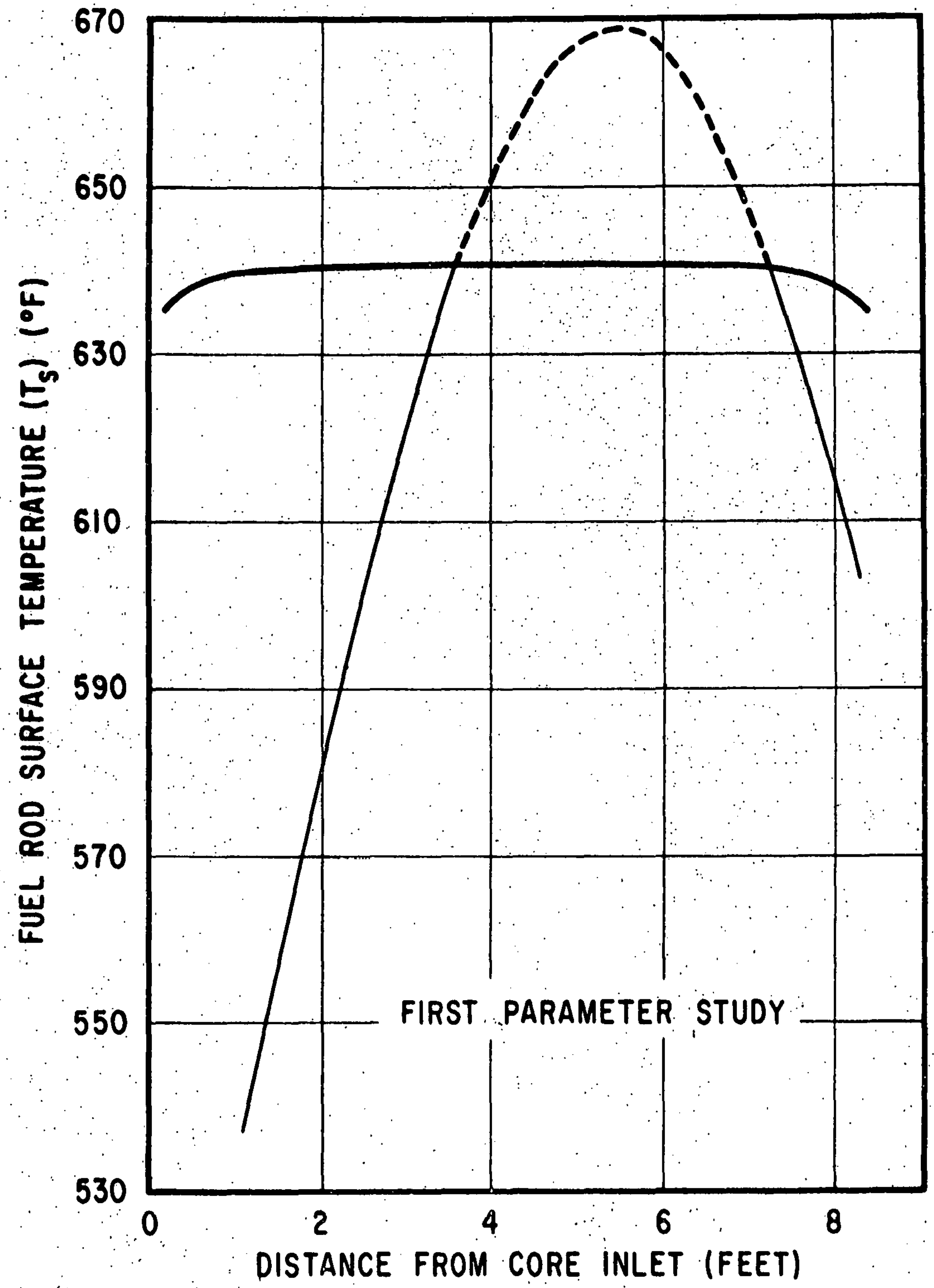


Figure 37 Limits of Local Boiling along a Hot Channel at the Center of the Core

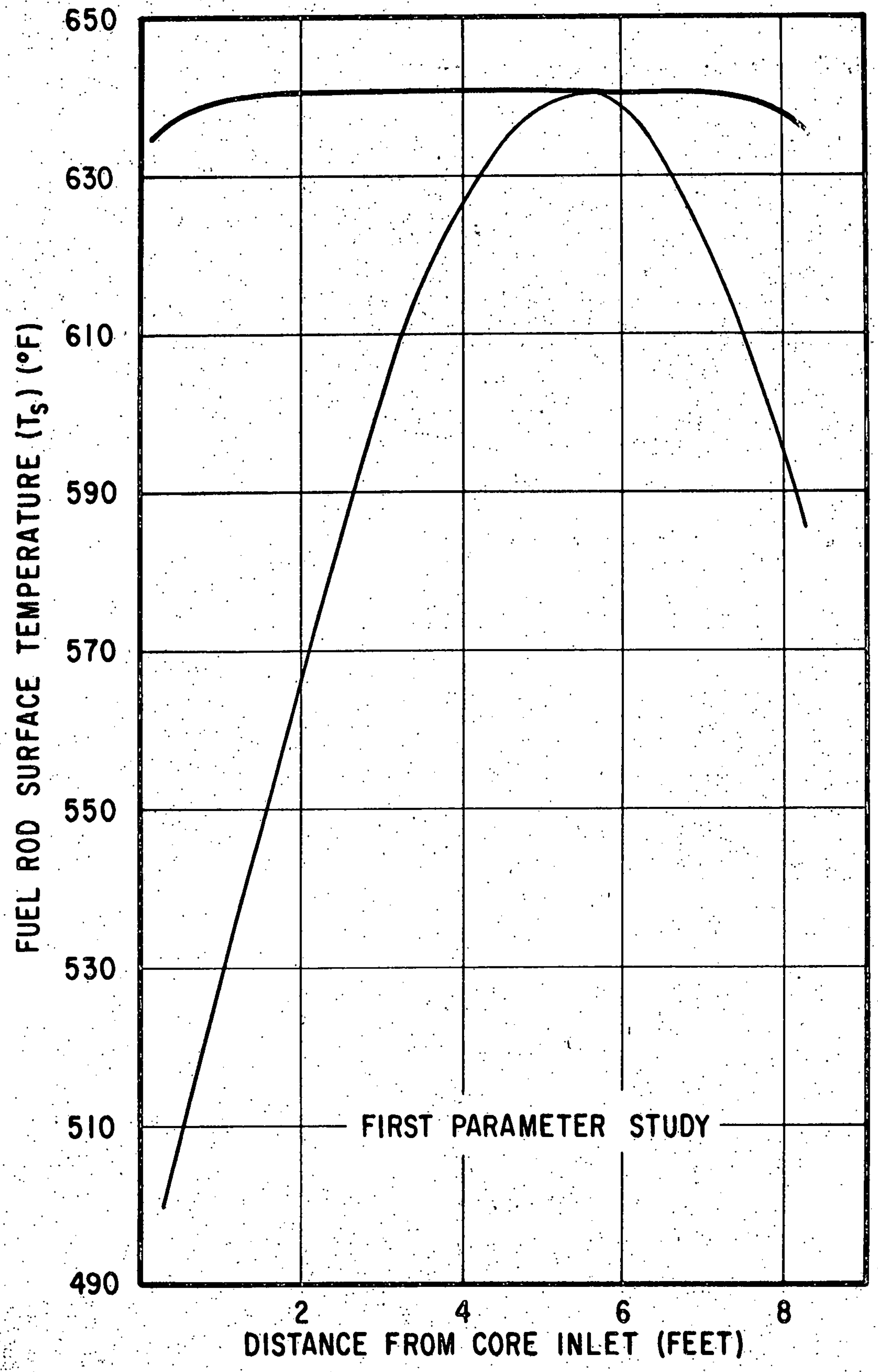


Figure 38 Limits of Local Boiling at a Radius of 1.16 Feet From the Center of the Core

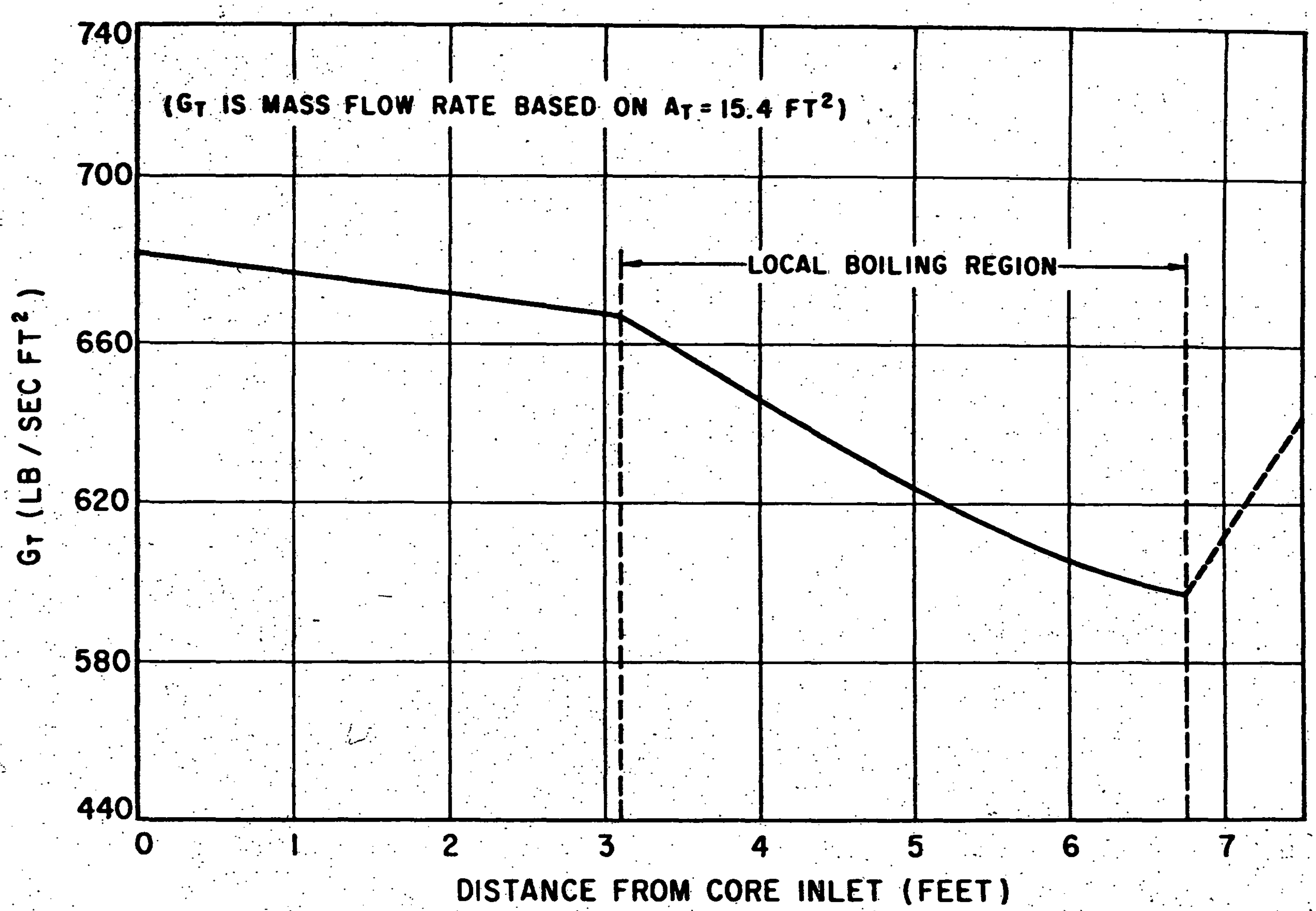


Figure 39

Mass Flow Rate in the Hot Channel
(First Parameter Study)

lattice of 0.335 inch OD rods on a 0.425 inch center to center spacing. A study of the pressure on the Yankee control rod due to hydraulic side forces indicated that the surface contact pressure due against the control rod guide blocks to flow distribution will not exceed 7.5 psi.

A topical report "Pressure Drop and Visual Flow Studies for a Heterogenous Reactor Vessel Model", YAEC-74, has been written and will be issued as soon as experimental data of flow distribution in the lower plenum of the reactor vessel model is compared with calculated values.

Cooling of Reactor Vessel Head

A design study was made on the Yankee reactor vessel hold-down flange with the objective of directing a constant flow of water against the inside of the reactor vessel head. The water would be used to cool the reactor vessel head. Water could be forced through holes drilled in the guide tube hold-down plate and flange assembly. Since the water directed against the head by-passes the core and must be charged to leakage, it was calculated that the maximum amount of water directed against the head should be limited to 0.57% of the total coolant flow. Local circulation is prevented, and circulation against the vessel head will be insured, if the hydraulic circuit is arranged so that the only connection between the upper plenum and the vessel head is through the control rod guide tubes. Thirty 3/8 inch diameter holes will cause 0.53% of the total coolant flow to be directed against the vessel head. This is based on the conservative assumption that the reactor total pressure drop will occur across the holes.

6.0 CONTROL ROD DEVELOPMENT

Metallurgy Section: R. K. McGeary, Manager

Specifications for reactor control rod material and control rod design are to be developed under this project.

Silver-Indium-Cadmium Alloy Creep Strength and Corrosion Studies

E. S. Foster

The creep strength of silver-indium-cadmium alloy can be improved from less than 200 psi to more than 1000 psi for a creep rate of 1% in 10,000 hours by increasing the grain size from approximately ASTM 4-7 to ASTM 2-3. From heat treatment of the alloy at Ⓦ APD in the range - 400°C to 600°C, it was determined that a temperature of 600°C for two hours was necessary to produce an ASTM grain size of 0 to 3.

Several specimens of silver-indium-cadmium alloy were exposed in PAR Loop "A" under the conditions stated in subproject 3.3. Figure 40 is a photomicrograph (250 X) of a portion of a corrosion coupon of 80%Ag-15%In-5%Cd alloy tested for 30 days in loop water containing 5 ppm boron at a pH of 6.8. Figure 41 (250 X) shows a portion of an identical coupon tested for 30 days in loop water containing 1600 ppm boron at a pH of 5.2 with no pH adjustment. Figure 42 (250 X) shows a portion of a third identical coupon tested for 30 days in loop water which was cycled two times from 1600 ppm boron at a pH of 5.2 to 5 ppm boron at a pH of 6.8. Metallographic examination and evaluation of the specimens indicated that they sustained internal oxidation. Other facilities working on the development of these alloys have indicated identical oxidation behavior for the 80%Ag-15%In-5%Cd alloy when tested in hydrogenated water without boron additions. The protection of the silver-indium-cadmium alloy by the use of electroplated nickel coatings is being investigated.

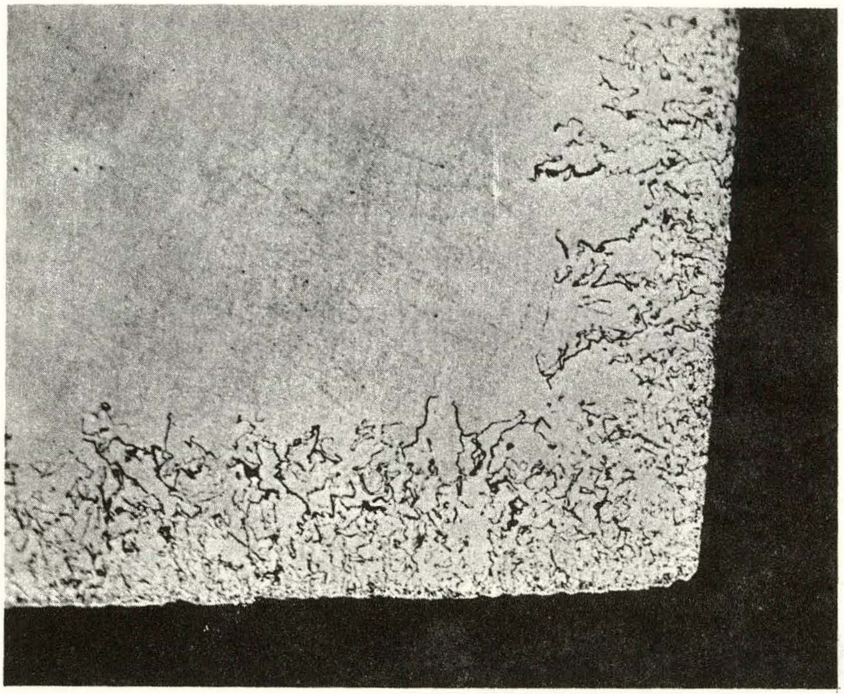


Figure 40 Corrosion Coupon of Silver-Indium-Cadmium Alloy Tested in 5 ppm Boron Loop Water (250X)

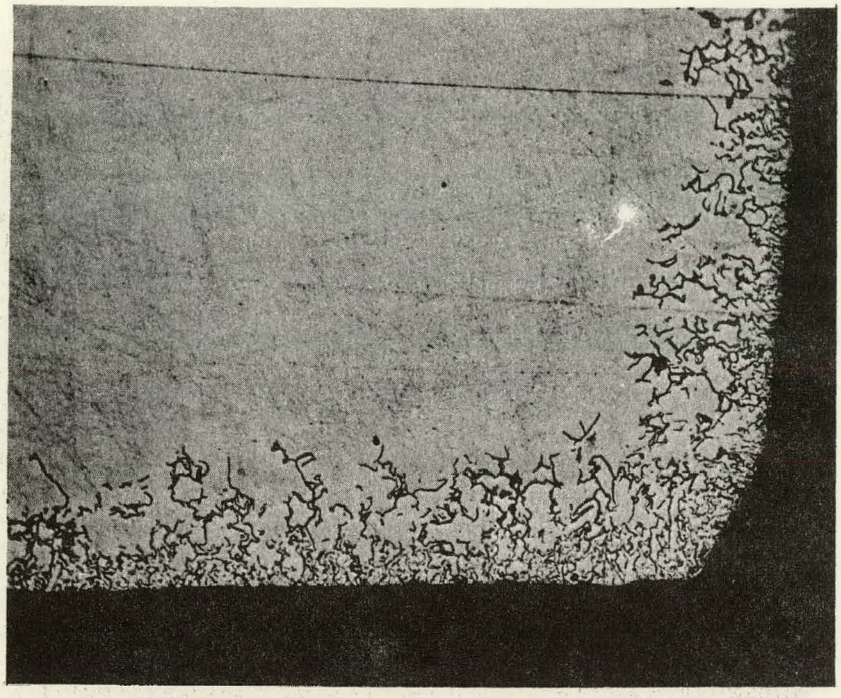


Figure 41 Corrosion Coupon of Silver-Indium-Cadmium Alloy
Tested in 1600 ppm Boron Loop Water (250X)

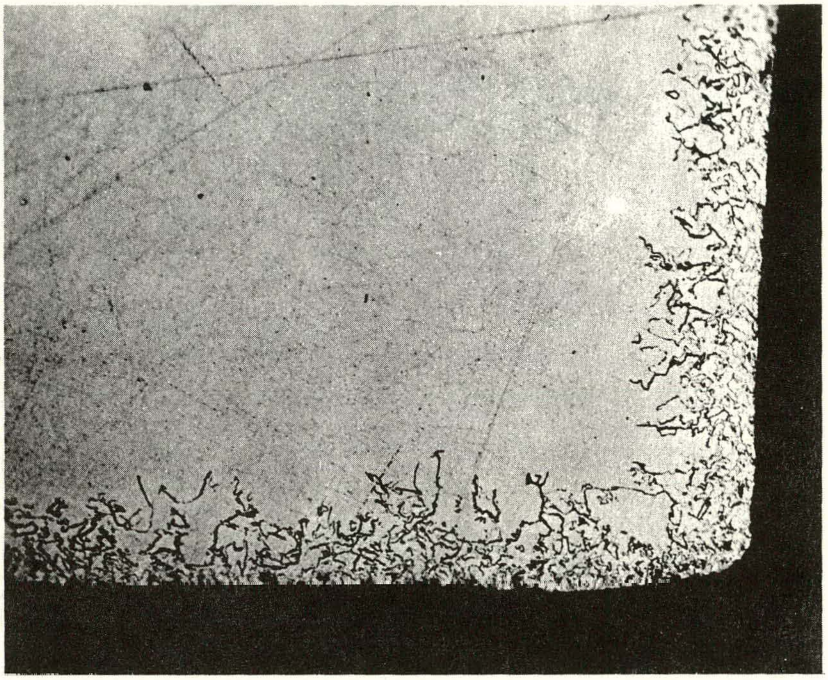


Figure 42 Corrosion Coupon of Silver-Indium-Cadmium Alloy Tested in Loop Water Cycled from 1600 ppm to 5 ppm Boron (250X)

7.0 INSTRUMENTATION AND CONTROL

Instrumentation and Control Section: C. F. Obermesser, Supervisor

This project covers the investigation and development of an overall control system and instrumentation including analyses of system functions and development of specifications for system components.

Digital Computer Representation of the Transient Behavior of the Steam Pressurizer

J. P. Cunningham S. Ehrenpreis

Preparation was begun of a digital computer code to represent the transient behavior of the steam pressurizer. The need for a detailed analysis of pressurizer behavior depends upon the importance of pressure effects on the overall reactor system dynamics. The variety and complexity of processes involved in the pressurizer make it difficult to produce a simple representation for use in systems analysis. Approximations can be especially misleading if system transients include both positive and negative volume surges or, if the transients are of long duration, as are found in studies on pressurized water reactors. The computer program is intended to provide solutions of pressure transients, and to aid in development of analog models appropriate for the analog studies of interest to designers of reactor control systems.

The equations have been written to describe the thermodynamic processes of expansion and compression of the steam and water, condensation and boiling, heat transfer to the pressurizer walls, mixing of incoming coolant, and pressure drop in the relief pipe due to friction and inertia. The water in the pressurizer was arbitrarily divided into three separate regions to control the degree of mixing. Operation of spray and relief valves and of the heaters are dependent on the pressure. The principal input to the program can be either the volume flow rate of coolant into and out of the pressurizer, or the net rate of heat input into the reactor primary cooling system, either input to be specified as a function of time. If the latter input form is used, the use of the coolant charge-and-discharge system as a pressurizer water-level control is possible. Programming of the system of equations describing this problem for the IBM-704 electronic digital computer is being done in collaboration with the Westinghouse Analytical Department.

The transfer functions necessary to determine the frequency spectrum of the average primary loop temperature, T_{avg}, at the points of insertion of the temperature sensing equipment were obtained. These transfer functions give the ratio of the hot leg temperature at the steam generator primary inlet to the reactor power generation, and the ratio of the cold leg temperature at the steam generator outlet to the load power demand. The actual frequency spectrum for the temperature variables requires a knowledge of the frequency spectrum for the reactor power generation and the load power demand. However, in order to determine whether or not the dynamic characteristics of the T_{avg} instrument channels should be included in a study of the controlled plant, a knowledge of the frequency spectrum of these two transfer functions is sufficient.

If bandwidth is defined as the frequency at which the gain of the transfer function has been reduced by 3 db from the zero frequency gain, the results can be summarized as shown in Table XIV.

TABLE XIV

Bandwith Values for T_{avg} Measurement

Transfer Function	Bandwith in CPS-Yankee
$\dfrac{T_{HL}^{*}}{q_c^{*}}(\omega)$	0.034
$\dfrac{T_{CL}^{*}}{q_1^{*}}(\omega)$	0.010

T_{HL}^{*} is the deviation of the hot leg temperature from the steady state value.

T_{cl}^{*} is the deviation of the cold leg temperature from the steady state value.

q_c^{*} is the deviation of the power generated in the core from the steady state value.

q_1^{*} is the deviation of the power demanded by the steam generator from the steady state value.

(ω) is $2\pi f$ where frequency is measured in cycles per second.

These results emphasize the fact that information relating to high frequency components of power or load transients is strongly attenuated by the reactor system response and is, therefore, not easily obtained from T_{avg} measurements.

8.0 PLANT SYSTEMS DEVELOPMENT

Reactor Systems Design Group: P. B. Haga, Supervisor
Plant Systems Design Group: H. A. Smith, Supervisor

The work on this project is directed toward providing the analysis, conceptual design, and preliminary parameter of the plant systems.

8.5 Chemical Handling and Control Systems

J. Baron　　A. R. Collier　　H. G. Hargrove

Coordination of the Research and Development Chemistry Program with the primary plant design requirements was continued. Subjects reviewed and investigated during the period included:

1. Isotopic leakage through lithium-saturated resins.

2. Water chemistry of the in-pile loop.

3. Effects of hydrazine addition in the main coolant system for oxygen scavenging.

4. Methods for determining decontamination factors using mixed resins.

5. Evaluation of decontamination agents and procedures.

8.11 Reactor Handling Tools and Plant Shielding Analysis

Reactor Handling Tools

Head Gasket and Seal Ring Fixture

E. V. Anderson　　A. F. Fritz　　H. S. Kresny

The conceptual design of the head gasket and seal ring fixture was altered to provide a downward vertical motion for the latching and unlatching operations that would be required to engage the metallic O-Ring head gaskets and the head seal ring membrane. The head gasket and seal ring will be manually loaded onto the fixture on the floor above the flooded reactor compartment and then lowered by a crane into the water to its proper position on the vessel flange.

The design of a mock-up of the head gasket and seal ring fixture was begun. The mock-up will be used in the Westinghouse APD High Bay Building Deep Pit to test the operational feasibility of the design.

Plate and Barrel Handling Fixture

The conceptual design of the guide tube holddown and support plate and the upper core support barrel handling fixture was modified to permit a downward vertical motion of the latching mechanism to engage the lifting lugs on the plate and barrel.

The design of a mock-up of the latching mechanism for the plate and barrel handling fixture was begun. The mock-up latching mechanism will be tested in the Westinghouse APD High Bay Building Deep Test.

Plant Shielding Analysis

The draft of Topical Report YAEC-71, "Fission and Corrosion Product Activities in the Main Coolant and Atmosphere of the Vapor Container" has been completed. The report presents the various activities, listed by individual isotopes, that can be expected in the main coolant and vapor container during normal full power operation.

9.0 PLANT SAFETY ANALYSIS

Reactor Engineering Department: W. E. Abbott, Manager
Large Plant Engineering: A. E. Voysey, Manager

This project involves the investigation of overall plant operational safety which is included in the development of the final plant design.

No work was performed under this project during the second quarter of 1958.

10.0 CRITICALITY EXPERIMENTS

Reactor Evaluation Section: P. W. Davison, Manager

Performance of Criticality Experiments at the Westinghouse Reactor Evaluation Center on stainless steel clad UO_2 fuel elements at differing water-to-metal ratios are included in the project. Reactivity parameters and control rod effectiveness are also to be determined.

S. S. Berg W. H. Bergman D. F. Hanlen

Criticality Experiments

Actual critical reactor operation techniques used in experimental evaluation of reactor parameters include such procedures as measuring the reactor response to a variable by reactor period and control rod position change. A second procedure consists of making flux measurements using a variety of foil and wire detectors; the detectors must then be counted and weighed. A draft of a topical report "Yankee Critical Experiments, 3:1 W/U Ratio", has been prepared. A tabulation of experimental runs performed during the second quarter of 1958 as part of the 3:1 W/U ratio core test program is as follows:

TABLE XV

CRX 3:1 W/U Ratio Core Test Runs (April 1 to June 30, 1958)

Clean Core	No. of Runs
Flux Plots	17
Core Reactivity Measurements	10
Control Rod Worths	2
Temperature Coefficients	9
Void Coefficients	4
TOTAL	42
Poisoned Core	
Flux Plots	13
Core Reactivity Measurements	26
Control Rod Worths	3
Temperature Coefficients	1
Void Coefficients	1
Partial water height	4
TOTAL	48

Flux Profiles

To obtain values for reflector savings and buckling, the relative values of the flux at various points along the diameter or the axis of the core are required. Several materials and techniques were employed in obtaining such flux profiles. Typical results are illustrated in Figures 43, 44, and 45. The experimental data were fitted to the appropriate theoretical curves by the least squares method.

Figure 43 shows the flux plots obtained by axially scanning two fuel units at one-inch intervals. One of these units was located at a radius of 6.25 cm from the center of the core, the other at a radius of 12.5 cm. A value of 7.05 cm for reflector savings can be derived from these curves.

Figure 44 is an axial scan taken during a reactor run by moving a semiconductor neutron detector (developed by Westinghouse Materials Engineering Laboratory) in one cm increments along the length of the core. This process is being developed at this time and shows promise as an experimental technique.

Radial flux traverses of the core, using gold and copper detector materials, are shown in Figure 45. Detector spacing is 0.435 inch, that is, the lattice spacing in the 3:1 core. The use of copper as a detector is also being evaluated.

Effects of Extra Stainless Steel in the Core

Nine hundred and seventy grams of stainless steel in the form of straps were inserted in one quadrant of the reactor. The decrease in core reactivity and the flux re-distribution were evaluated. A cylindrical core configuration was used containing 1909 fuel rods, a central control rod and three peripheral control rods as shown in Figure 46. The clean period (stable reactor period with all control rods withdrawn) for the core containing the extra steel (AISI 304 stainless) amounted to 119 seconds, or a reactivity of 5.9×10^{-5} $\Delta k/k$. The "clean" period for the core without the steel straps was 32 seconds, a reactivity of 1.4×10^{-4}. The stainless steel therefore reduces the reactivity by 8.6×10^{-5} $\Delta k/k$. Figure 47 illustrates the axial perturbation of the steel and compares it with that caused by one of the lucite spacer plates.

Radial Traverse Through a Four-Fuel-Unit Water Gap

A measurement was made of the radial flux distribution through a vertical water gap to check the analytical treatment of flux peaking. The results obtained have been used by the Westinghouse APD Nuclear Design Section for comparison with their analysis of such a water gap. Indium foil techniques used in studying the water gap created by removing a 2 x 2 array of fuel units were sufficiently different from standard foil work to warrant further explanation. The foil was mounted in a 1/2 inch wide continuous strip, six inches long, and backed by an appropriate lucite strip to provide the necessary mechanical strength; Mylar tape was used

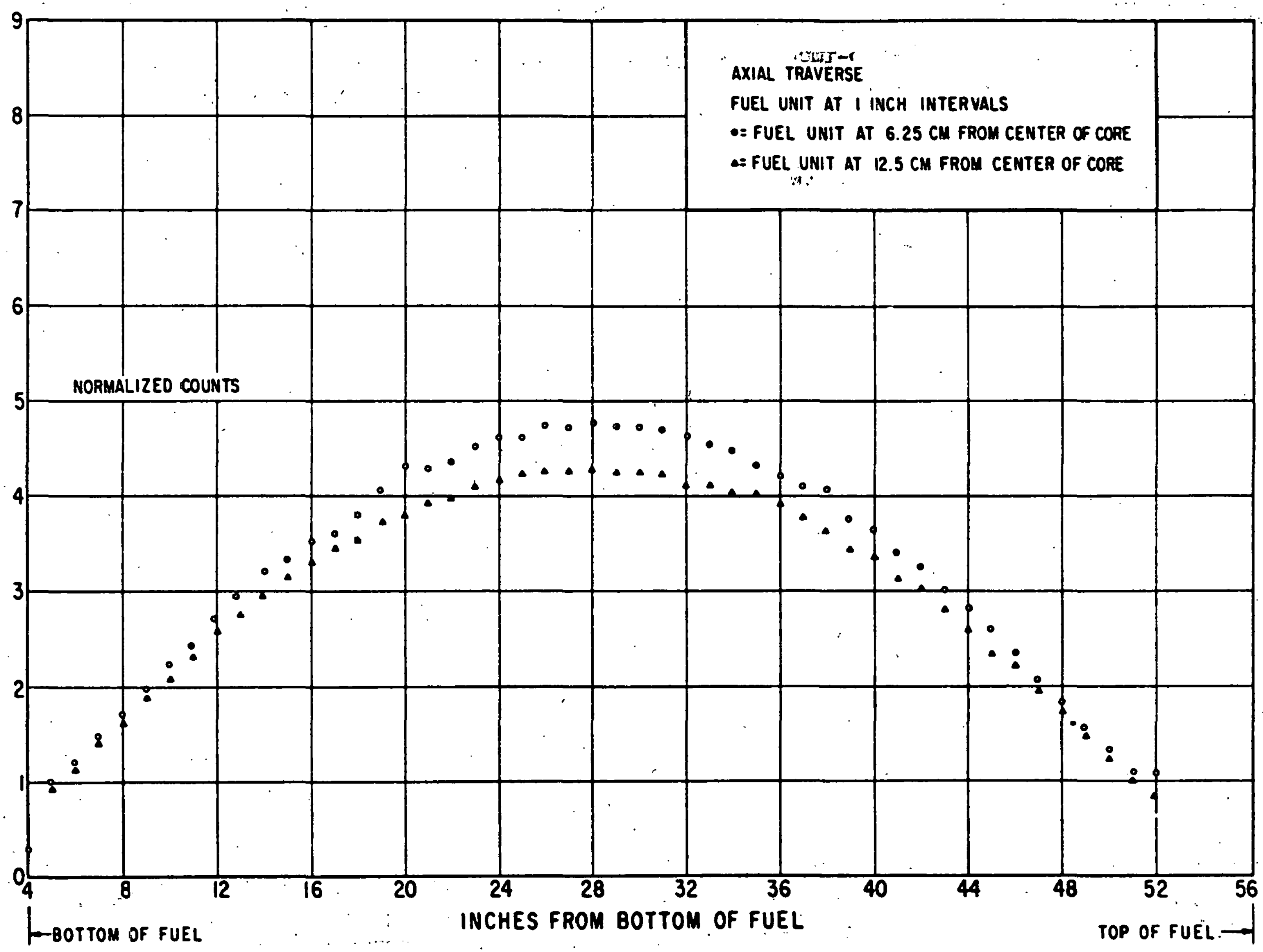


Figure 43 Axial Flux Traverse of Two Yankee Fuel Units

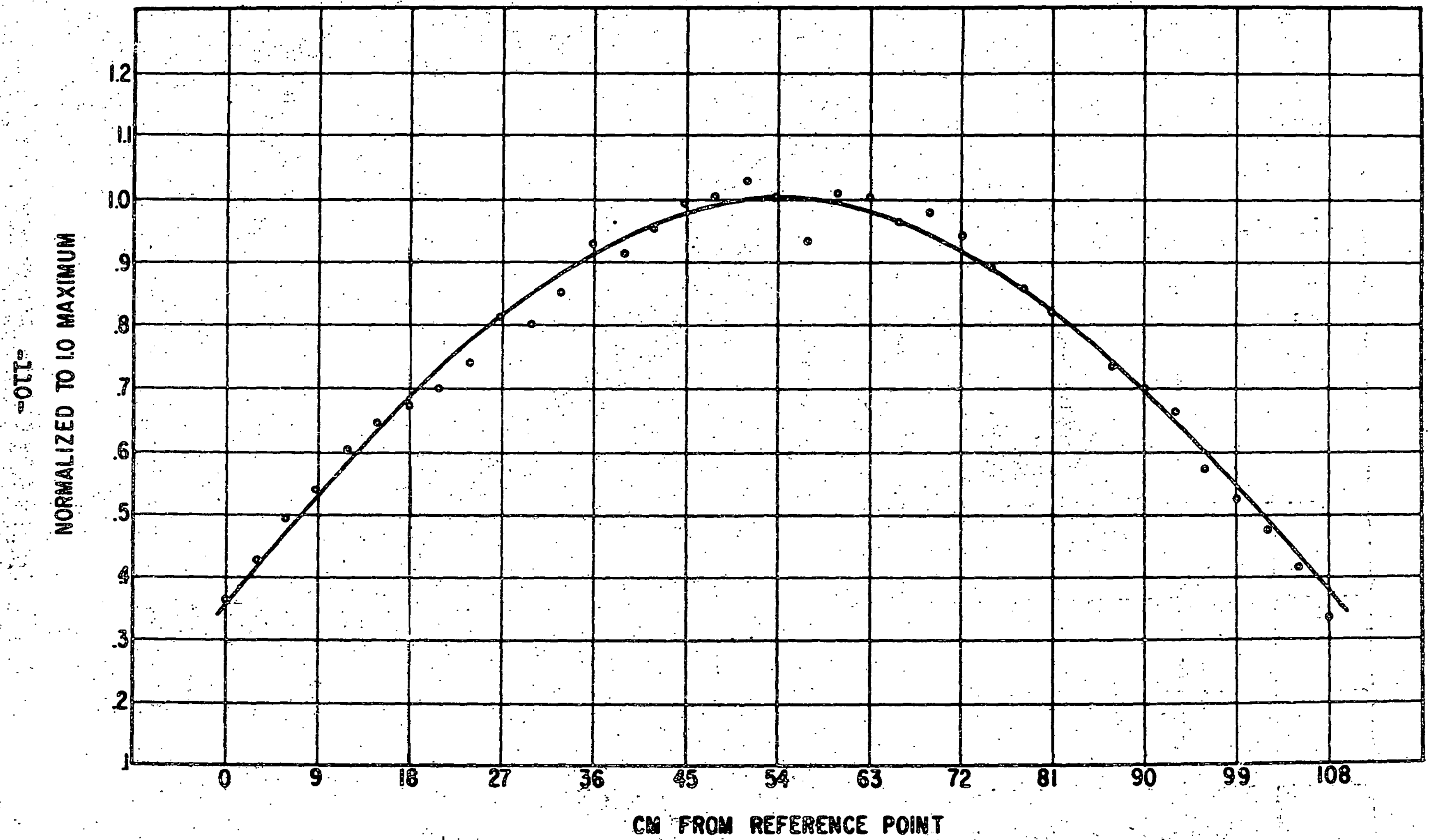


Figure 44 Axial Traverse of Core with Semi-conductor

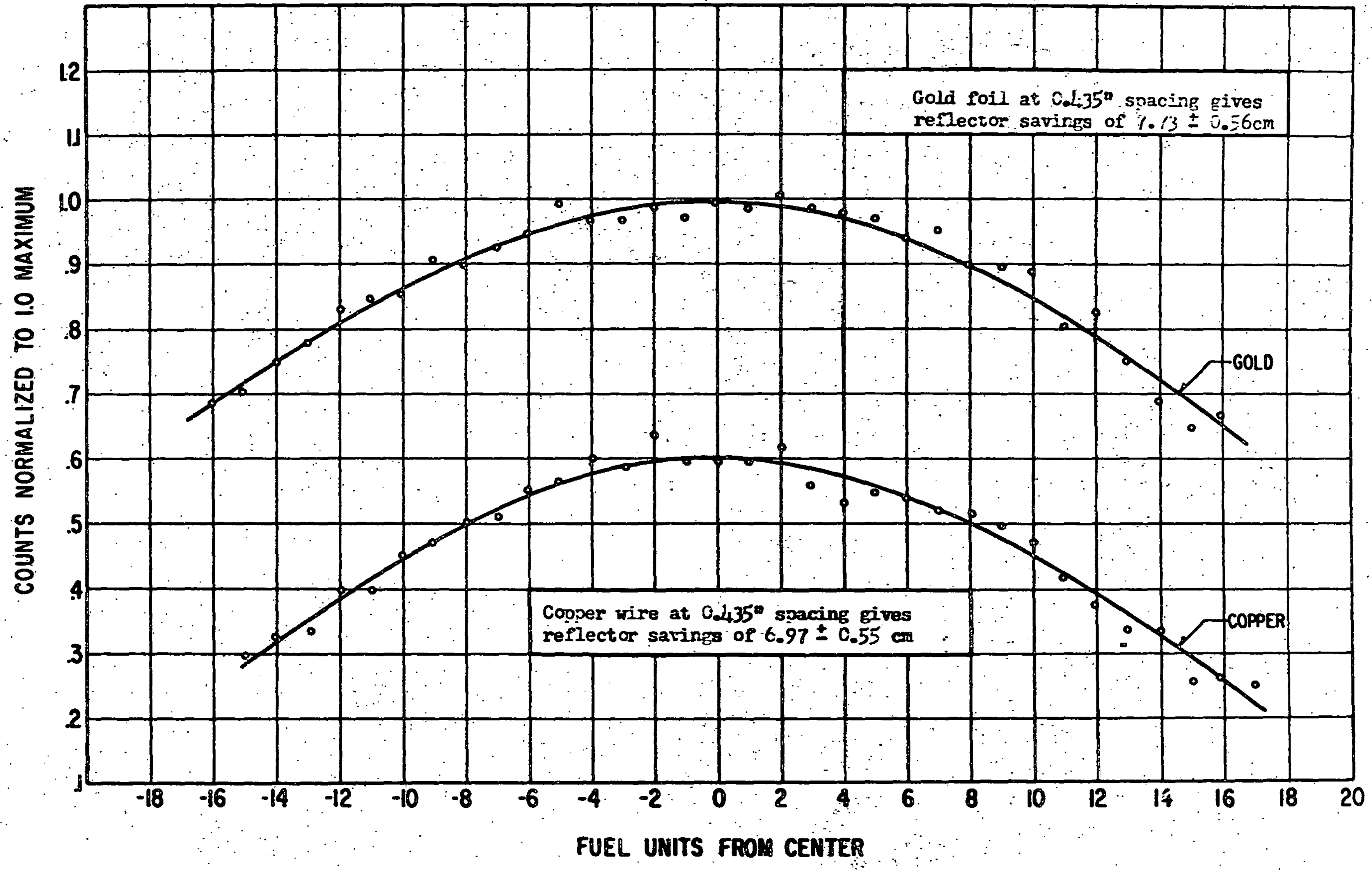


Figure 45 Radial Traverses with Gold and Copper Detectors

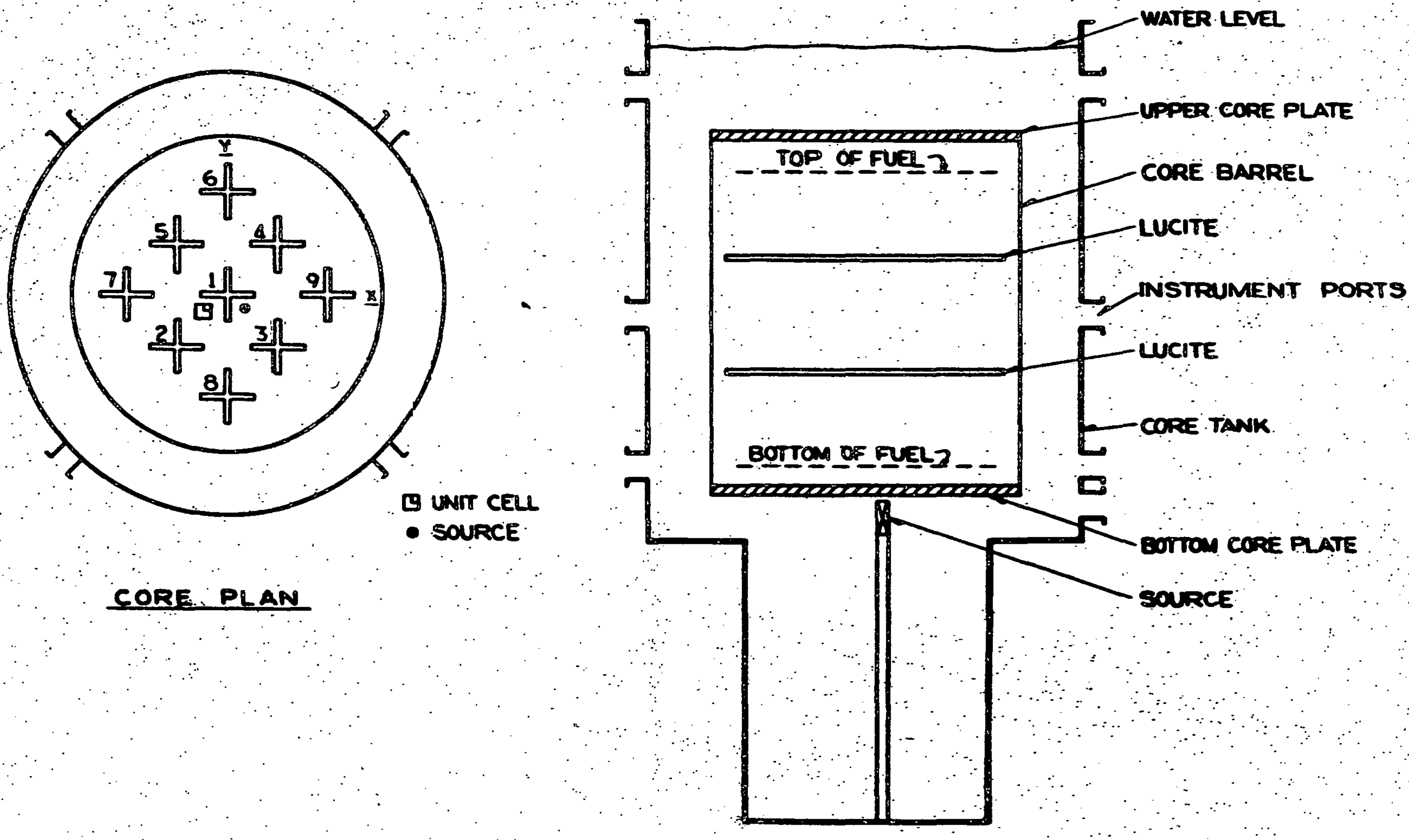


Figure 46 Yankee CRX Core Arrangement

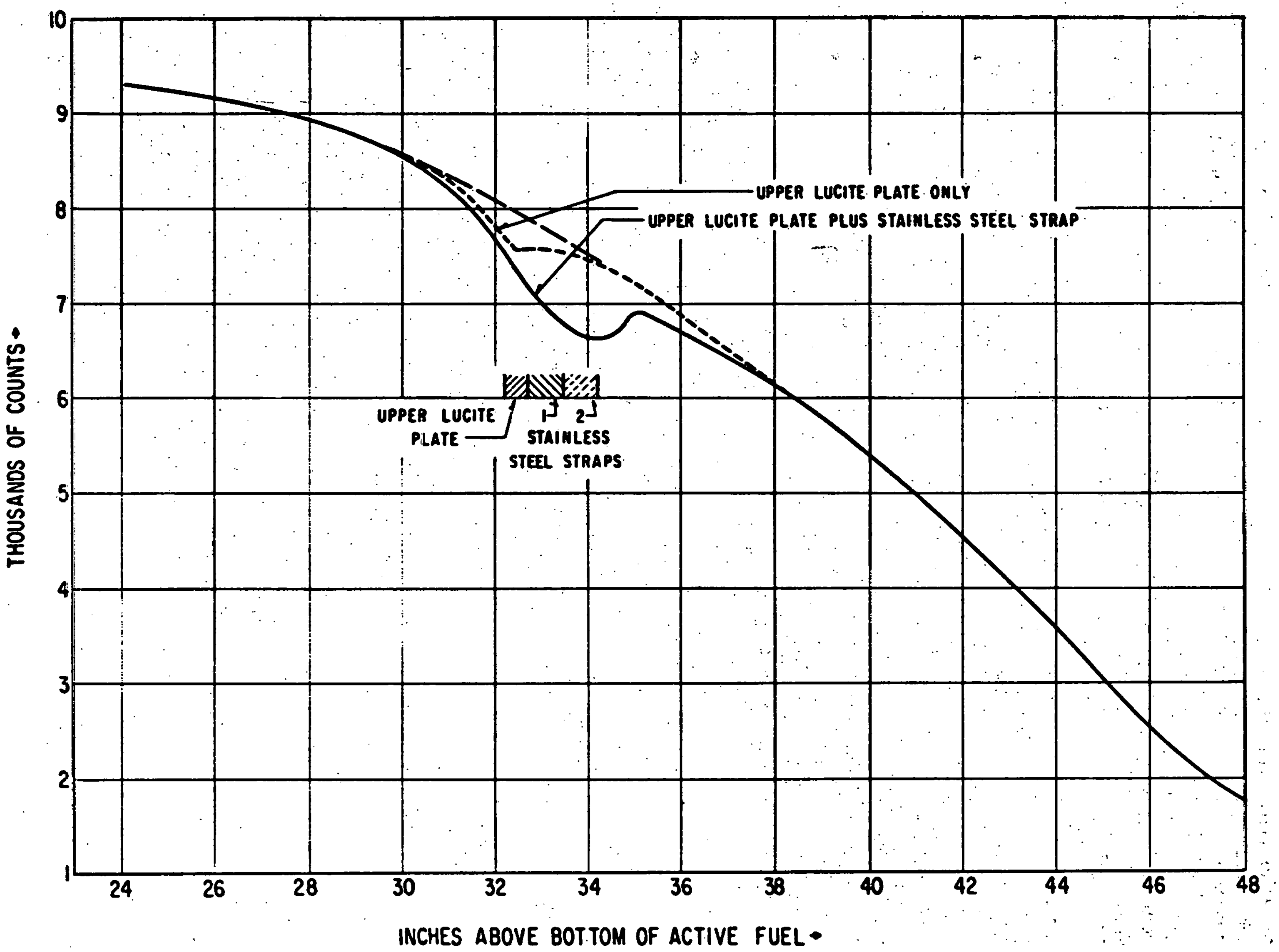


Figure 47 Axial Traverse of Fuel Unit through Perturbation Regions

as adhesive. The strip was inserted horizontally in the reactor through the water gap and irradiated. The strip was then withdrawn from the reactor and the indium removed from the lucite support, still adhering to the Mylar tape. The tape was marked in 0.1088 inch increments, the foil and tape were sliced at each mark, and the sections were then mounted on lucite slides and counted by shielded Geiger tubes. After an appropriate "cooling time", all foil segments were stripped from the Mylar tape and weighed to three significant digits.

During the same run a copper wire was irradiated in a position symetrical with the indium strip. After activation, this wire was also segmented and counted. The resulting flux plot is shown in Figure 48 together with the plot obtained with the indium strip.

Disadvantage Factor

The disadvantage factor is an experimentally measured quantity which can be used to determine the thermal utilization factor. It is the ratio of the average thermal neutron flux in the moderator to that in the fuel. It was measured in the following manner: U-235 (95% enriched) was alloyed with aluminum to make foils having the same effective enrichment (2.7%) as the fuel. These were cut into two different shapes - one (called fuel foils) to match the fuel pellet diameter and the other (called moderator foils) to match the horizontal cross section of the moderator between fuel rods. Three sets of these foils were used, and each set consisted of two fuel foils and one moderator foil.

The sets of foils were placed near the axis of the reactor in, slightly above, and slightly below the central horizontal plane. After the foils were activated by operating the reactor, they were removed and their activity measured by differences in size and for radioactive decay were applied to determine the specific activity.

The ratio of the specific activity of the moderator foil to that of the fuel foil was taken to be the disadvantage factor. After consideration of the errors involved in the three measurements that were taken, it was concluded that the best value for the disadvantage factor is 1.13 ± 0.02. This value has not been corrected for the difference between the cadmium ratio in the moderator and that in the fuel. However, other measurements indicate this difference to be small.

Cadmium Ratios

Cadmium ratios were explored with dysprosium, gold and indium detectors in one series of experiments. Radial location was 7.03 cm from core center. The results are tabulated in Table XVI.

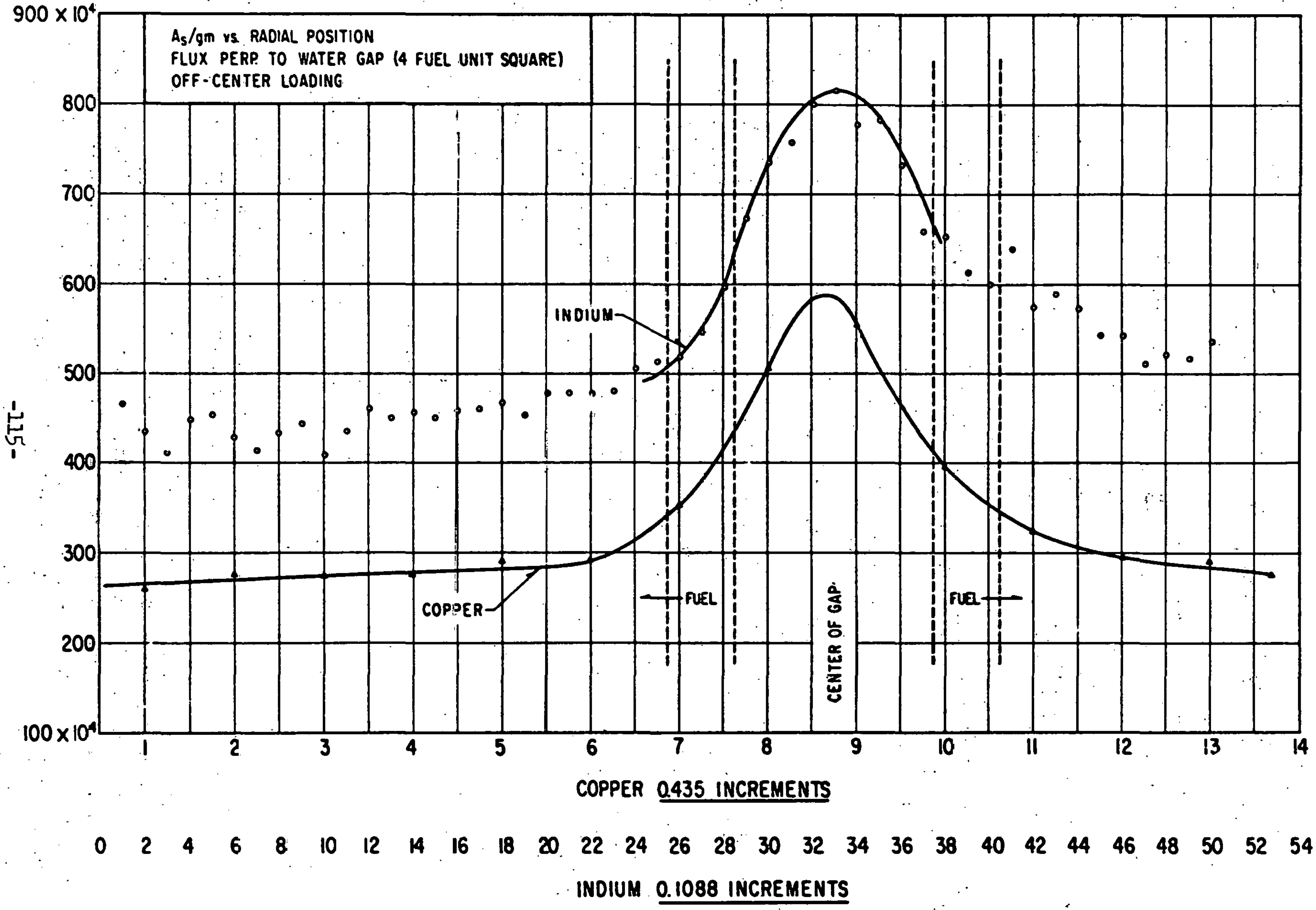


Figure 48 Radial Traverse through a Water Gap of Four Fuel Units

TABLE XVI

Cadmium Ratios of Foil Materials

Displacement from Core Centerline	Dy	Au	In
+ 4.5 inches	-	1.9	2.2
+ 1.5 inches	24	-	-
- 1.5 inches	-	1.8	2.1
- 4.5 inches	23	-	-

A second series of experiments to determine the cadmium ratio of U-235 were also performed. Table XVII indicates the results.

TABLE XVII

U-235 Cadmium Ratios

Fuel Rod Coordinates		Experimental Ratios	Environment
2.5	- 4.5	5.3	Moderator
2.5	-14.5	5.6	Moderator
2.5	-22.5	6.4	Moderator
-4.5	- 2.5	5.3	Moderator
-14.5	- 2.5	6.0	Moderator
-22.5	- 2.5	6.3	Moderator
2	4 and 14	4.7	Fuel

Void Coefficient

The void coefficient was measured in three steps:

1. Ten aluminum tubes, open at both ends so that they should fill with water, were inserted in the core in the axial direction, and the period of the reactor with all control rods fully withdrawn was measured.

2. The aluminum tubes were withdrawn and the period of the reactor with all control rods fully withdrawn was again measured.

3. The ten aluminum tubes were sealed on the lower ends and the tubes, with air enclosed, were inserted in the reactor in the same positions they had occupied in the first run. The period was again measured with all control rods fully withdrawn.

The results of the experiment using a cylindrically loaded core having 1863 fuel rods and two control rods are shown in Table XVIII.

TABLE XVIII

Void Coefficients in Yankee 3:1 W/U Ratio CRX Core

Period Changes	Period	Reactivity
Clean Core	59 secs.	.100% $\Delta k/k$
Core with ten aluminum thimbles filled with water	96 secs.	.070% $\Delta k/k$
Core with ten aluminum thimbles filled with air	274 secs.	.029% $\Delta k/k$

The void coefficient for aluminum and for air are as follows:

Al	0.49% $\Delta k/k$ per % void
Air	0.47% $\Delta k/k$ per % void

Temperature Coefficient

The temperature coefficient of reactivity was measured by determination of the period change associated with a temperature change. Table XIX illustrates the results.

TABLE XIX

Temperature Coefficient of Yankee 3:1 W/U Ratio CRX Core

T°C	$\frac{\delta p}{\delta T}$ %/°C
50.5	-0.65×10^{-2}
57.7	-1.16×10^{-2}
69.3	-1.27×10^{-2}
71.7	-1.42×10^{-2}

Control Rod Worth

The total worth of a centrally located Ag-Cd (70% Ag-30% Cd) control rod was determined to be 7.3% $\Delta k/k$. The method used was to determine the differential worth (Δp/cm) at various points along the length of the rod (Figure 49) and then to integrate that curve to derive the integral curve (Figure 50). The total worth of a centrally located Ag-Cd control rod was compared with that of a Ag-In-Cd (80% Ag-15% In-5% Cd) control rod in the same position. It was determined that the Ag-In-Cd rod has a worth 1.02 times that of a Ag-Cd rod.

Peripheral Fuel Rod Worth and Migration Area

Figure 51 illustrates inferred variation of fuel rod worth as a function of the total number of fuel rods in the core. This information will be used to determine the migration area. However, this work is still in progress and the final results will be reported at a later date.

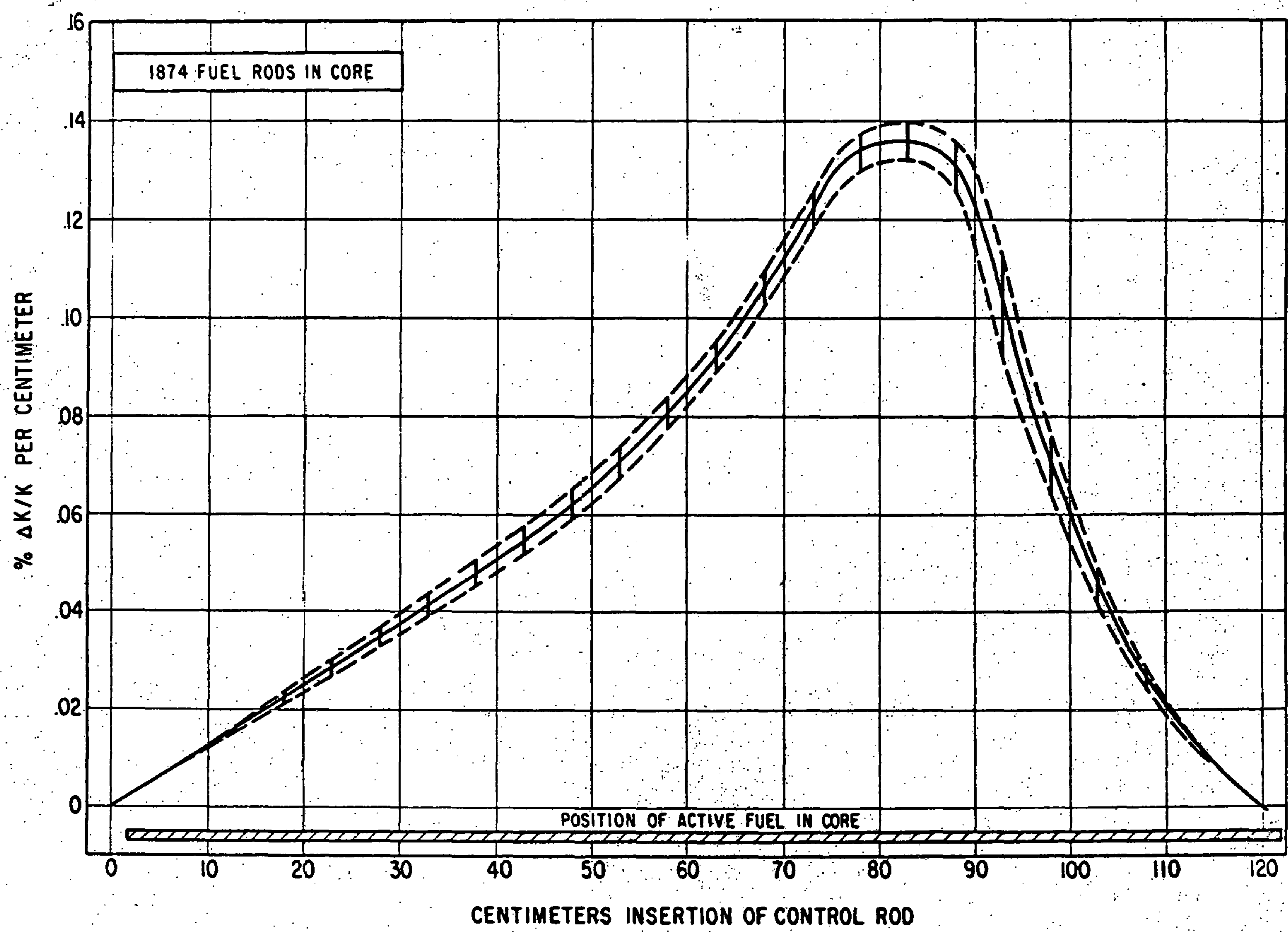


Figure 49 Differential Worth of a Centrally Located Silver-Cadium Control Rod

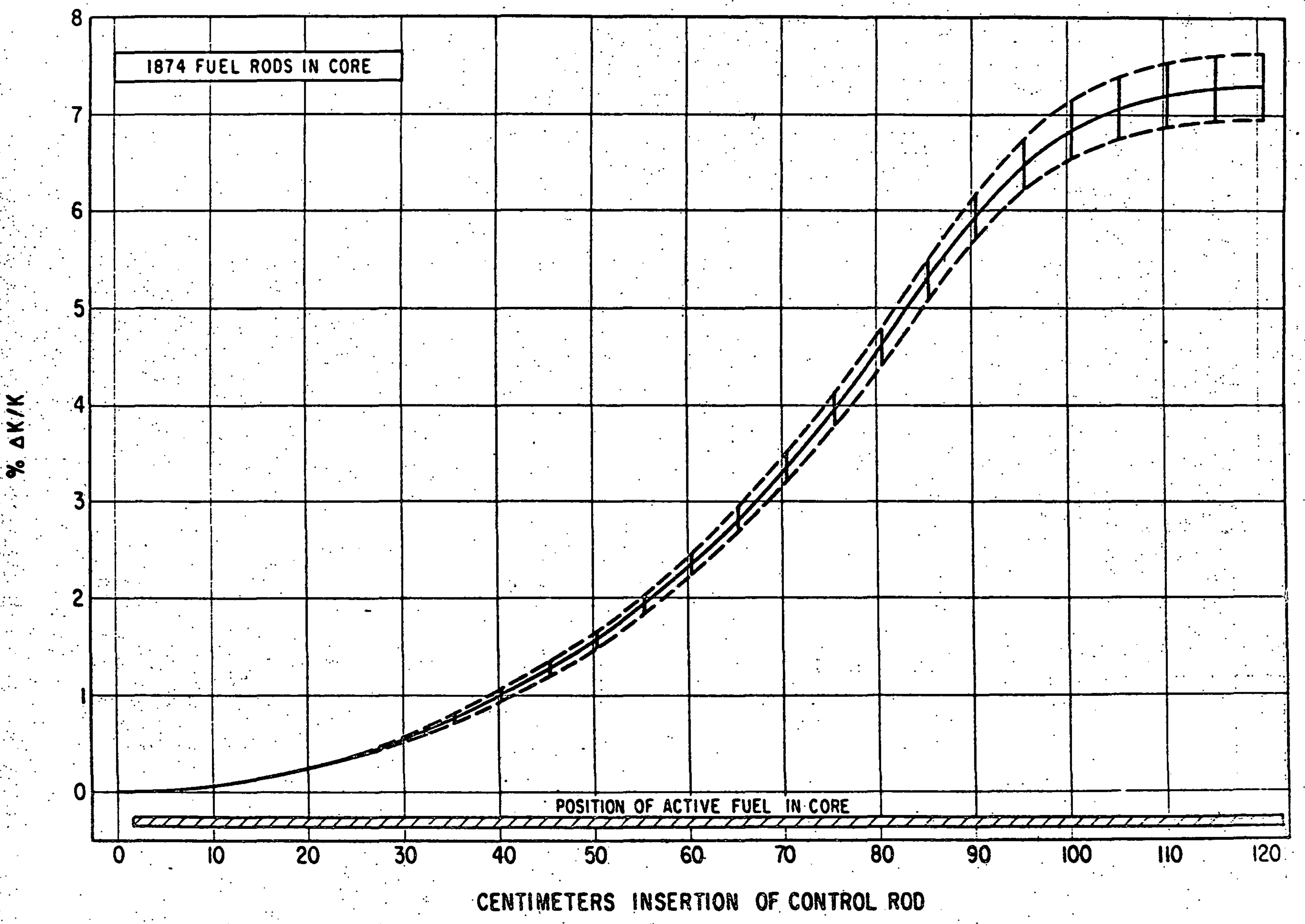


Figure 50 Integral Worth of a Centrally Located Silver-Cadium Control Rod

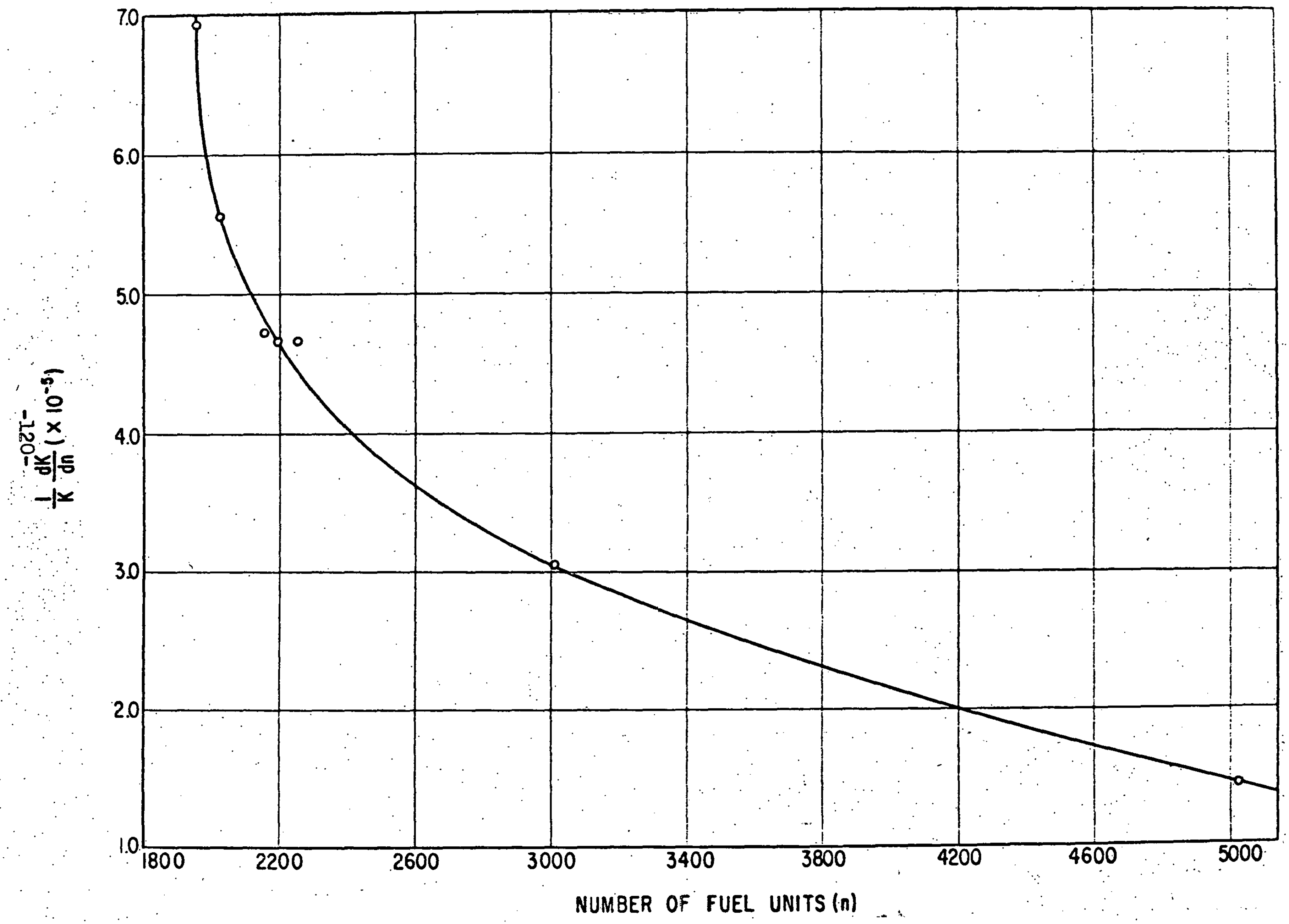


Figure 51 Inferred Peripheral Fuel Unit Worth

11.0 RADIATION DAMAGE EXPERIMENTS

Mechanical and Thermal Design Section: A. G. Thorp, Manager

This project involves the design and procurement by ⓌAPD of a pressurized water loop for in-pile irradiation tests in the MTR; the installation and operation of the in-pile loop; the performance of radiation damage experiments to demonstrate irradiation stability of Yankee core elements and post-irradiation examinations.

11.1 Design and Fabrication of the In-Pile Test Loop

M. Vogel C. Dishman R. Berringer R. Goldstein

Design of all the in-pile test loop components was completed, and all drawings were approved. Arrangement and fabrication drawings of the loop primary system piping were received from the vendor and approved by ⓌAPD. Welding specifications furnished by the fabricator are now being reviewed by ⓌAPD.

A topical report "WCAP-4 In-Pile Test Loop" was prepared as part of the loop pre-installation requirements as established by the MTR organization. The report included a description of the loop, startup and operational procedures, component and material descriptions, and the recommended procedures in case undesirable operating conditions develop.

On June 20, 1958, the loop fabricator indicated that procurement of material is on schedule, and that he feels confident the scheduled completion date of August 12, 1958 would be met. Following completion on fabrication, the loop will be given a 120 hour operational test before shipment to the MTR.

11.2 Performance of Radiation Damage Experiments

Post-Irradiation Examination of WCAP-1 and WCAP-2 Specimens

The first group of samples that were removed from the MTR and reported during the first quarter of 1958 were WCAP-1-1, 1-3, 1-5, and 2-1. The following work was performed on the second group of samples (WCAP-1-2, 1-4, 2-3 and 2-6) removed from the MTR during the second quarter of 1958:

a. Disassembly of the capsule
b. Removal of end cap flux monitors
c. Identification of flux monitors and their shipment to the Nuclear Science and Engineering Company
d. Visual examination of bundles and ferrules

The following post-irradiation work was performed on WCAP-1-4 and 2-3 specimens only:

a. Disassembly of bundles into individual tubes
b. Measurements of the diameter and length of all tubes in the bundle assemblies
c. Visual examination of all tubes.
d. Puncturing and collecting fission gases at room temperature in order to determine the gas pressure and to perform mass spectrographic analyses of each gas sample. The samples will be sectioned for examination of the fuel following the gas analyses.

A metallographic examination of the cracked ferrule brazes on the WCAP-2-1 specimens was completed. Hair-line cracks were seen at approximately 6X magnification. Brazes that were cracked through showed little evidence of diffusion bonding. The brazing specifications have since been revised. Conformance with the revised brazing specifications should improve the quality of the brazed joints in subsequently brazed specimens.

Specimens of the original WCAP-1 and -2 experiment that remain in the Materials Testing Reactor, WCAP-2-2, 2-5, 2-4, and 2-7, have received the following exposure at the end of Cycle 105, June 16, 1958.

Capsule Identification 5.4% Enrichment	Accumulated Unperturbed NVT
2-2	4.8×10^{20}
2-4	5.3×10^{20}
2-5	5.0×10^{20}
2-7	4.8×10^{20}

WCAP-4 In-Pile Loop Experiment

Minor modifications are continually being made in the in-pile loop experiment as a result of discussions with MTR personnel and problems that evolve during the planning stage of the experiment. Further modifications can be expected. An important technical modification is the substitution of mock samples for the actual samples during the check-out periods of the loop at the Lummus Company and at the MTR test site. The actual samples will thus not be subjected to handling operations that they otherwise would receive during loop test operations. After the Lummus Company has completed the operational testing and MTR completes their operational testing, the actual test samples will be inserted in the in-pile loop system just prior to operation in the MTR.

12.0 LONG LIFE FUEL EXPERIMENTS

This project was originally intended to cover the insertion of a UO_2 stainless steel fuel element bundle in a high temperature water cooled power reactor for burnup of up to 20,000 MWD per ton, in order to establish the radiation stability of the fuel assembly.

A suitable test facility has not become available. As a result, greater emphasis is being placed on the process water and in-pile irradiation testing programs under Project 11.0. No work was performed under this project during the second quarter of 1958.

(This project will be eliminated from future reports due to the unavailability of a suitable test or power reactor.)

APPENDIX

Section A - Design Data

Yankee Atomic Electric Company Reactor

I. Mechanical and Thermal

(All dimensions are for the "cold" core unless otherwise specified.)

	First Core
Total Heat Output, Btu/hr	$1,338 \times 10^6$
Total Heat Output, MW	392*
Coolant Flow:	
Total Rate, lbs/hr	37.8×10^6
Heat Transfer Rate, lbs/hr	34.0×10^6
Flow Area in Fuel Rod Cross Section, ft^2	15.4
Velocity Along Fuel Rods, ft/sec	14.0
Pressure, psi	
System Pressure, Nominal	2,000
System Pressure Maximum	2,150
System Pressure, Minimum	1,850
System Pressure, Design	2,500
Total Drop Across Vessel	29.7
Drop Across Core	13.7
Heat Transfer	
"Active" Surface, ft^2	15,400
Average Flux, Btu/hr-ft^2	87,000
Maximum Flux, Btu/hr-ft^2	450,000
Average Film Coefficient, Btu/hr-ft^2 - °F	6,050
Burnout Flux, Btu/hr-ft^2 (Jens & Lottes Correlation)	1.47×10^6
Thermal Conductivity of UO_2, Btu/hr-ft °F	1
Temperature, °F	
Coolant in the Primary Loop, Average	514
Coolant in the Fuel Bearing Portion of the Core, Average	516
Coolant Rise in Core, Average	33
Coolant Rise in Vessel, Average	30
Film Drop, Average	14.4
Surface, Maximum	651
Center of Fuel, Maximum	4,200
Outlet of Hot Channel	603
ΔT_m at Heat Exchanger	37
Steam Temperature	475
Steam Pressure, psia	540
Coolant at Inlet to Vessel	499

* - If center of fuel temperature of 4500°F is used, the total heat output can be raised to 482 MW. Lifetime is reduced proportionately.

II. Nuclear

	First Core
Total Heat Output, MW	392
Lifetime, hr.	10,066
Burnup, MWD/Metric Ton Uranium, avg.	8,208
Core Average Diameter, in.	75.1
Core Active Height (Between Fuel Ends), in.	90

Core Materials (Between Fuel Ends)

	Volume, in^3	Wt., lbs.
Fuel (10.07 gm/cm^3)	138,000	50,200
Water (.7814 gm/cm^3)	198,000	5,610
Zirconium (6.56 gm/cm^3)	12,600	2,980
Stainless Steel (8.03 gm/cm^3)	44,600	12,900
Void (Gap between clad & pellet)	4,900	-
TOTAL	398,900	71,690

B^2, geometric, cm^{-2} (Assumes reflector savings of 7.5 cm)	7.128×10^{-4}
Volume Ratio, H_2O/U (18.7 gm/cm^3)	3.03
Initial Enrichment (Atom percent)	3.02
Final Enrichment (Atom percent)	2.29
Resonance Escape Probability, p	0.735
Fast Fission Factor, ε	1.043
Age, τ, cm^2	59.5

Typical performance data and nuclear parameters based on an initial enrichment of 3.02% for the first core:

Initial Conversion Ratio	0.691
k_{eff} (Cold, clean)	1.191
(Hot, equil. Xe)	1.065
(Hot, clean)	1.110
Pu Obtained, kg.	99.5
Pu -240 Cont. (Atom Percent)	9.14
Fraction of Energy from:	
U-235	0.728
U-238	0.067
Pu	0.205
Nominal Shutdown Concentration of Chemical Poisons, Gms. Natural Boron/Liter of H_2O	1.2
Moderator Temperature Coefficient $\frac{1}{k}\frac{\partial k}{\partial T}$/°F at 516°F	-2 to -4×10^{-4}
Fuel Temperature Coefficient $\frac{1}{k}\frac{\partial k}{\partial T}$/°F at 516°F	-1 to -4×10^{-5}

Yankee Atomic Electric Company Reactor (Cont'd)

	First Core
Hot Channel Factors:	
F_Q, Heat Flux	5.17
F_θ, Film Drop	7.36
$F_{\Delta T}$, Coolant Rise	3.36
General:	
Total Core Area, ft^2	30.8
Equivalent Core Diameter, ft.	6.26
Length-to-Diameter Ratio of Core	1.2
Length-to-Diameter Ratio of a Flow Channel	260
Fuel Rod	
Outside Diameter, in.	0.337
Tube Wall Thickness, in.	0.021
Number Per Cross Section	23,218
Fuel Length per Rod, ft.	7.5
Rod Lattice, in.	0.425
Rods Per Fuel Assembly	305 & 306
Total Number of Fuel Assemblies	76
Fuel Pellet	
Diameter, in.	0.290
Height, in.	0.60
Fuel Assembly	
"Active" Length, in.	90
Overall Length, in.	120.75
Sectional Dimensions, in.	7.61 x 7.61
Weight, lbs.	1,000
Control Rods	
Active	24
Shim	8

(The following dimensions are for the "hot" core)

Fuel Rod	
Outside Diameter, in.	0.339
Rod Lattice, in.	0.427

Section B - Information Availability

Westinghouse Atomic Power Department

During the previous periods of the R&D Program, the availability to Ⓦ APD of documents and other sources of technical information from AEC facilities and contractors, pertaining to Nuclear Power Reactors was reported in the following:

1. The first year of the Contract (June 6, 1956 to June 30, 1957), was covered in detail bv Report YAEC-30 dated September 30, 1957, entitled "Information Availability Report".

2. YAEC-44, Quarterly Progress Report, gave data on document availability for the period from July 1 to September 30, 1957.

3. YAEC-52, Quarterly Progress Report, advised of the availability of documents and reports during the October 1 to December 31, 1957 quarter.

4. YAEC-65, Quarterly Progress Report, advised of the availability of documents and reports during the January 1, 1958 to March 31, 1958 quarter.

The availability of reports and documents during the second quarter of 1958 has continued to improve. Most reports of value and interest to the R&D Program are received under the "Standard" and "Positive" distribution systems. Other reports, when ordered through TISE, are received promptly if available at TISE, otherwise, a prompt and informative report with definite information is received.

The quantity and type of reports and microcards received by Ⓦ APD as a result of being on the "Standard" and "Positive" distribution lists during the second quarter of 1958 were:

Month 1958	Unclassified	Classified
April	686	12
May	1748	21
June	942	6

The number of documents not on the "Standard" or "Positive" distribution lists which were requisitioned and received during the second three months of 1958 were:

Month 1958	Ordered	Received
April	25	10
May	17	9
June	12	5

On reports which were previously ordered, but not received, definite information was furnished promptly, such as:

4 - Will be placed on "Standard" distribution list.

16 - Attempts will be made to obtain the report.

It has been the opinion of Westinghouse APD that considerable information developed for the naval pressurized water reactors and still classified would be of value to the Yankee Research and Development Program.

The type of information needed is that covering maintenance, operation, water chemistry, radiation, reactor physics and other experience and tests pertaining to pressurized water reactors. The information released for the PWR Project is helpful but, at the present state of the art, additional information, particularly from plants with long experience records, can be most helpful and could reduce research and development costs by eliminating duplication of effort.

Yankee Atomic Electric Company

The quantity of documents and replies to requests received by the Yankee Atomic Electric Company during the second quarter of 1958 has continued to show improvement. The availability of information applying to nuclear power reactors remains satisfactory.

Summaries of documents received as a result of requests, requested, and received under "Standard" distribution but not requested are:

Summary of Classified Documents Received - as of June 30, 1958

Total Documents Received	372	
Total Documents Retained		166
Total Documents Returned		206

Summary of Requested Unclassified Documents - as of June 30, 1958

Total Documents Requested	320	
Documents Received and Retained		270
Documents Received and Returned (on loan)		5
Documents Not Yet Received (Include documents for which some notification was received i.e. not at TISE, no copies available, etc)		33
Documents Denied		12

Summary of Unrequested Unclassified Reports Received on Standard Distribution - as of June 30, 1958

Total Documents Received	408	
Documents Retained		137
Documents Returned		271

Visit Requests to AEC Facilities

Visit requests for APD personnel were approved or denied as follows:

Visit to	Requisitors	Date of Visit	Approved/Denied
Bettis	J. Baron, A. R. Collier	4/11/58	Approved
KAPL	J. J. Lombardo	4/14-4/17/58	Approved
KAPL	W. H. Arnold, Jr.	4/14-4/17/58	Approved
KAPL	J. J. Lombardo	5/1-11/1/58	Approved
KAPL	McGeary and Danko	5/20-5/22/58	Denied
S1W	A. E. Voysey	6/6/58	Denied
MTR	A. E. Voysey, J. J. Lombardo	6/5/58	Approved
NYOO	Wells, Voysey, Coen, Chalupa, Graves	6/16/58	Approved

Section C - Abstracts of Trip Reports

Abstracts of trip reports written as a result of visits made by Westinghouse Atomic Power Department and Yankee Atomic Electric Company personnel to national laboratories, installations of AEC contractors and privately owned facilities are contained in this section. Only those visits made for the purpose of furthering the progress of the YAEC Research and Development Program have been included.

To: ALCO Products, Inc., Schenectady, N. Y. (April 3, 1958)
Visit made by:

R. F. Sterling - Ⓦ APD   P. B. Haga - Ⓦ APD   W. J. Miller - YAEC

The operation of the ARMY PACKAGE POWER REACTOR was described for the period subsequent to the completion of the 700 hour test which ended in the summer of 1957. Specific topics of discussion included: Coolant system integrity; startup and makeup water handling; the purification system; fission products in the primary coolant; control rod withdrawal and core burnup; nature of crud formed; dissolved oxygen and chlorine; and the test programs that were performed to study induced radioactivity and crud transport and deposition.

To: The Lummus Company, Newark, N. J. (April 9, 1958)
Visit made by:

R. P. Goldstein - Ⓦ APD

The following technical details of the WCAP-4 in-pile test loop were discussed:

1. Layout of the major loop structures
2. Prefabricated cable assemblies
3. Electrical plans and wiring diagrams
4. Review of individual components - the neutron detection equipment, the instrument cubicle drawing, the transducer, thermocouples, wells, vacuum gages and miscellaneous components
5. Numbering of weld radiographs
6. Process water pressure

To: The Lummus Company, Newark, N. J. (April 14, 1958)
Visit made by:

C. D. Dishman - Ⓦ APD

A tour of the plant and facilities was conducted by Lummus personnel. The plot plan layout of the loop equipment at the MTR was reviewed. Component design and fabrication requirements were discussed in detail. Lummus indicated that the piping drawings for the arrangement of components would be prepared by them.

Abstracts of Trip Reports (Cont'd)

To: Knolls Atomic Power Laboratory, Schenectady, N. Y. (April 14, 15, 1958)
Visit made by:

J. J. Lombardo - ⓦAPD W. H. Arnold - ⓦAPD

The modifications and additions to the MTR process water irradiation experiments were presented in detail to KAPL personnel. Cost estimates for the post-irradiation examination of all the process water samples and in-pile loop samples were discussed.

To: The Fairchild Camera and Instrument Corporation, Syosset, N. Y. (April 25, 1958)
Visit made by:

Robert M. Watkins - ⓦAPD

Information was obtained on the types of cleaning agents, and methods used in the radioactive decontamination of stainless steel. Arrangements were made to exchange information on new decontamination procedures.

To: Materials Testing Reactor, Idaho Falls, Idaho (June 5, 6, 1958)
Visit made by:

A. E. Voysey - ⓦAPD J. J. Lombardo - ⓦAPD W. J. Miller - YAEC

Discussions were held with MTR personnel concerning:

1. Installation, operation schedule and chemistry of the in-pile test loop.
2. Neutron charges for the in-pile test loop.
3. Manpower charges for operation and control of the loop during reactor operation.
4. Monitoring of the in-pile loop experiment.
5. Inspection, review and approval for the installation of the WCAP-4 experiment.
6. Flux monitoring of the process water specimens.
7. Review of the irradiation program for the six additional process water specimens.

To: The Lummus Company, Newark, N. J. (June 9, 1958)
Visit made by:

D. G. Brunstetter - ⓦAPD

The inspection facilities of the Lummus Company and its sub-contractor, the Steel Alloy and Tank Company, were inspected. Inspection procedures and methods for handling requests for changes in specifications were discussed.

Abstracts of Trip Reports (Cont'd)

To: The Babcock and Wilcox Company, Lynchburg, Va. (June 19, 1958)
Visit made by:

H. W. Graves, Jr. - Ⓦ APD
A. E. Voysey - Ⓦ APD
J. F. Chalupa - Ⓦ APD
A. G. Thorp - Ⓦ APD
C. E. Minnick - YAEC
W. J. Miller - YAEC

A technical description of the Consolidated Edison Thorium Reactor was presented by B & W personnel. Reactor physics, interpretation of critical experiments and the reactor hazards analysis pertaining to the CETR were discussed.