YAEC-65

YANKEE ATOMIC ELECTRIC COMPANY
RESEARCH AND DEVELOPMENT PROGRAM

QUARTERLY PROGRESS REPORT

FOR THE PERIOD

JANUARY 1, 1958 TO MARCH 31, 1958

R&D SUBCONTRACT NO.1 under
USAEC-YAEC CONTRACT AT (30-3)-222

APRIL 30, 1958

WESTINGHOUSE ELECTRIC CORPORATION

ATOMIC POWER DEPARTMENT

PITTSBURGH, 30

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Yankee Atomic Electric Company
Research And Development Program

QUARTERLY PROGRESS REPORT

for the period
January 1, 1958 to March 31, 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

April 30, 1958

APPROVED:

A. E. Voysey
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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 January 1, 1958 to March 31, 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 first quarter of 1958 consisted of:

1. Indications that abbreviated press feed preparations of UO_2 powder would be suitable for the automatic pressing of UO_2 pellets on a production scale.
2. Achieving compartment uniformity in disc-compartmented stretch-formed fuel tubes and the controlling, within reasonable limits, of the free space for pellet expansion within the compartments.
3. Completing the design of additional test specimens for MTR process water irradiation experiments, and of the test specimens for MTR in-pile and out-of-pile test loop irradiation experiments.
4. Developing a reference control rod program based on progressive "outside-in" rod withdrawal in order to approach uniform fuel burnout and fission product build-up in the core.
5. Concluding that stainless steel specimens suffered no detrimental attack as a result of their exposure in low boron concentration water under simulated operating conditions in a dynamic test loop.
6. Successfully fabricating a full-size strap design fuel assembly model demonstrating the welding feasibility of this method of joining tubes into bundles.
7. The obtaining of pressure drop versus flow curves experimentally for the reactor vessel model.
8. Performing instrumentation evaluations, control rod worth determinations, and material buckling measurements by loading the CRX core to various fuel configurations. A top view of a slab-loaded, critical core is shown on page 8.
9. Contracting for the procuring, fabricating, and testing of the MTR In-Pile Test Loop.

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 first quarter of 1958 were responsible for the changes made in these data from the data presented in Quarterly Progress Report YAEC-52, covering the fourth quarter of 1957.

Section-B, "Information Availability", reviews the accessibility to Westinghouse APD and Yankee Atomic Electric Company of documents and reports pertaining to Nuclear Power Reactors. The quantities and types of reports and microfilms received by Westinghouse APD during the first 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 first 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, and YAEC-52 describe and evaluate the work accomplished from the beginning of the program, June 6, 1956 to December 31, 1957.

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 and YAEC-52, for the third and fourth quarters of 1957 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 three reports also included a section containing abstracts of trip reports made under the YAEC Research and Development Program.

Figure 1. Yankee Critical Experiment Slab Loading -
Used for Initial Criticality with 1964 Fuel
Rods, each containing 1 1/4 lb. of UO_2 en-
riched to 2.7%

1.0 FUEL ELEMENT DEVELOPMENT

Chemical and Ceramics Section:
Metallurgical Section:

R. F. Sterling, Manager
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 was performed using abbreviated press feed preparation methods, with results which suggested that some of the procedures may also suffice for the automatic pressing of UO_2 pellets on a production scale. The procedures used in the laboratory trials were as follows:

1. Ball mill two hours, dry, granulate, add 0.3% Sterotex, and press.

Not only is this procedure much shorter than the presently employed production procedure, but the elimination of the organic binder addition should enable the costly and slow pre-sintering step to be eliminated.

2. Ball mill two hours with Carboxmethylcellulose binder in mill, dry, rewet to granulate, partially dry, granulate, dry, add 0.3% Sterotex, and press.

By stopping the initial drying at the proper time, the rewetting and second drying steps probably can be eliminated. This procedure then would be simpler than the presently employed production process, and may possibly be used if procedure (1), above, does not prove satisfactory. This study will be continued.

3. Ball mill two hours with polyvinyl alcohol binder in mill, dry, granulate, dry, add 0.2% Sterotex, and press.

This mixture did not feed well to the die. Segregation of binder (by migration) occurred in drying. Use of this procedure on a production scale appears feasible, and would be less costly than the present process.

4. Ball mill two hours with 0.4% Sterotex, dry, granulate, and press.

Pellets did not press or eject satisfactorily. Milling in the lubricant distributes the Sterotex throughout the powder rather than coating the granules. If sufficient Sterotex were added throughout to approach 0.2% at the surface of the granules, satisfactory results could possibly be obtained. However, the organic material content may result in high porosity with consequent low density, and would require a presintering operation.

5. Add Carbowax binder at the mill, mill two hours, dry, rewet with Carbowax solution, dry, granulate, and press.

When Carbowax only was added at the mill, the powder would not press, but when the powder was rewet with Carbowax solution before granulation, results were promising.

Green pellets produced to-date from the new procedures appeared satisfactory. The pellets will be sintered and then re-examined so that an evaluation of the different press feed preparation methods may be made.

During the above laboratory-scale trials for eliminating steps from the existing UO_2 granulation procedure, indications were noted that satisfactory granulation could be effected without rewetting and binder additions, provided the drying ball mill slurry was granulated at the proper time. In attempts to control and select this time, and to achieve uniformity of drying, the ballmill slurry was poured onto a pottery plaster batt. The drying was not uniform because the batt was not plane, and the fines migrated with the water into the plaster pores, resulting in release difficulties. Areas requiring further investigation to improve the procedure include the use of a plane level batt, determining the batt loss of UO_2 - as fines in the plaster pores, and the pickup of CaSO_4 contamination from the batt. /

Preparation of Uranium Dioxide - Water Suspensions

Stable UO_2 -water suspensions were readily prepared by adding UO_2 to rapidly stirred water until there were approximately 100 grams of UO_2 for each 19 grams of water - at higher water contents the UO_2 settled. The addition of deflocculent (in this series, about 0.15% Daxad 23, Dewey and Almy Chemical Co.) reduce to 10.4% the amount of water necessary for suspensions of less than 2000 centipoise viscosity. (The amounts of water and deflocculent necessary vary from sample to sample, finer powders requiring more of both.)

Reprocessed Uranium Dioxide Powder Analysis

At 71.5 pound lot of 2.7% enriched UO_2 powder was offered for acceptance by a UO_2 powder supplier as the final shipment of the order placed with the supplier. The lot was reprocessed after being rejected as it did not meet specifications. The following analysis of the reprocessed material resulted in a second rejection of the lot. Additional processing of the material is being performed.

	<u>Analysis</u>	<u>W.P.D. Specifications</u>
UO_2	92.4%	95%
F	1500 ppm (160 ppm after sintering)	250 ppm
Cr	100 ppm	75 ppm
Fe	1000 ppm	150 ppm
Pb	80 ppm	15 ppm
Sn	20 ppm	15 ppm
Σa	$38 \times 10^{-5} \text{ cm}^{-1}$	$30 \times 10^{-5} \text{ cm}^{-1}$
Bulk Density	1.4 g/cc	2.5 g/cc (minimum)

Fuel Material Preparation Program Review

The UO_2 fuel material preparation program was reviewed. It has been agreed that:

1. Studies proposed for developing extrusion techniques to form UO_2 fuel elements will not be performed due to the insufficient time remaining before the manufacture of the first core.

2. Studies previously considered for developing spray drying techniques to be used in preparing UO_2 powder for press feed will not be pursued due to the excessive costs that would be incurred.

1.3.1 End Closure of Fuel Rods

G. G. Lessman

End-plug welding conditions were varied to produce defects in the welding zones of fuel tube end closures so that the cause and resultant defects could be analyzed. The arc gap, welding speed, helium flow rate, and axial position of the electrode were varied independently and in combination to produce blow holes and poor penetration. Twenty of the forty defective welds produced were sectioned for weld penetration, configuration and latitude. All of the unsectioned defective welds were easily repaired. A topical report is planned to be issued which will contain the results obtained from the experiments.

End-Plug to Spacer Disc Gap

A required gap of 0.030" was established between the fuel tube end-plug and spacer disc to prevent the formation of blow holes from the increase in air pressure resulting from the welding operation. The frequency of blow-holes decreased as the 0.030" gap was approached, with no blow-hole occurrence at the 0.030" spacing.

1.3.2 Joining Fuel Bundles into Assemblies

P. P. King

J. R. Dazen

M. D'Amore

Brazed Joint Studies

AISI 304 stainless steel "T" joints containing brazed fillets of Microbraz 10, 20, 30, 45, 50 and Coast Metals NP were sectioned after 2925 hours (approximately 122 days) in 650°F neutral degassed, demineralized water. Results are as follows:

1. Microbraz 10 (Ni / 10 P) had insignificant attack as indicated in Figure 2.
2. Microbraz 20 (Ni / 12 Si) showed some evidence of localized pitting.
3. Microbraz 30 (Ni / 9 Si / 18 Cr) was relatively unaffected.
4. Microbraz 45 (Ni / 6 P / 5 Si) had selective attack of the eutectic.
5. Microbraz 50 (Ni / 10 P / 13 Cr; reference braze alloy) had insignificant attack as indicated in Figure 3.
6. Coast Metals NP (Ni / 4 P / 12 Si / 26 Fe / 4 Mo) had some evidence of minor eutectic pitting.

Further testing of the "T" joints in 650°F neutral degassed, demineralized water is continuing to verify the tentative selection of Microbraz 50 as the reference brazing alloy.

Figure 2. Microbraz 10 (10 P 90 Ni) Brazed Fillet after 122 days in 650°F Neutral Water.

Figure 3. Microbraz 50 (9 P, 11 Cr, balance Ni) Brazed Fillet after 122 days in 650°F

Work is progressing on improving the strength of brazed joints. In general, the strength of the joints increase as the time at temperature is increased. Typical data for the two reference braze materials are tabulated as follows:

Nicrobraz 50 Slurry

<u>Time at 1850°F</u>	<u>Average Fracture Load</u>
1 hour	300 lbs.
2 hours	400 lbs.
4 hours	803 lbs.

Electroless Ni-P (Ni + 9 - 11 P):
0.001" Plated on the Spacer Tubes

<u>Time at 1850°F</u>	<u>Average Fracture Load</u>
1 hour	Approx. 200 lbs.
4 hours	525 lbs.
8 hours	900 lbs.

Figure 4 (50X) shows the diffusion bond formed during the eight-hour annealing period. This annealing is responsible for the high joint strength. The braze penetration into the base metal was slight (less than 0.003") in all cases.

Thirty-Six Tube Bundle Sub-assembly Brazing and Testing

A 36-tube, 95" long ferrule-spaced fuel tube bundle sub-assembly was horizontally brazed at the Ferrotherm Company. Some tubes were filled with Al_2O_3 pellets and others were filled with steel pellets to simulate loaded tubes. The duration of the brazing cycle amounted to 35 minutes at 1840°F using Nicrobraz-50 in a dry hydrogen atmosphere. Thermocouples were located at the ends and middle of the sub-assembly. Figure 5 shows the fixture and spacer grids while Figure 6 shows the assembled and fixtured sub-assembly. The white run streaks on the fixture are from excess braze stop-off. Figure 7 shows the brazed bundle. Inspection after brazing indicated a maximum bow of 0.030" along the length with a minimum tube to tube spacing of 0.041" (design spacing 0.088"). This poor spacing was infrequent and could be greatly improved in future bundles by an improved pattern of distribution of the spacer strips. Ten joints in the front end of the sub-assembly were fractured due to rough handling during shipment. However, fracture of the joints would not have occurred if diffusion bonded joints had been obtained during brazing. The thermocouple shorted out at the front and rear end and temperature readings were not obtained. From the surface appearance of the tubes, there is evidence that this end was somewhat cold during brazing which would explain the brittle joints.

Figure 4. Diffusion Bond Formed During 8-Hour Anneal
at 1850°F

Braze material: chemically deposited 0.001" layer of Ni-P alloy. Note single-phase alloy formed across interface of tubes as a result of depletion of phosphorous from the eutectic (dark structure). Breaking load (900#) of this joint approaches tensile strength of stainless steel.

Figure 5. Brazing Fixture for 36 Bundle.
First Layer of Tubes and Vertical
Spacer Strips are in Position.

Figure 6. Completely Fixtured Bundle Prepared
for Furnace Brazing.

Figure 7. A 36 Tube Yankee Bundle Brazed with Microbraz 50.

The sub-assembly was then sectioned and destructively tested after a complete dimensional survey. The following results were obtained:

1. Both ends of the sub-assembly were brazed at a lower (estimated 1750 to 1800°F) than desired temperature (1850°F) as evidenced by the brittle fractures at the ends.
2. The spacers and fuel tubes were not in intimate contact in some cases which resulted in excessively brittle joints. All of these defects can be minimized by raising the brazing temperature, holding at temperature longer, and reducing the thickness of the fixture grids to give approximately 0.004" clearance.

Fuel Tube Compartmentation

A. Stretch Forming of Tubes

A study of ovality change and rate of I.D. decrease during stretching of 50" long AISI 304 stainless steel weld-drawn tubing with an 0.014" wall was completed. The I.D. of three tubes was measured for ovality and average diameter at each 1% elongation interval for a total elongation of 5%. The data are tabulated in Table I.

TABLE I

Inner Diameter and Ovality Changes for 0.014" Wall
Thickness AISI 304 Weld-drawn S/S Tubing During Stretching

Tube Elongation %	Tube Specimen Number					
	<u>13</u>		<u>16</u>		<u>18</u>	
	Diameter Reduction per 1% Elongation (mils)	Maximum Ovality (mils)	Diameter Reduction per 1% Elongation (mils)	Maximum Ovality (mils)	Diameter Reduction per 1% Elongation (mils)	Maximum Ovality (mils)
0	0	1.74	0	1.56	0	1.70
1.0	1.11	1.93	1.24	1.80	1.19	1.80
2.0	1.62	1.66	1.52	1.77	1.51	1.59
3.0	1.45	1.42	1.50	1.62	1.57	1.82
4.0	1.44	1.65	1.40	1.49	1.17*	1.58
5.0	<u>1.45</u>	1.38	<u>1.44</u>	1.42	<u>1.64</u>	1.69
Total Dia. Reduction	7.07		7.10		7.08	

* Low figure due to insufficient stretching

B. Stretch-Forming of Tubes onto Discs

Three AISI 304 stainless steel, 0.014" wall, weld-drawn tubes containing accurately machined stainless steel bars to space the compartmentation discs were evaluated for uniformity of compartment lengths after stretching. Each tube had been horizontally stretched 2% with a retaining ram in position to hold the pellets in a column without axial gaps, followed by a 2% stretching operation without the ram. Data for the three tubes are tabulated in Table II.

TABLE II

Compartment Uniformity in Stretch-Formed Tubes

Compartment Number	Tube Number					
	<u>12</u>		<u>14</u>		<u>17</u>	
	Stretched Compartment Length (inches)	Expansion Space (inches)	Stretched Compartment Length (inches)	Expansion Space (inches)	Stretched Compartment Length (inches)	Expansion Space (inches)
1	16.295*	0.295*	8.134	0.134	8.138	0.138
2			8.130	0.130	8.135	0.135
3	8.140	0.140	8.130	0.130	8.140	0.140
4	8.140	0.140	8.135	0.135	8.135	0.135
5	8.140	0.140	8.125	0.125	8.138	0.138
6	8.130	0.130	8.123	0.123	8.133	0.133

* Double Compartment (16")

The data indicate that a fuel rod can be compartmented, and that free space for pellet expansion within the compartment can be controlled to reasonable limits by the present stretching method. The theoretical expansion space in an 8" long compartment stretched 2% with the retaining ram and 2% without the retaining ram should be 0.160" ($2\% \times 8" = 0.160"$). Actual practice indicates that longitudinal springback of the tube occurs when the tension on the tube is released upon completion of the stretching operation and results in an expansion space per compartment which is less than the theoretical figure as shown in the reported data.

C. Spacer Disc Brazing

Several experiments were run to continue evaluation of brazed disc-tube joints. The object has been to determine the least amount of Ni-P deposit on the cylindrical face of a disc required to produce a sound joint at the disc-tube interface. Results of four tests are shown in Table III. Even though the data shown in Table III indicated that three of the specimens passed a helium leak test, the integrity of all joints was poor.

TABLE III

Evaluation of Four Brazed Disc-Tube Joints

Specimen Mark	Ni-P Layer (Radial) (Inches)	Red. of Tube on Disc Diameter (Inches)	<u>Brazed Joint Evaluation</u>		
			Visual	Helium Leak Test	Metallo- graphic
32	0.0002	0.0025	sound fillet	no leak	voids in joint
33	0.0002	0.0025	sound fillet	no leak	voids in joint
34	0.0002	0.0015	starved fillet	leak	voids in joint
35	0.0002	0.0015	starved fillet	no leak	voids in joint

Figure 8 (a) shows a portion of a typical brazed cross section with small "black islands" indicating voids in the joint. Additional voids and a starved joint are shown in Figure 8 (b) which represents an area of a longitudinal view of the brazed disc-tube joint. Results obtained thus far indicate that Ni-P deposits of the order of 0.0005" on the cylindrical surface of stainless steel discs, positioned in cladding which had been stretch formed an additional 0.001" - 0.002" down on the discs produce metallurgically sound joints. Brazing was performed successfully in a dry hydrogen atmosphere at 1850°F for 30 minutes.

D. Spacer Disc Slippage During Brazing

An experimental program was initiated to investigate whether discs will slip during vertical-position brazing. AISI 304 stainless steel tube specimens having a 0.014" wall thickness were prepared in the following manner: One disc containing approximately 0.0005" Ni $\sim$ 9 - 11P on all surfaces was positioned approximately 1.5" from one end. After stretching, and prior to brazing, critical measurements were taken with depth and air plug gauges. A depth gauge was used to measure the short distance from one end of tube to disc. In most cases this distance was slightly greater than 1.5 inches. An air plug gauge was used to measure the tube I.D. within 1/8" from the disc before and after brazing. During each brazing operation with the specimen in a vertical position, a stainless steel rod 7.8" long rested on the disc. This rod acted as dead weight. A snugly fitting end closure was placed in the bottom end of the specimens to close the tube and to prevent a possible loose disc from dropping through. Initially, brazing was performed at 1850°F for one hour in a dry hydrogen atmosphere with approximately 0.0005" of Ni-P deposited only on the cylindrical surface of the disc. Results of this test indicated that brazing of the disc to the tube had occurred only at a few contact points around the periphery of disc. It is necessary to use about 0.0005" Ni-P on the entire disc and braze at 1875°F to insure better flow and wetting of the braze by capillary action at the disc-tube interface. Results of the brazing runs are recorded in Table IV.

Fabrication of Spacer Discs

Attempts to produce uniform discs by slicing centerless ground bar stock on a cut-off machine or a lathe were unsuccessful. The cut-off machine produced a disc of tapered thickness while the lathe tool caused deformation on the periphery of the disc which could not be removed. Two methods of manufacturing satisfactory discs were developed. The discs can be produced by machining bar stock or by powder metallurgy techniques using the procedures itemized in Table V. The discs are produced more easily and at less expense by the powder metallurgy method. The data obtained in the preparation of five experimental discs from - 325 mesh AISI 304 stainless steel powders are summarized in Table VI.

Figure 8(a)

Figure 8(b)

Figure 8. Photographs of Typical Cross Sectional (a) and Longitudinal (b) View of Brazed Disc-Tube Joint at 100X.

TABLE IV

Spacer Disc Slippage Brazing Data
(all dimensions are in inches)

Specimen No.	Tube I.D. After Stretching	Average Reduction of Tube on Disc	Measured Distance	Slippage During Brazing	Brazing Temp. (°F)	Sealed Braze Joint Helium (leak test)	Remarks
1	Min. 0.30226 Max. 0.30337	0.0002	Before Brazing 0.410 After Brazing 0.406	0.004	1850	No	Starved fillet; Brazed only at contact points
2	Min. 0.30264 Max. 0.30286	0.0002	Before Brazing 1.584 After Brazing 1.460	0.124	1875	Yes	Well-formed fillet; Good wetting
3	Min. 0.30239 Max. 0.30285	0.0004	Before Brazing 1.688 After Brazing 1.669	0.019	1875	No	Some Porosity
4	Min. 0.30304 Max. 0.30253	0.0004	Before Brazing 1.568 After Brazing 1.487	0.081	1875	Yes	Well-formed fillet; Good wetting
5	Min. 0.30272 Max. 0.30296	0.0004	Before Brazing 1.636 After Brazing 1.157	0.479	1875	Yes	Well-formed fillet; Good wetting

TABLE V

Procedure for Manufacture of Compartmentation Discs of Uniform Diameter and Flat Circumferential Surface

By Machining:

1. Machine grooves 1/8" wide x 1/8" deep along the rod 1/16" apart.
2. Centerless grind rod to remove defects and obtain correct diameter.
3. Plate rod with Ni-P braze alloy. (Ni $\sim$ 9 - 11P)
4. Heat treat to increase Ni-P strength. (1 hour at 1400°F)
5. Re-grind rod.
6. Cut-off 1/16" thick discs through grooved regions.
7. Face each side of disc.

By Powder Metallurgy:

1. Press -325 mesh AISI 304 stainless steel powders in die at 55 to 65 TSI.
2. Sinter compacted powders for one hour at 2100°F.
3. Coin sintered compact in pressing die at 100 TSI.
4. Plate disc with Ni-P braze alloy.

TABLE VI

Powder Metallurgy Data for Experimental Disc Manufacture

Specimen Number	Pressing Pressure (TSI)	Coining Press (TSI)	Sintering Temperature (°F)	Density* (%Theoretical)
1	38.5	100	1 hr @ 2100	91.3
2	44.8	100	1 hr @ 2100	90.0
3	51.3	100	1 hr @ 2100	92.5
4	57.7	100	1 hr @ 2100	93.2
5	64.0	100	1 hr @ 2100	92.7

* Density calculated from weight-volume measurements.

No binder was required to obtain good green strength in the pressed compacts. Close diametral tolerance and a flat circumferential surface are obtained on the discs by the coining operation. The final density of the discs calculated from weight-volume measurements was approximately 93% of theoretical. The discs for experimental studies are currently being produced by the machining method due to the lack of a powder die of the correct diameter.

Pressure and Composition of Gas in Sealed Stainless Steel Tubes Heated at 1875°F in Hydrogen

An apparatus was constructed for puncturing sealed metal specimen tubes within the apparatus for the purpose of measuring the pressure of the residual gas in the tubes and for collecting a representative sample of the gas for mass spectrometer analysis of gas composition. Specimen tubes of AISI 304 stainless steel were sealed in air by welding, and annealed in dry hydrogen at 1875°F for times varying from 1/2 to 24 hours. Both empty (air filled) tubes and tubes filled with crushed UO_2 pellets and UO_2 powder loaded in air, were used.

The data indicate that for the particular steel fuel tube dimensions employed, hydrogen readily diffuses through the tube wall at 1875°F and is retained within the tubes when cooled to room temperature.

For empty tubes (air filled) the theoretical expectation of at least one partial pressure of hydrogen at 1875°F within the tubes is fulfilled; this theoretical value is 23% of atmosphere total gas pressure as measured at room temperature.

For UO_2 -filled tubes (contain air as well as UO_2) residual gas pressure readings were inexplicably low (3 to 4% of an atmosphere) although most of the gas was hydrogen. The data offer no clue why the hydrogen content within the tubes is low.

Gas composition analyses after annealing show a high proportion of hydrogen under all conditions, and a rapid decrease of the oxygen and nitrogen of the original atmosphere of air. This decrease is particularly pronounced for UO_2 -filled tubes. The oxidation reaction is manifested in a dark scale on the inner wall surfaces of empty tubes, and in a bright inner wall surfaces of empty tubes, and in a bright inner wall surface for UO_2 -filled tubes where the oxidation is mainly of the UO_2 .

The gas analyses of UO_2 -filled tubes annealed for the longer times show a relatively high proportion of argon (impurity in air) as a result of the very low total quantity of gas present.

1.4 Fabrication of Fuel Elements for Critical Assembly and Irradiation Tests

J. Lombardo

D. Brunstetter

Critical Reactor Experiment - Pellet Fabrication

Shipment of 5509 fuel rods to the Westinghouse Reactor Evaluation Center was completed on February 20, 1958. Fabrication of these fuel rods was started September 1, 1957, and the first shipment of fuel rods to the WREC was made on October 25, 1957. Deliveries of fuel rods to the WREC were as follows:

October	1957	800
November	1957	2,200
December	1957	1,804
January	1958	800
February	1958	705

There are 11,425 grams of 2.7% enriched UO_2 pellets available for loading into fuel rods; these pellets will make approximately twenty additional fuel rods.

CRX Control Rod Fabrication

The fabrication of the two experimental control rods (1) Epithermal and Absorption (combination rod) and (2) silver-indium-cadmium extruded rod was completed by the C. L. Gougler Machine Company. The two control rods were delivered to the WREC for testing. The two absorber sections (stainless steel and aluminum-lithium alloy) of the combination rod are joined in such a manner that the overall control rod length is identical to that of the nine CRX silver-cadmium control rods.

Inspection and Fabrication, Critical Experimental Fuel Rods, Topical Report

The draft of YAEC-64, Inspection and Fabrication, YAEC Critical Experimental Fuel Rods, was completed. Comments and recommendation have been received and are being incorporated into the final edition of the report.

Fabrication of MTR Process Water Test Specimens

Five additional capsules similar in size to the WCAP-1 and WCAP-2 specimens have been designed to determine the performance of stretch-formed, and disc-compartmented, stainless steel clad, uranium oxide, fuel rods under irradiation in MTR process water, and are being re-evaluated.

Tube Configuration 1 - Stretch-formed until necking-down between pellets occurs; brazed compartmentation disc in the center of the tube. This design is being considered to test brazed ferrule type fuel assemblies.

- Tube Configuration 2 - Stretch-formed until the annular clearance between pellets and cladding is 1 mil $\pm$ 1/2 mil with a brazed disc in the center of the tube.
- Tube Configuration 3 - Stretch-formed until the annular clearance between pellets and cladding is 3 mils $\pm$ 0, - 1/2 mil with a brazed disc in the center of the tube.
- Tube Configuration 4 - Stretch-formed until the annular gap between pellets and cladding is 1/2 to 1-1/2 mils with a brazed disc at one end of the compartment with no zero axial gap between pellets.
- Tube Configuration 5 - Stretch-formed until necking starts between pellets, without a brazed disc and with an axial gap of 0.110" $\pm$.010".
- Tube Configuration 6 - Stretch-formed until the annular gap between the pellets and cladding is 1/2 to 1-1/2 mils without brazed discs and with an axial gap of 0.110" $\pm$ 0.010".

Two, 30" long, three-tube assemblies were designed to determine the possibility of growth occurring by a thermal ratcheting in a relatively long free-standing tube as compared to a compartmented and stretch-formed fuel tube of the same overall length. The three types of tube configuration are as follows:

- Tube Configuration 1 - A fully compartmented, stretch-formed fuel tube with a 1/2 to 1-1/2 mil pellet to clad annular gap, with an axial gap of 0.190" $\pm$ 0.010" between the pellets and discs in each compartment.
- Tube Configuration 2 - Stretch-formed rod in which heavy necking occurs between the pellets. No compartmentation.
- Tube Configuration 3 - Stretch-formed tube in which the annular clearance between pellets and cladding is 2 to 3 mils; axial gap to be determined. No compartmentation.

All fuel tube cladding is AISI 304 stainless steel with a nominal 0.021" tube wall thickness.

Fabrication of MTR In-pile Loop Test Specimens

Test specimens have been designed for location in the in-pile and out-of-pile sections of the MTR test loop. The specimen to be located in the high flux leg of the in-pile section:

Sample "A" - Consists of a 30" long, AISI 304 stainless steel, 0.021" wall thickness, stretch-formed and compartmented fuel tube containing 10% enriched UO_2 pellets.

The four samples being considered for the low flux leg of the in-pile section are as follows:

Sample "B" - An eight inch long capsule containing four metal couples: coupled Zircaloy-2 and silver-indium-cadmium; coupled AISI 304 stainless steel and hafnium; and corrosion test samples of silver-indium-cadmium and hafnium.

Sample "C" - An eight inch long capsule, containing an AISI 304 stainless steel 0.021" wall thickness, stretch-formed and compartmented fuel tube with 27% enriched UO_2 pellets.

Sample "D" - Similar to Sample "C", except the tubing will not be stretch-formed.

Sample "E" - Similar to Sample "C", except with a 0.015" tube wall thickness.

Fuel specimen designs "A", "C", and "E" are stretch-formed until the annular gap between the pellets and the cladding is between 1/2 to 1-1/2 mils. The annular clearance between the pellets and cladding in Sample "D" is between 5 and 6 mils.

The out-of-pile control specimens will be identical to the in-pile specimens, except that the fuel tubes will contain depleted UO_2 pellets.

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 design 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 first quarter of 1958.

2.2 Core Steady State Analysis

R. D. Cummings V. E. Grob A. B. deSaint Maurice
G. G. Tirellis

Nuclear Design and Fuel Cost Calculations for "Reactor Core Comparison Report"

The nuclear design and fuel cost calculations have been completed for the revised edition of YAEC-6 "Reactor Core Comparisons" and issued. This study covered five basic core designs, each analyzed with three sets of hot channel factors.

Topical Report "Analysis of Experimental Data, Reactivity of Plutonium-Bearing Fuel Rods"

An analysis of Brookhaven experimental data on depleted uranium fuel rods containing 1% plutonium has been made at WAPD. Topical Report, YAEC-57, has been issued, containing the results of the analysis.

Local Flux Peaking and Control Rod Worth Studies

A study has been completed in which the QED Program was used to investigate local flux peaking and control rod worths in the first core. Various approaches to these problems by means of quarter-core calculations and cell calculations were used. Procedures were developed for the obtaining of equivalent "smeared poison" cross sections for rods and for homogenizing control rods or followers and adjacent water channels. The convergence of iterative calculations and the use of different mesh intervals were included in the study. Contour maps of the neutron flux in the quarter core and in the control rod cells were plotted. A topical report is being prepared covering the results of the study.

Maximum Fission Product Gas Calculations

The amount of stable fission product gas (85% Xe, 15% Kr), which would escape from the pellets of the first core into the void between pellet and clad was calculated. The results given in Table VII, apply to a rod located in the maximum radial flux but represent an average over the rod length. The gas which would accumulate per inch of rod length was determined for three different times in core life.

TABLE VII

MAXIMUM FISSION PRODUCT GAS EVOLVED IN A FIRST CORE FUEL ROD
DURING NORMAL OPERATION

<u>Time of Operation</u> <u>(hours)</u>	<u>Fission Product Gas</u> <u>(10^{-5} gram-mole/inch)</u>	<u>% Release of</u> <u>Gas Formed</u>
3,300	0.46	5.1
6,700	1.7	9.8
10,000	3.9	14

2.3 Core Kinetic Analysis

H. C. Hecker
D. Hunter

J. M. Gallagher
R. E. Wolf

Loss-of-Coolant Flow Studies

A simplified representation of the loss-of-coolant flow accident using generalized parameters has been compared with the detailed results described in the preceding Quarterly Progress Report, YAE-52. This comparison has borne out the assumptions made in the development of the simplified model. Results from both studies are shown in Figure 9 for the first core. This figure shows the average coolant outlet temperature as a function of time after loss-of-coolant flow for various scram delays. The accuracy of the analog computer is demonstrated by the pair of results obtained from identical input data for certain values of scram delay. It is also seen that the results from the

Fig. 9 Average Coolant Temperature in the First Core vs Time After Loss of Coolant Flow for Various Scram Delays

detailed study using a four-section core are not substantially different from the simplified general study.

Temperature Coefficient Calculations

Calculations of the temperature coefficient for the first core have been completed and are shown graphically in Figure 10. The moderator or slow coefficient was obtained at various temperatures for five cases. These are, (1) at beginning of life, clean, (2) middle of life, without Xe, (3) end of life, without Xe, (4) end of life, without Xe and with three times the normal plutonium, and (5) end of life with Xe. It is seen that the temperature coefficient does not vary substantially during core life and that increased plutonium will result in a more negative temperature coefficient.

Contribution of Control to the Temperature Coefficient

The contribution of control to the temperature coefficient has also been calculated as a function of temperature using thermal constants obtained from the SOFOCATE code and an improved method for obtaining control rod worths. Three combinations using both boron concentration and control rods during reactor startup 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 sufficiently 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 total moderator coefficient was obtained for these three cases for the clean core at beginning of life. These results are shown in Figure 11. This set of curves gives the final total moderator temperature coefficient for the entire core under various conditions of control. As would be expected, the temperature coefficient becomes less negative as the boron concentration is increased. However, even in the case where only boron is used for control, the total moderator coefficient is $-0.42 \times 10^{-4}/^{\circ}\text{F.}$ at 300°F. Thus, above 300°F. the moderator temperature coefficient is always negative. In the case where a minimum of boron was used (control rods fully inserted), 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 boundaries set by these curves.

Fig. 10 Temperature Coefficient of "Just Critical" First Core as a Function of Moderator Temperature for Three Methods of Startup Control

Fig. 11 Calculated Values of the Moderator Temperature Coefficient vs Moderator Temperature - First Core

Void Coefficients for Various Core Conditions

Void coefficients have been obtained at two temperatures for various core conditions and are given in Table VIII.

TABLE VIII

VOID COEFFICIENTS FOR THE FIRST CORE

	<u>% $\Delta k/k$ Per Percent Void</u>	
	<u>368°F.</u>	<u>534°F.</u>
Beginning of Life	-.14	-.16
End of Life	-.14	-.18

Pressure Coefficient of Reactivity

The pressure coefficient of reactivity for the hot first core has also been obtained. Since a large part of $\frac{1}{k} \frac{\partial k}{\partial T}$ is due to the change in water density, the moderator temperature coefficient can be used to calculate the pressure coefficient. The following relation was used:

$$\text{Pressure coefficient} = \frac{1}{k} \frac{\partial k}{\partial P} = \frac{1}{k} \frac{\partial k}{\partial T} \frac{\partial p}{\partial P} / \frac{\partial p}{\partial T} \text{ psi}^{-1}$$

where p is the water density. A value of $+2.7 \times 10^{-6}$ psi⁻¹ was obtained for the pressure coefficient at 516°F. using the end of life temperature coefficient previously computed.

Preliminary Hazards Report Addendum - Core Kinetic Characteristics

An addendum to the Yankee Preliminary Hazards Report, discussing the effects of plutonium on the kinetic characteristics of the core has been prepared. This addendum includes the effects of plutonium on temperature and void coefficients, delayed neutron fraction, and prompt neutron lifetime. Table IX shows some values of the prompt neutron lifetime.

It can be seen that the entire effect of fuel changes such as uranium burnup, plutonium buildup, and fission product poisoning, cause a decrease of less than 20% in the neutron lifetime.

TABLE IX

PROMPT NEUTRON LIFETIME IN THE FIRST CORE

<u>Core Temperature</u>	<u>(sec)</u>	
	<u>Beginning of Life</u>	<u>End of Life</u>
68°F.	2.35×10^{-5}	2.10×10^{-5}
516°F.	2.46×10^{-5}	2.07×10^{-5}

2.4 Control Rod and Chemical Poison Analysis

J. J. Hindman

G. G. Tirellis

R. E. Wolf

Control Rod Worth Studies

A study of control rod worths for the two region loaded first core has been completed and the results compared to control rod worths for the uniformly loaded first core. The two region core considered utilizes an outside-in loading scheme. Another purpose of the study was to compare the total control available using 16, 24, and 32 control rods. The control rod worths were computed for rods black to thermal neutrons; the additional worth due to resonance neutron absorptions was not calculated. The WANDA code was used to determine the decrease in k_{eff} when the rods are fully inserted by replacing the control rods by an effective poison cross section added to the total core absorption cross section. The thermal neutron constants used in determining the results were obtained from the SOFOGATE code. The total worths of 16, 24, and 32 control rods are shown in Table X. These results indicate the relative worths of the outer 8 and 16 control rods compared to the inner 16.

TABLE X

COMPARISON OF ROD WORTHS FOR THE TWO REGION
CYCLED AND UNIFORMLY LOADED FIRST CORE

	<u>Two Region Loading</u>				<u>Uniform Loading</u>	
	0	16	24	32	0	24
Beginning of Life (no Xe)	k_{eff}	$-\Delta k$	$-\Delta k$	$-\Delta k$	k_{eff}	Δk
Hot	1.074	.078	.130	.149		

		<u>Two Region Loading</u>				<u>Uniform Loading</u>	
Cold		1.140	.067	.111	.127	1.192	.090
End of Life (no Xe)							
Hot		1.047	.071	.115	.136		
Cold		1.108	.061	.099	.117	1.150	.082

These results show that the control rod worths are greater in the two region core than in the uniform core. Thus, the control problem is considerably less serious in the two region core, since the control rods are worth more than in a uniform core and the k to be controlled is smaller. Reducing the total number of rods in the two region core from 24 to 16 decreases the total rod worths by 5% while increasing the number of rods from 24 to 32 will result in an added 2% in Δk . The added control does not seem to warrant increasing the total number of control rods from 24 to 32.

Alternate Startup Procedure Involving Boron Removal

An alternate startup procedure for the reactor involving boron removal during startup has been analyzed. With control rods fully inserted and sufficient boron for 5% cold shutdown, the coolant temperature is increased to a predetermined level by the heat dissipated by the coolant pumps. Control rods are then removed to bring the reactor critical. Boron is then removed from the coolant without further rod motion. The negative temperature coefficient thus requires that the temperature of the core rise to maintain the critical condition. The time necessary to reach operating temperature by the removal of the boron poison was determined for initial boron concentrations of 800 and 1600 ppm. This removal is accomplished by bleeding poisoned water and feeding clean water at a maximum rate of 125 gpm, the rate normally being determined by restrictions on the maximum rate of temperature rise. This study was performed for maximum heating rates of 100 and 200 °F/hr. Since the removal of boron for concentrations less than 80 ppm is accomplished by the ion-exchange system, this boron concentration was taken as a limiting value, resulting in two cases having an equilibrium temperature which was less than operating temperature. The results of this study are given in Table XI. The startup time is seen to vary widely under different conditions.

TABLE XITIME REQUIRED TO REACH TEMPERATURE BY BORON REMOVAL

<u>Initial Boron Concentration</u>	<u>Initial Temperature</u>	<u>Limit of Coolant Temperature Rise Rate</u>	<u>Time to Reach Temperature</u>	<u>Final Temperature</u>
<u>ppm</u>	<u>°F.</u>	<u>°F/hr.</u>	<u>Min.</u>	<u>°F.</u>
1600	200	100	190	516
1600	200	200	102	516
1600	300	100	130	516
1600	300	200	74	516
800	200	100	460 +	480 *
800	200	200	460 +	480 *
800	300	100	320 +	516
800	300	200	320 +	516

* Boron concentration reduced to 80 ppm.

+ Temperature rise rate never reaches the limiting value.

Control Rod Programming

The hot channel factors at any time in core life are determined by the spatial distribution of power at that time which, in turn, is dependent on the control rod program. The control rod placement at any time not only will specify, in a direct fashion, the neutron distribution but, in addition, the history of their application from startup will determine the spatial build-up and burnout of fissionable material and poison. Thus, selection of the control rod program which will optimize the time-dependent hot channel factor is of basic importance in reactor design.

One approach to the problem on control through core life shows promise and has provided the basis for the selection of a reference program. Figure 12 depicts rods of the first core grouped by number according to

<u>Group</u>	<u>Relative Worth</u>	<u>Actual Worth[★] ($\% \frac{\Delta k}{k}$)</u>
1	.162	2.43
2	.162	2.43
3	.207	3.11
4	.207	3.11
5	.124	1.86
6	.138	2.07

★ Based on an estimate of 15% for total worth.

Fig. 12 Reference Control Rod Program: Rod Group, Identifications and Worths

their worths and their function in the reference program. Listed on this figure are the relative worths of the rods and their absolute worths in the hot core based on an assumed value for total worth of 15%.

From preliminary analysis, optimum hot channel factors throughout core life should be obtained by a program based on progressive "outside-in" rod withdrawal. It is estimated that shortly after startup, the core can be controlled by fully inserting groups 2 and 4 and partially inserting groups 5 and 6. Groups 2 and 4 are equivalent in worth to Groups 1 and 3. In the course of the program, Groups 1 and 3 would be frequently substituted for 2 and 4, respectively, in order to obtain more uniform fuel burnout and fission product build-up in the central part of the core. With this understanding, only Groups 2 and 4 are referred to in Table XII, which summarizes the step program assumed for the analysis in terms of k excess.

TABLE XII

REFERENCE CONTROL ROD PROGRAMMING FOR THE FIRST CORE

<u>Step No.</u>	<u>Rod Group Configuration at Beginning of Step</u>	<u>k Excess at Beginning of Step</u>
1	2 and 4 fully inserted 5 and 6 halfway (avg position)	6.5%
2	2 and 4 fully inserted	5.5%
3	2 fully inserted 4 halfway (avg position)	4.0%
4	2 fully inserted	2.4%
5	2 halfway (avg position)	1.2%

2.5 Critical Experiment Planning and Analysis

Critical Experiment Hazards Report.

All calculations for the Hazards Summary Report for the 2.23:1 water-to-uranium volume ratio CRX core have been completed and the report has been written. The reactor accidents studied for this report have been analyzed on the Westinghouse APD analog computer.

The most rapid ramp increase in reactivity which can occur in the CRX core is continuous withdrawal of all control rods simultaneously at the design rate of 6 inches/min. ($5 \times 10^{-4} \Delta k$ per sec.). For such an

accident to occur, both period and power scrams must be assumed inoperative. In a fully loaded core containing 7200 lbs. of UO_2 , this accident results in a total energy release of 150 MW-sec before the excursion is terminated by the operation of the negative fuel and moderator coefficients. Since the average fuel temperature at the center of the core does not go above 620°F ., no core damage is anticipated.

A step increase of reactivity corresponding to the sudden removal of the central control rod has been considered as the maximum credible accident which can occur at the critical facility. The largest possible reactivity step occurs with a core which is initially shut down by 2% with a single central control rod. This core contains 5700 lbs. of UO_2 . Since a central control rod has a worth of 3.3% in this core, this results in a step increase in reactivity of 1.3%. Such an accident would result in an energy release of 220 MW-sec. and an average fuel temperature at the center of the reactor of 1250°F . Since some of the steel may melt if vapor blanketing of the fuel rod occurs, some release of radioactivity may occur. This problem is currently being analyzed. In any event, the maximum credible accident is not as serious as that presented for the 3/1 water-to-uranium volume ratio in YAEC-31, "Yankee Critical Experiments - Hazards Summary Report". In that case, no undue hazard to the public was incurred as a result of the maximum credible accident.

Critical Mass Calculations

Evaluation of computational methods on the basis of experimental results proceeded with the calculation of the critical mass of the 3/1 water-to-uranium volume ratio core using a modified approach to the WANDA code. The core region was subdivided into concentric rings representing fuel and control rod follower regions. Calculations predicted 2038 fuel rods for criticality, with a critical radius of 29.07 cm and a reflector savings of 8.46 cm. This result overestimated the critical mass by 176 rods or 9.5% for a comparable experimental configuration. As the results are closer to experimental results than previous WANDA calculations in which the control rod followers were homogenized with the fuel, it is recommended that this concentric ring approach be used in future one dimensional calculations.

Buckling and Reflector Savings Calculations

Axial and radial flux measurements obtained from the CRX for the 3/1 core are being processed and interpreted. These experimental flux measurements are being obtained by irradiating indium foils in the core and measuring their activity. The counting rates are corrected for

counter dead time, background radiation, counter efficiency and radioactive decay of foils during counting. The axial and radial data are being fitted to cosine and Bessel J_0 functions respectively, thereby determining the best values of the arguments of these functions. This in turn yields the axial and radial bucklings of the core which can be compared to the calculated value. The reflector savings can also be experimentally verified from the buckling and dimensions of the critical assembly. The COFIT program for determining a least square fit of the function $A \cos [B(x - C)]$ to experimental data was obtained from IBM and checked out. Results and comparisons of axial buckling and reflector savings calculations from data obtained at WREC for both hand and machine methods are given in Table XIII.

TABLE XIII

CALCULATION OF BUCKLING AND REFLECTOR SAVINGS

<u>Method of Calculation</u>	<u>Run No.</u>	<u>Direction of Traverse</u>	<u>Buckling B^2 (cm⁻²)</u>	<u>Reflector Savings one side (cm)</u>
Hand	Preliminary	axial	0.00062	2.22
COFIT	Preliminary	axial	0.00046	12.13 [±] 1.56
Hand	64-III-7	axial	0.00063	1.50
COFIT	64-III-7	axial	0.00058	4.31 [±] 0.51
Hand	50-II-32	radial	0.00546	5.27
Hand	60-III-3	radial	0.00377	11.91

The errors indicated for machine calculations are half the standard deviations of the extrapolated core height.

Although the above results are preliminary the discrepancies between the hand and the machine calculated values are ascribed to the fact that the smaller number of significant figures carried in hand calculations results in greater error when differences between numbers of the same order of magnitude are taken.

A modification of the COFIT program has been made to permit least squares fitting of radial experimental data to a function of the form $AJ_0 [B(x - C)]$. This program (JOFIT) is undergoing a pre-operational check on the IBM-704 computer. It will facilitate the analysis of radial flux distributions obtained experimentally in cylindrical critical assemblies.

Relative Worths of Silver-Cadmium and Silver-Indium-Cadmium Control Rods

The experimental comparison between a 70% Ag - 30% Cd control rod and an 80% Ag - 15% In - 5% Cd control rod has indicated the two are equal in worth. Since the rod with 15% indium was expected to have a worth about 15% greater than the Ag-Cd-rod, consideration was given to the blackness of the two rods in the epithermal range. The ratio of current to flux at the surface of the rod κ , may be used as a measure of rod worth, where a black rod has a value of 1/2. Computed values of κ as a function of energy are listed in Table XIV for energies greater than 0.3 ev; below this energy both rods are black.

TABLE XIV

RELATIVE WORTHS OF AG-CD AND AG-IN-CD CONTROL RODS AT VARIOUS NEUTRON ENERGIES

<u>E(ev)</u>	<u>Ag-Cd</u>	<u>Ag-In-Cd</u>
0.30	0.5	0.482
0.35	0.5	0.444
0.40	0.494	0.404
0.45	0.473	0.377
0.50	0.437	0.359
0.55	0.393	0.348
0.625	0.333	0.342

The Ag-In-Cd control rod is thus seen to be less black in this range than the Ag-Cd rod. In process at this time is the computation of a Wigner-Wilkins spectrum using the SOFOCATE code to find energy averaged values of κ . The increase in resonance absorptions in the rod containing indium may be offset by the decrease in its effectiveness in absorbing low energy epithermal neutrons.

Preliminary Calculations for CRX 4:1 Hazards Report

A simulation procedure has been specified for use in the analysis of ramp insertions of reactivity in connection with the 4/1 water to uranium volume ratio Critical Reactor Experiments. The analog computer setup is being checked, and initial test runs are in progress. Ramp insertions of reactivity corresponding to maximum control rod motion and water insertion rates will be used as inputs to determine the kinetic behavior of the core. From this behavior maximum core temperatures and time integrated power will be obtained.

2.6 Irradiation Experiment Design and Analysis

Maximum Clad Surface Heat Flux Versus Enrichment in WCAP-4 Specimens

Calculations were completed on the variation of the clad surface heat flux with enrichment of pellet samples intended for the in-pile test loop at the MTR. These calculations are based on pellets with a 0.290 inch diameter and on cladding 0.021 inch thick. Typical operating heat fluxes are being simulated in these in-pile tests.

Figure 13 depicts the results of these calculations for samples placed in both the high and low flux legs of the pressurized loop. Contributions to the thermal neutron flux from fissions within the samples themselves were taken into account. Pertinent data on the necessary pellet enrichments for a maximum heat flux of 4.5×10^5 Btu/ft²-hr through the clad for both legs of the in-pile loop is given in Table XV. The unperturbed flux in the MTR reflector is reduced by approximately a factor of two because of the high absorption cross section of the loop piping and samples.

TABLE XV

ENRICHMENTS NECESSARY TO SIMULATE MAXIMUM WCAP-4 HEAT FLUXES IN BOTH LEGS OF THE IN-PILE TEST LOOP

	<u>High Flux Leg</u>	<u>Low Flux Leg</u>
Pellet U235 Enrichment w/o	10	27
Maximum Heat Flux Btu/ft ² -hr	4.5×10^5	4.5×10^5
Unperturbed Thermal Neutron Flux before Insertion of Experiment - ϕ_0 (n/cm ² -sec)	4.35×10^{13}	2.85×10^{13}
Ratio of U235 absorption to total absorption in the experimental hole	0.60	0.81
Ratio of maximum flux at pellet edge to ϕ_0	0.61	0.40
Ratio of average neutron flux within pellet to flux at pellet edge	0.95	0.77
Pellet center temperature °F.	3900	3500

It has been calculated that the addition of 600 ppm of boron to the water of this loop would increase the total absorption cross section of the leg containing the 10% enriched pellets by only 6% and would reduce the heat flux by less than this amount. The addition of 80 ppm of boron would cause virtually no reduction of heat flux.

Fig. 13 Maximum Clad Surface Heat Flux (Q) vs Enrichment (λ) in the WCAP-4 In-Pile Test Loop at the Materials Testing Reactor

First core design fuel rods are being inserted into "in-process water" irradiation holes in the reflector of the MTR.

Calculations have been made to determine into which test hole, characterized by an "unperturbed" thermal flux, the new samples should be inserted in order that the heat flux at the clad surface be 6×10^5 Btu/ft²-hr. According to the terminology used by MTR, the "unperturbed" thermal flux characteristic of a given test hole is the center-line flux measured when the hole itself and the other three holes in the particular "assembly" are filled with coolant water. There are no guarantees in any respect concerning the variation with loading cycle of the number and kind of samples in surrounding assemblies.

The calculation was performed by considering each hole and the proportional amount of Al surrounding it to be a cylinder concentric about the cobalt wire which measured the flux. From this "unperturbed" flux value, the magnitude of the neutron current diffusing inwards at the outer boundary of the cylinder was determined. The nuclear characteristics of the cylinder were recalculated to account for the insertion of a sample and the maximum neutron flux available to the fuel pellets was found by assuming the inward current at the boundary to be the same as that calculated for the "unperturbed" case. The results are given in Table XVI.

TABLE XVI

UNPERTURBED THERMAL FLUX REQUIRED FOR 6×10^5 BTU/FT²-HR
AT VARIOUS FUEL ENRICHMENTS

<u>Enrichment</u> <u>w/o</u>	<u>Required "Unperturbed" Flux</u> <u>(10^{13} n/cm² sec)</u>
2.7	14.0
5.4	7.8
6.8	6.7

2.7 Shielding Analysis

H. T. Williams

Neutron Counter Background Count Reduction Studies

A detailed study has been conducted to determine the best means of reducing the background counts in the neutron counters located in the neutron shield tank. The importance of this study rests in the fact that the neutron counting sensitivity obtainable is improved by reduction of the background counting rate. One way to accomplish this is to use an additional lead wall around the thimble containing the counter, which

would shield the counter from external (core and vessel) gamma rays without materially reducing the neutron flux. However, this lead wall would not shield the counter from prompt capture and long-lived activation gammas which would be produced by neutron capture in the steel thimble wall. The use of an aluminum thimble wall was therefore examined, since aluminum has a much lower thermal absorption cross section and hence gives rise to fewer gammas than stainless steel.

Table XVII illustrates the importance of the composition of the thimble wall in determining the amount of lead required to reduce the gamma dosage registered at the counter during full power operation to 1×10^4 R/hr. D_1 is the dosage due to all external sources which are attenuated by the lead and D_2 is the dose produced in the thimble wall itself and thus not attenuated by the lead. It can be seen that the aluminum wall allows the use of a much smaller thickness of lead.

TABLE XVII

LEAD THICKNESS NECESSARY TO REDUCE DOSAGE AT THE NEUTRON FLUX COUNTER
IN THE NEUTRON SHIELD TANK

<u>Thimble Wall Composition</u>	<u>Lead Thickness (inches)</u>	<u>D_1 (R/hr.)</u>	<u>D_2 (R/hr.)</u>	<u>D_{tot} (R/hr.)</u>
Stainless Steel	2.5	1.4×10^3	8.6×10^3	1.0×10^4
Aluminum	0.75	9.9×10^3	0.6×10^3	1.0×10^4

2.8 Startup Experiment Assistance

No work was performed under this subproject during the first quarter of 1958 in accordance with the Research and Development Program for the period from January 1 to June 30, 1958 as outlined 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

S. Eleftheriou

N. Michael

Ion Exchange of Radioisotopes

A synthetic solution of corrosion products was prepared by the addition of chloride salts of iron, chromium, and cobalt to demineralized water. Each of the metals was radioactively "tagged" so that the quantity of corrosion materials being retained by Rohm & Haas XE-150 resin could be easily determined. Experiments were performed first at a low pH of 2.8, to attempt to prevent colloid formation. They were then repeated at a pH of 8.3. The decontamination factors obtained at both pH values were quite poor - less than four. As evidence that a radiocolloid was formed, a portion of the test solution was passed through a Millipor filter (pore size: 1.2 micron). The DF averaged 1.07. Thus, it is certain that no ordinary, suspended precipitate of radioactive material was present. Experiments are in progress to determine whether one or more of the individual corrosion products is leaking through the resin.

The same solution used in the previous run was diluted 100 fold with demineralized water and adjusted to a pH of 8.4 with NH_4OH . A 1000 ml volume of the diluted solution was passed through 100 ml of XE-150 resin at a flow of 20 ml/min. The experiment was carried out at 130°F. Dilution of the original test mixture to 1:100 very favorably influenced the decontaminating ability of the resin. Since the influent solution had a specific activity of 5.9×10^4 c/m/5 ml sample, and the effluent had a 0 c/m/5 ml sample, a decontamination factor of more than 6×10^4 was thus presumed to be obtained. This compares with a D.F. of 3.5 obtained with the original solution. The fact that increased decontamination was obtained with the more dilute solution would tend to suggest that the reason for the improved pickup might be due to a gel formation, resulting from partial hydrolysis of the metal salts. Thus, it is possible that improved efficiency of the XE-150 resin resulted from the presence of a filterable component, rather than increased ionization of solutes.

Thermal Stability Tests of Ion Exchange Resins

Experiments were begun to determine the temperature stability of promising ion-exchange materials, by exposing Rohm & Haas XE-150 resin in elevated temperature baths at 140 and 170°F. The resin was sampled periodically, loaded into columns and the capacity determined. A solution of sodium chloride was used in evaluating the capacity; the effluent from the columns was monitored with a conductivity cell; break-through capacity was determined by an arbitrary conductance value (2.0 microhms/cm - an approximate value which might be expected under operating conditions).

Breakthrough with the exposed resin occurred after a throughput of 940 ml of NaCl solution. Breakthrough with the control was after 980 ml. The difference between the two values is probably within experimental error. Therefore, it cannot be stated that breakthrough capacity was affected by "cooking" for one month at 170°F.

Conditions for Nuclear Poison Storage

Studies, previously reported in YAEC-52, Quarterly Progress Report for the October to December, 1957 period, have shown that, at both ambient temperature and 140°F, there was no attack on AISI 304 stainless steel by saturated solutions of boric acid. Corrosion was severe, however, on AISI 4135 H carbon steel under the same conditions.

AISI 4135 H carbon steel specimens were then tested in a saturated boric acid solution brought to neutrality (pH = 7.2 - 7.7) by the addition of lithium hydroxide to determine whether neutralized boric acid solution can be stored in carbon steel storage tanks. The specimens were positioned so that one-half of each specimen was located in air and the other half in the solution. The following corrosion rates were obtained:

TABLE XVIII

CORROSION RATES ON AISI 4135 H CARBON STEEL
IN SATURATED NEUTRALIZED H₃BO₃ SOLUTION

	<u>Solution</u>			<u>Temperature</u>		
	<u>70°F</u>			<u>140°F</u>		
	(at)	<u>2 wk.</u>	<u>8 wk.</u>	<u>2 wk.</u>	<u>4 wk.</u>	<u>8 wk.</u>
Cumulative weight change, mg		-0.4	-4.6	+0.4	-17.4	-488
Initial weight, gm		46.7		44.2		
Penetration rate, in/yr		3.1 x 10 ⁻⁵ *		3.0 x 10 ⁻³ **		

* Localized at the air-water interface

** Accelerating corrosion

On the basis of the accelerating penetration rate at 140°F, AISI 4135 H carbon steel was considered as being unacceptable for the storage of saturated, neutralized, boric acid solutions.

3.3 Corrosion of Materials of Construction

C. R. Bergen

A. Krieg

Westinghouse MED Loop "A" Low (40 ppm) Boron Concentrations

Test No. 1 - AISI 304 stainless steel coupons of varied surface finish were exposed at 600°F, 2000 psi and a flow velocity of 7 feet per second to water of the following composition for 65 days:

B (as boric acid) = 40 ppm

LiOH-H₂O = as required to yield pH = 10

Hydrogen = 25-35 ml (STP)/kg of solution

A statistical analysis of the completed experiment revealed the following weight changes:

1. Descaled weight changes (difference between the original weight and the weight of the descaled specimen after exposure) as a function of exposure time were:

<u>Exposure Time (day)</u>	<u>Test Section</u>	<u>Weight Loss mg/dm²</u>
2	Top	7.4
6	Top	7.0
14	Top	9.6
30	Top	5.2
65	Top	7.9
65*	Bottom*	7.2

* Undisturbed for the entire duration of the test.

2. Descaled weight changes as a function of surface finish were:

<u>Finish</u>	<u>Weight Loss mg/dm²</u>
As received	6.9
As received, descaled*	6.6
Sanded	7.9
Ground	7.2

* Preliminary descaling prior to exposure

3. Descaled weight changes as a function of position in the specimen holder (upstream or downstream) were negligible.

In general, descaled weight losses (loss of metal from the specimens) of the stainless steel coupons in the environment described and under the conditions tested were negligible after the first two days of exposure. The metal loss within the first two days corresponded to approximately 7 mg/dm². Final analysis of the test concluded that the coupon specimens and pre-stressed U-bend specimens suffered no detrimental attack of a micro-structural nature after 65 days of exposure.

Test No. 2 with AISI 304 stainless steel coupons of varied surface finish was terminated after 44 days of operation. Conditions and environment were the same except that lithium hydroxide was omitted from the aqueous composition (avg. pH = 6.9). The results of this test should demonstrate the effect of the pH factor on the corrosion behavior of AISI 304 stainless steel under the conditions tested. Specimens in the upper test section were inspected after 2, 4, 6, 14, 30 and 40 days of exposure. The test data are undergoing analysis.

Westinghouse MED Loop "B" - High (1600 ppm) Boron Concentration

Test No. 1 with AISI 304 stainless steel coupons of varied surface finish in water containing nominally 1600 ppm boron (as boric acid), lithium hydroxide as required for a pH value of 10 and 30 ml hydrogen (STP) per kilogram of solution was terminated after 23 days of exposure because of excess leakage. At the 2-day and 6-day openings of the upper test section, no deposits were found inside the test section or adjacent loop areas.

At the 14-day opening white solid deposits were found inside the test section and on internal adjacent loop areas as well as on the outside of the test flanges. The deposits were removed and analyzed both by spectroscopy and chemical procedures. X-ray analyses revealed the deposits had a non-crystalline structure. Among the elements found in the deposits, Li and B weight per cents varied from 4.0 to 10.2 and 13.7 to 18.0, respectively. No definite composition could therefore be determined for these deposits. No deposits were found in the loop piping at the end of the 23-day test period.

After the termination of Test No. 1, the pressurizer was disassembled for inspection. A sharply defined white deposit was found inside the pressurizer on the wall, the face of the base plate and on the immersion heaters to a height of 6 inches from the base flange corresponding to the position of the drain connection. The deposit may have been the result of evaporation of residual liquid in the pressurizer during the shutdown process. The material coating the walls was chemically analyzed and the material deposited on the base plate and the immersion heaters was similarly analyzed. The ratio of lithium to boron in these deposits is similar to that found externally on flanges and to the composition of the pressurizer fluid at the time of shutdown.

Test No. 2 was then performed as a repetition of Test No. 1. Before initiation of the test, the loop was operated with pure de-aerated water at test conditions for 24 hours. No appreciable leakage was observed. Shortly after addition of boric acid and lithium hydroxide, evidence of leaks was observed by the formation of white deposits externally on flanges. The 2-day opening for specimen inspection

revealed no internal deposits. Test No. 2 was terminated after 17 days of operation at 1600 ppm B (as boric acid), 25 - 35 ml H₂ per kilogram of solution with lithium hydroxide added as required to maintain the pH value at 9.5 -10.5. Although excessive leakage recurred, no evidence of internal deposition of lithium-boron containing solids was found. A metallographic examination is being made of the specimens and weight change data are being statistically analyzed.

Loop operations were terminated in the Westinghouse MED Loops on January 17, 1958. Topical Report YAEC-68, "Corrosion of AISI 304 Stainless Steel in Boronated Water" is being prepared to summarize the results of the tests.

Westinghouse APD PAR Loop "A" Operations

1. A visual inspection was made of the AISI 316 stainless steel fuel bundle which had been exposed as Test No. 1 to water containing 40 ppm boron (as boric acid) and 30 ml hydrogen (STP) per kilogram of solution for 321 hours at 600°F, 1850 psi and a flow velocity of 41.7 feet per second. The metal was coated with a dark brown adherent deposit. No evidence of attack was visible on the assembly. A further examination is being made by the Westinghouse APD Metallurgy Section.
2. Test No. 2 in PAR Loop "A" involved the exposure of an AISI 304 stainless steel brazed tube bundle to the same environment as Test No. 1. The test was completed after 720 hours of exposure. The results are being analyzed.
3. Test No. 3 was terminated at the conclusion of 715 hours of operation at high boron concentration without pH adjustment. Various proposed materials of construction were exposed to the aqueous environment. Average conditions for the test were as follows:

T	598°F
P	1792 psi
Flow velocity	39.4 ft/sec.
B (as boric acid)	1590 ppm
H ₂ (ml (STP)/kg solution)	27
pH	5.0 - 5.8
O ₂	0.02 ppm
Cl-	0.05 ppm

Examination of the specimens exposed in Test No. 3 is in progress. Specimen surfaces have been scanned by X-ray fluorescence analysis prior to descaling. The base metal will be similarly analyzed after descaling to obtain information on the nature of the surface film. Exposed weight measurements have been completed. Specimens have been descaled by two procedures (electrolytic and chemical) and descaled weight changes have been determined and are being evaluated. The exposed weight changes (difference between the weight after exposure and the original weight) are as follows:

<u>Material</u>	<u>Literature values</u> (mg/dm ² /mo (no boric acid present))	<u>Exposed Weight</u> <u>Change in 1 month</u> (mg/dm ²) (1)
AISI 304 stainless steel- quenched	-15 ⁽²⁾	-21
AISI 304 stainless steel- heat sensitized	- 5 ⁽²⁾	-17
Inconel	- 5 ⁽²⁾	-77
Ag-In-Cd	+ 190 to 250 ⁽³⁾	+56
Zircaloy-2	+ 5 ⁽²⁾	+17
Hafnium	- 5 ⁽²⁾	+ 4
AISI 410 stainless steel	-10 ⁽²⁾	-65
Nicrobraz 50 - on AISI 304 stainless steel		-32
AISI 304 stainless steel - weld coated		-15
<u>Coupled Specimens:</u>		
AISI 304 stainless steel - quenched vs heat sensitized		-13
AISI 304 stainless steel - quenched vs Zircaloy-2		-6
AISI 304 stainless steel - heat sensitized vs Ag-In-Cd		+22

- (1) Negative values represent weight losses; positive values represent weight gains.
- (2) From TID-7006 "Corrosion and Wear Handbook". Data at conditions most nearly approximating those used in Westinghouse APD PAR Loop "A", Test No. 3.
- (3) From Westinghouse APD-FWR-PMM-1207, PWR Alternate Control Rod Material Program.

An AISI 304 stainless steel specimen stressed at 25, 50, 75 and 100% of the yield point at 600°F is being examined. The specimen was exposed in a dead-end arm near the base of the pressurizer (above the inlet). Deposition of a grayish material occurred on the holder containing the specimen. The following composition of the material was determined:

Fe	26.3%
Cr	5.3%
Ni	6.2%
Co	0.3%
Th *	24.8%
U *	1.2%

* Residues from previous PAR test in Loop "A".

The specimen appears to have suffered some general corrosive attack on the underside. Further examination is in progress. Test No. 4 was begun to screen the corrosion behavior of materials similar to those exposed in Test No. 3 under conditions of boron concentration cycling from 1600 ppm B to less than 5 ppm B. One cycle of operation at 1600 ppm B for one week followed by operation at 2-4 ppm B for one week has been completed. Concentration lowering was accomplished by feed and bleed dilution using de-aerated demineralized water with loop temperature maintained at 600°F. Concentration increase was accomplished by feed and bleed injection of a de-aerated 5.4% boric acid solution with loop temperature maintained at 600°F. Analyses were made to follow the progress of both operations. Results are being summarized. It is apparent that special means will be required to avoid unbalanced boron concentrations between the loop and the pressurizer since the pressurizer is essentially stagnant. Feed and bleed operations in the loop do not appear to exert a marked influence on the pressurizer boron concentration. During the early part of Test No. 4, a pinhole leak appeared at a welded joint in the bypass sampling system. This appears to have been due to a poorly executed weld with evidence of low freezing point inclusions at the grain boundaries in the weld zone and voids which initiated cracking in the weld zone.

Static Autoclave Tests on Silver-Indium-Cadmium Control Rod Material

A static autoclave test is currently being run on the corrosion of Ag-In-Cd control rod material in neutral water in an effort to verify results of a previous test. Two one-week exposures have been completed. Oxygen has been held below the detectable limit (less than 0.05 cc/KgH₂O). So far the results are in good agreement with a previous run in which the oxygen concentration had been questioned. Test conditions are 650°F and saturated steam pressure. It is planned to continue this study for six more weeks.

Static Autoclave Test on AISI 304 Stainless Steel

A test was conducted to determine the effect of boric acid concentrations (40 ppm and 1600 ppm boron) on the corrosion rate of AISI 304 stainless steel under static conditions at a pH of 9.5 (by the addition of lithium hydroxide) with 30 cc/kg hydrogen (STP) and at 600°F. The following as received corrosion rates were obtained by differentiation of weight change vs exposure time curves:

"As Exposed" Corrosion Rates of AISI 304 S/S Specimens With Various Surface Finishes

40 ppm Boron 0.23 (Minimum) to 2.3 (Maximum) mg/dm²/mo

1600 ppm Boron 1.1 (Minimum) to 3.5 (Maximum) mg/dm²/mo

Corrosion rates for each type of surface finish remained constant from 400 to 1200 hours.

No evidence of change of lithium or borate concentration or deposition of boron-containing material was detected in 1200 hours of testing. The autoclave was opened for inspection of deposits and analysis of solution approximately every 250 hours, and then recharged with fresh test solution at the beginning of each period. Nominal concentrations were 1600 ppm boron as boric acid, and 700 ppm lithium as lithium hydroxide.

3.4 Interactions Between Chemical Absorbers, Corrosion Products and Fission Products

D. D. Whyte

C. R. Bergen

To simulate a fuel cladding failure, a UO_2 pellet was encapsulated in contact with damp boric acid in an AISI 304 stainless steel capsule, and held at 1000°F . for twenty-four hours. The pellet was then placed in boiling water for one-half hour. The total weight loss, and from this, the rate of attack on the pellet, was determined. For the test conditions a tentative corrosion rate of the UO_2 pellet was determined as $(4.1 \pm 2) \times 10^{-6}$ cm/mo or approximately 0.0001 inch in three years. This rate is believed to be insignificant.

Specimens are being fabricated by the Westinghouse APD Metallurgy Section for a study in which corrosion data will be obtained on the effect of the pH control agents, NH_4OH , KOH , and LiOH , on materials of construction. Insulated and coupled specimens will be exposed to 1600 ppm B solutions brought to a pH of 10 by the addition of the respective bases. Specimen materials consist of Ag-In-Cd alloy, Zircaloy-2, Inconel, AISI 304, and AISI 410 stainless steel and Microbraz 50 coupled to AISI 304 stainless steel. It is planned to conduct a test at 600°F ., and to follow, if possible, interactions among each pH control additives, the specimen materials, and the soluble neutron poison.

Crud Transport and Deposition Studies under Van de Graaff Radiation

Modifications to convert a corrosion and test loop at the Westinghouse Radiation and Nuclear Laboratory to a crud transport and deposition loop are near completion. Temperature and pressure testing of the loop will follow the hydrostatic pressure tests now being conducted. Materials to be tested in the crud transport and deposition studies in the presence and absence of Van de Graaff radiation are:

AISI 304 stainless steel at pH 7.0
AISI 304 stainless steel at pH 9.5 to 10.5 (with additive)
Zircaloy-2 at pH 10.0 (with additive)

The initial tests will be performed using stainless steel wool as a crud source.

3.5 Decontamination and Waste Disposal Studies

R. Watkins

The results of spectroscopic analysis of the final solutions produced in the basic permanganate-citrate decontamination treatment of 80% Ag, 15% In, 5% Cd control rod material indicated that there was no preferential solution

of any element in the alloy. Metallographic examination of the control rod alloy specimens indicated that the slight attack which had occurred in the case of the treated specimens was quite general with no significant tendency to form pits. The maximum depth of penetration as determined by examination of photomicrographs taken of both the treated and untreated specimens was approximately 0.2 mil. The average rate of attack, as determined by weight loss, was less than 0.1 mil per 1/2 hour treatment.

Sensitized specimens of AISI 410 stainless steel were tested for corrosion effects caused by static contact with the basic permanganate-citrate clean-up solutions. The average depth of penetration as determined by weight loss data was less than 0.02 mils for the first treatment and less than this amount for a subsequent standard treatment. The treated specimens have been sent to the Westinghouse APD Metallographic Laboratory for examination in an attempt to determine whether any serious pitting or intergranular attack had occurred.

Specimens of Graphitar-14 bearing material as used in the Westinghouse canned pumps were tested by standard treatment for compatibility with the basic permanganate-citrate decontamination solutions. Weight loss data taken in three series of exposures indicated an average penetration of up to 0.15 mils per treatment. The results of a comparative examination of the Graphitar-14 samples before and after treatment with the standard basic permanganate-citrate decontamination procedures revealed an increase from 140 to 160 microinches rms surface finish, Figure 14. No other effect of the treatment weight loss, other than the above, was apparent. Although the rate of attack seems quite small, it is believed that adequate precautions should be taken during an actual plant decontamination operation, so that the concentrated clean up solutions will not come into contact with the pump bearings.

A study was begun concerning possible procedures for chemical treatment of the basic permanganate solution which will facilitate disposal of the reagent after use in reactor cleanup. Preliminary theoretical results indicate that precipitation of the major portion of the activity can be accomplished, and that ion exchange resins should be able to handle the problem of removing most of the activity that remains.

A series of trial experiments were performed in an attempt to determine the degree of attack caused by the basic permanganate-citrate decontamination solutions on sintered UO_2 pellets. Early results indicate that partial solubility can easily be achieved. A quantitative study is now in progress. These results are significant from the standpoint that UO_2 containing highly active fission products may be released to the primary coolant system through defective fuel rods. The solution of this material would be desirable in any subsequent decontamination operation.

Fig. 14 Photomicrograph (15X power) of Graphitar - 14 Specimens
Top Untreated, Bottom Treated Three Times According to the
Standard Basic Permanganate-Citrate Decontamination Procedure

3.6 Crud Inhibition, Suspension and Removal

D. D. Whyte

AISI 410 stainless steel specimens were exposed to demineralized, degassed water with a pH of 9.5 obtained by the addition of potassium hydroxide, at 600°F and 1525 psi for 936 hours to determine the corrosion rate of the proposed control rod drive mechanism material. The initial exposed weight gain amounted to 14 mg/dm²/336 hours, after which, the corrosion rate remained constant at 8 mg/dm²/mo. for the duration of the test.

4.0 MECHANICAL DESIGN

Mechanical and Thermal Section: E. Frisch, Consultant

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

Spring-Locked Control Rod Joint

The mechanical design was begun of a "rigid" spring-locked control rod joint. The joint holds two rod sections together by lowering the top section onto the bottom section and rotating the top section into the locked position. The control rod locking mechanism was designed to produce a locking force of 65 pounds per spring with a locking engagement of 0.060 inch. With an Inconel-X spring in each of two opposite legs of the control rod, the locking and unlocking torques are approximately 1000 inch-pound maximum. A model is being constructed.

Full Size Strap Design Fuel Assembly Model

Fabrication of the full-size strap design fuel assembly model was completed by the Westinghouse APD Model Shop. The successful fabrication demonstrated the welding feasibility and conformance with design size that can be maintained with this method of joining tubes into bundles. The assembly consist of 305 AISI 316 stainless steel tubes having an outside diameter of 0.335 - 0.339 inch, a wall thickness of 0.021 inch, joined together by AISI 304 stainless steel straps 0.025 inch thick. Each bundle was mechanically assembled into an "egg-crate" by spot welding 5/8 inch wide straps to nine rows of nine tubes each and then spacing the rows by interlocking 15/16 inch wide notched straps to the welded straps. The completed assembly is shown suspended by a ball type fuel handling tool in Figure 15.

4.2 Control Rod Drive Mechanisms

E. Frisch

Assembly and detail drawings are being prepared by the Westinghouse Atomic Equipment Department preliminary to the fabrication of the positive engagement type control rod drive mechanism prototype. It is planned that the prototype will be available for testing in the latter half of 1958. Application of the completed design to the actual fabrication is being coordinated by the Westinghouse Atomic Power Department.

Fig. 15 Full Size Strap Design Fuel Assembly Model

4.3 Core Support Plate Stress Analyses

The experimental stress analysis on the model of the core support plate was continued. Deflection measurements made on a specimen consisting of two plates jointed by 76 cylindrical spacers indicated the following:

1. In addition to the 76 spacers which separate the top and bottom plates, a circular reinforcing rib joining the two plates at the periphery of the smaller diameter top plate was added. The rigidity of the support structure was increased approximately 25%.
2. By the application of dimensional analysis, the maximum deflection of the Yankee conceptual design of the lower core support plate, Figure 16, was found to be approximately 0.032 inch. The total static load was assumed to be 100,000 pounds which is the summation of the weights of the fuel assemblies, control rods, control rod follower shrouds, shroud support, and lower core support plate.

The core support plate structure is of such a complex form that it is impossible to predict where maximum stress levels may occur. In order to get an overall view of the surface stresses and to locate regions of dangerous stress concentrations, Stresscoat No. ST 12045 was sprayed on the test specimens. The strain sensitivity of the coating was 0.0009 inch per inch as indicated by a calibration bar at the time of the test. Results of the Stresscoat test were photographed. To render the crack patterns in the Stresscoat more visible and to simplify the photography, a red dye etchant (ST-1300) was applied to the coated surface.

The following information was obtained from the Stresscoat test:

1. Initial cracking indicated the location and direction of regions of maximum tensile strain.
 - a. Figure 17, shows the formation of final cracking patterns in a single plate perforated with holes and cruciform openings. Cracks started symmetrically at the ends of the cruciform openings. As the load was increased, cracks appeared in the narrow areas between the holes.
 - b. Figure 16, shows the 1/4 inch scale model of the Yankee lower-core support plate. Figure 18, shows a closer view of the plate model revealing the stress concentration at the outer region of the plate, at the outer sector of the flow holes.
2. The stress trajectories, or the direction of principal stresses, were distinctly indicated, thereby showing where and in what direction to place strain gauges.

Fig. 16 Stresscoat Crack Pattern on a Lucite Model of the Conceptual Design of the Lower Core Support Plate

Fig. 17 Stresscoat Crack Pattern in a Lucite Core Plate Model Perforated With Holes and Cruciform Openings

Fig. 18 Stresscoat Crack Pattern in Part of a Lucite Model of the Conceptual Design of the Lower Core Support Plate

3. A comparison of the crack-pattern density of the tested specimen with the crack density formed on a calibration bar under known strain showed the approximate level of strain within $\pm 15\%$ accuracy. This information is of value principally for calibration of strain gauges to be used in subsequent tests.

The results obtained from the tests will be included in a Topical Report now being prepared.

Reactor Vessel Internals Specifications

A design specification was prepared to cover the design of all of the reactor vessel internals. The purpose of the specification was to provide a single and complete source of accurate information for the final design so that complete layout and assembly drawings can be made. The following dimensions were specified, at ambient temperature:

1. Core Cross Section

- a. Overall cross sectional dimension of the core spanning ten fuel assemblies - 76.500 inches.
- b. Clearance between two fuel assemblies - 0.098 inches
- c. Control rod pitch - 7.660 inches.

2. Control Rod

- a. Span - 7.865 inches
 - b. Thickness - 0.285 inches
 - c. Active length - 94-9/16 inches
 - d. Distance of travel - 90 inches
3. Overall length between the two support shoulders of a fuel assembly - 100-7/8 inches.
 4. Distance between the upper and lower core support plates - 101-1/4 inches.
 5. Clearance between the fuel assembly support shoulder and the upper core support plate - 3/8 inch.

Core Hold-Down Ring Design

Hydraulic calculations revealed the possibility that a net resultant upward force may act upon the internal support structure due to the pressure drop across the core. In order to prevent the core from "floating" within the vessel during operation, a hold-down force of

30,000 lbs. has been specified. This force was increased from the original force of 15,000 lbs. specified for the core hold-down ring because a one psi error in pressure drop calculation can result in an upward force of about 7,000 lbs. The total deflection of the core hold-down ring will be 0.142 instead of 0.194 inch. The core hold-down ring thickness was increased from 1-1/4 to 1-3/4 inches. To accommodate this new design, the dimension of the vessel support shoulder was changed from 7-3/4 to 8-1/4 inches in depth.

4.4 Design For Critical Experiment and Irradiation Tests

R. D. Edsell

G. H. Eng

MTR Process Water and Test Loop Specimen Design

Design characteristics of fuel rod and control specimens that have been determined for the MTR process water and test loop irradiation experiments are described under Project 1.4, "Fabrication and Analysis of Fuel Elements for Critical Assembly and Irradiation Tests".

Fabrication of CRX "Poison Thimble"

A "Poison Thimble" was designed previously for use in the CRX Core. The vertically mounted thimble will be filled with steel, boronated water, or other neutron absorbing materials for certain planned experiments. The design was changed as a result of a required increase in length. To fulfill the additional requirement of straightness and to minimize the fabrication cost, extruded aluminum square tubes, 3 inches by 3 inches, 1/16 inch thick, and approximately seven feet long, were used. Fabrication of the redesigned thimble has been completed and delivery was made to the Westinghouse Reactor Evaluation Center.

5.0 THERMAL AND HYDRAULIC DESIGN

Thermal and Hydraulic Design Section: E. Frisch, Consultant

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

Temperature Distribution in a Fuel Tube Disc

Calculations were made to determine the temperature distribution in a fuel clad tube in which AISI 304 stainless steel discs were located between the pellets at six inch intervals. The study was made for the following conditions:

- a. The tube lies in the hot channel.
- b. There is contact between pellets and the tube.
- c. AISI 304 stainless steel discs are brazed into the tube.
- d. There is contact between the pellets and discs.

The calculations indicate that, with a clad surface temperature of 642°F., the 0.060 inch thick AISI 304 stainless steel discs will not melt in the hot channel. The calculated temperatures are:

<u>Disc Thickness</u>	<u>Maximum Temperature</u>
0.060 inch	1292°F.
0.090 inch	1128°F.
0.125 inch	1044°F.

Stress calculations will be made to determine if the thermal stresses exceed an allowable limit.

Alternate First Core Thermal Analysis

A thermal analysis of an alternate first core was performed using the new Bettis Plant value of thermal conductivity of UO_2 , - 1.0 Btu/hr-°F. The design is similar to that of the first core except a 17 x 17 array is substituted for an 18 x 18 array fuel assembly.

The following changes in heat transfer and hydraulic characteristics were obtained from the analysis:

Area in fuel rod cross section, ft ²	16.1
Velocity along fuel rod, ft/sec	13.4
"Active" surface area, ft ²	14,190
Average heat flux, Btu/hr-ft ²	94,300
Maximum heat flux, Btu/hr-ft ²	487,000
Average film coefficient, Btu/hr/ft ² -F.	5,540
Burnout heat flux, Btu/hr-ft ²	1.44 x 10 ⁶
Average film drop, °F	17
Maximum center of fuel, °F.	4,565
Rod outside diameter, inch	0.349
Tube wall thickness, inch	0.022
Number per cross section	20,710
Rod lattice, inch	0.447
Equivalent diameter of unit cell, ft.	0.0316
Rods per assembly	272 and 273

In addition to the preceding analysis, the maximum thermal rating was obtained on the basis of a maximum center-of-fuel temperature of 5000°F., which is the melting point of the fuel. The maximum rating is 440 MW. This power output would be a limiting condition for load transients.

"YAEC Reactor Core Comparisons" Topical Report

Topical Report YAEC-6, Revised, - YAEC Reactor Core Comparisons - was issued after being revised to include new conductivity data for the uranium dioxide fuel.

Fuel Rod Gap Medium Thermal Studies

Calculations were performed to determine the center of pellet temperature when the fuel rod gap is filled with various mixtures of: (1) air and hydrogen, and (2) air and helium. In order to estimate the thermal conductivity of the gas mixture, the method proposed by Messrs. A. L. Lindsay and L. A. Bromley in "Thermal Conductivity of Gas Mixtures" is being used (Industrial and Engineering Chemistry, Volume 42, No. 8, p. 1508-1510, 1950). The data obtained are being analyzed. The study of the fuel rod gap medium was then expanded to include a study of the fuel rod in the hot channel of the first core. The following studies were begun: (1) axial temperature distribution of coolant, rod surface, and center of fuel (2) axial distribution of the average temperature of the clad, gap, and fuel (3) internal pressure due to the differential expansion of the pellet and clad, and the effect of the higher operating temperature (4) determination of the thermal conductivity of varying compositions of hydrogen and air and of helium and air (5) maximum center-of-fuel temperature when the gap is occupied with varying compositions of the above mixtures, and (6) effect of pressure on the thermal conductivity of the gas mixtures considered.

5.2 Hydraulic Design

A. A. Bishop

Pressure Drop in the Reactor Pressure Vessel Model

Pressure drop experiments were conducted with the core of the reactor vessel model empty, packed with one inch Raschig rings, 3/4 inch Berl saddles, and a mixture of Raschig rings and Berl saddles. Each type of packing has a characteristic pressure drop and void volume, or free flow area, because of its unique geometry. The pressure drop was measured from vessel inlet to outlet, across the inlet enlargement and direction turn, and across the thermal shields. Flow rates were varied between 50 to 300 gpm. Although the agreement between the overall experimental and theoretical pressure drop was good, calculated pressure drops across individual sections in the vessel differed somewhat from the experimental pressure drops. The apparent paradox of the overall agreement between calculated and theoretical pressure drops and the poorer agreement between drops across individual regions is explained by the fact that there are regions where the calculated energy loss is low. This low energy loss has been balanced by the conservative approach used to calculate the enlargement and contraction losses. A more complete analysis will be possible when data across the fuel assembly model are obtained.

Graphs of the data showed that the pressure drop expressed in psi was proportional to the total flow expressed in gpm to the 1.95 power for an empty or Berl saddle filled core. (Δp (psi across vessel) = $KG^{1.95}$) G is expressed in gpm.

The method used to calculate pressure drop in the reactor vessel model was as follows: The entrance enlargement and direction turn were calculated by the law of cosines at the inlet which gives a conservative value. The included angle was 90° . The resultant velocity was used with a coefficient of 1.0 to calculate the pressure drop in the region. Flow paths through the thermal shields were in parallel. The drop was calculated by trial and error since both the velocity and friction factor in the thermal shield region were unknown. The procedure was to assume a friction factor and then calculate the pressure drop. Enlargement and contraction coefficients at the exit and entrance to the thermal shields were assumed to be 1.00 and 0.4 respectively.

Using the calculated velocity as a first approximation, the velocity was calculated for each path through the thermal shields and core baffle region. The Reynolds number was then calculated to find a friction factor which was checked against the assumed friction factor. The difference converged rapidly, and only a few trial friction factors were required. To calculate the drop across the 180° turn as the water enters the lower plenum, the law of cosines was again used. The loss coefficient was assumed to be 1.00. No pressure loss was charged to the cross flow that might exist in the lower plenum. (The velocity distribution in the lower plenum will be measured by means of special pitot tubes). Enlargement and contraction losses across the support plates were based solely on area ratios. In calculating the pressure drop between the top support plates and the outlet flow nozzles, the flow across the shrouds was assumed to be similar to the flow across staggered tubes. The maximum change in slope of a pressure drop versus flow curve for the model amounted to approximately 10-15% by varying the internal geometry of the core. Each type of packing used in the core to economically simulate the actual fuel assemblies had a different percent free flow volume when the packing was placed into the core chamber. The size of the holes in the core support plate had a more pronounced effect on the entrance flow distribution to the core than the internal geometry of the core.

Flow in the lower plenum had a high degree of turbulence throughout the flow range used (50 to 300 gpm). Large, clearly defined coplanar vortices, which rotated in a counterclockwise direction, were observed in the lower plenum. The nature and size of these vortices suggested that straightening vanes might be effective in reducing the pressure drop in the plenum. Smaller vortices, somewhat random in nature, were also observed in regions of the lower plenum. These localized disturbances were especially evident around the circumference of the core support plate. Additional water holes will be placed in the core support plate model in an attempt to reduce the turbulence around the periphery of the plate.

An unexpected result was obtained when simulated shroud tube supports in the form of brackets and plates were used in the model. The pressure drop versus flow curve remained the same. One explanation

of this observation is that the vortex formation in the neighborhood of the simulated shroud support is broken up, thereby decreasing the pressure drop in this region.

No difference in static pressure around the lower plenum at a given elevation was detected with "U" tube manometers. Thus, it appeared that flow in the vicinity of the plenum wall was symmetrical around the vessel.

Work to be completed on the pressure drop phase of the reactor vessel includes: (1) a mathematical interpretation of pressure drop data, (2) insertion of directional vanes in the reactor inlets, (3) streamlining the entrance and exit regions in the vessel, and (4) an attempt to correlate the experimental data with the prototype by simulative reasoning.

Friction Factor Versus Heat Transfer Coefficient Correlation

Because of the similarity in the mechanisms of momentum and heat transfer, it is often possible to calculate the heat transfer coefficient in a duct if the velocity distribution is known. Conversely, if the heat transfer coefficient is known, the friction factor can be determined. Using an average heat transfer coefficient of 6,050 Btu/hr-ft²-°F., as calculated from Dittus-Boelter correlation and reported for the first core, the derived values of the friction factor coefficients are as follows:

Reynolds Analogy	0.0134
Modified Reynolds Analogy	0.0156
Prandtl Analogy	0.0151
Von Karman	0.0149
Colburn	0.0141

Each of the above values is lower than the design value of 0.0175 obtained from the Moody diagram. This value, which is used to calculate the pressure drop across the reactor, omits the effect of local boiling as do factors obtained by analogy.

Shock Absorber Design

High speed pictures at 3500 frames per second + 10% were taken of a CRX shock absorber to compare operation with theoretical calculations. The films were viewed first at 16 frames per second and then at a slower speed by means of a hand viewer. Results of the experiments are as follows:

1. No idiosyncrasies in the behavior of any particular rod were observed, and the transition between free fall and buffered fall appeared to be smooth.
2. Although no perceptible bounce or large oscillations were observed as the control rods came to rest, close observation did reveal what could be interpreted as minute oscillations superimposed on the average time-distance curve.
3. The drop time was not the same for all nine control rods. This difference can be attributed to variations in instrument response time and/or mechanical friction in the control rod mechanism.
4. Based on a minimum total effective dashpot travel of 3.4 inches and an impact velocity of 158 inches/sec, the deceleration is $3624 \text{ inches/sec}^2$ or 9.6 times the acceleration of gravity.

Flow Distribution in the Reactor Core

The flow distribution in the reactor core is being studied by considering a matrix consisting of a flow channel and the four hydraulically connected surrounding channels. The pressure drop is balanced axially and radially in the matrix over an incremental axial distance. The apparent pressure difference between adjacent channels at a given height causes flow through the orifice (the gap or separation between fuel rods). Mass transfer due to turbulence is neglected. A second pressure drop balance over the given increment is made with the new flow rate. This procedure is repeated until the pressure is equal at the given elevation and no cross flow exists. The next incremental height is then analyzed and the procedure is continued until the exit of the flow channel is reached. The pressure drop increase due to local boiling is included in the analysis. Based on preliminary calculations for a two channel analysis (hot channel and average channel) of the core, the flow redistribution is approximately 10% less at the outlet of the hot channel.

The results obtained from the analytical studies will appear in a Topical Report now being prepared, "The Effect of Local Boiling on Flow Distribution in the Yankee Reactor".

6.0 CONTROL ROD DEVELOPMENT

Metallurgy Section: R. K. McGeary, Manager
Chemistry & Ceramics Section: R. F. Sterling, Manager

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

E. Foster

Extruded Silver-Indium-Cadmium Test Control Rod

The silver-indium-cadmium billet received from the Federated Metals Division of the American Smelting and Refining Company was successfully extruded to a cruciform design by the Revere Copper and Brass Company. Attempts to draw-form the absorber section to close tolerance as a means of eliminating or minimizing machining were unfavorable due to the poor condition of the surface. The complete control rod assembly was delivered to the WREC for comparison tests of the worths of a silver-indium-cadmium control rod and the silver-cadmium control rods now being used at the facility.

Procurement of Hafnium Strip for a Test Control Rod

Wah-Chang Corporation attempted to produce 100 pounds of hafnium strip for a CRX control rod by arc melting sponge to ingots to eliminate the crystal bar (refining) step. Upon hot rolling the ingots to strip, porosity and cracking were observed, necessitating rejection of the rolled strip. Further arc melting trials will be performed so that the two refining methods may be evaluated.

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.

The principle activity under this project during the first quarter of 1958 was the reviewing of the loss of coolant and startup accident studies which were carried out under Project 2.0.

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

H. G. Hargrove

A. R. Collier

J. Baron

The Westinghouse APD Research and Development Chemistry Program was reviewed to clarify suggested revisions in scope and emphasis in order to keep the chemistry program and plant design needs coordinated.

8.11 Reactor Handling Tools and Plant Shielding Analysis

Reactor Handling Tools

E. V. Anderson

A. F. Fritz

The fabrication of an experimental model of the control rod drive shaft handling tool was completed. Fabrication work is continuing on the "mock-up" drive shaft and the necessary supporting equipment in the Westinghouse APD Experimental Pit so that the handling tool for the control rod drive shafts may be tested. Figure 19 shows the model of the control rod drive shaft handling tool.

A conceptual design and an equipment specification were prepared for the plate and barrel handling fixture.

Preparation work was continued on a Topical Report covering radioactivity in the Vapor Container and Main Coolant Loops which will show the various activities, their source and means of removing such activities.

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 first quarter of 1958.

Fig. 19 Control Rod Drive Shaft Handling Tool Model

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 R. D. Leamer

Criticality Experiments

A series of critical experiments is being conducted at the Westinghouse Reactor Evaluation Center on stainless steel clad uranium dioxide fuel rods in water moderated lattices.

Criticality was initially attained on December 18, 1957 by loading 1964 fuel rods each containing approximately 1-1/4 lb. of UO_2 pellets enriched to 2.7%. A slab configuration was selected to offer the safest approach to initial criticality with the given control rod arrangement. With a cylindrical configuration, criticality was first achieved on January 21, 1958 with a loading of 1823 rods. Figure 20 shows a typical configuration in this geometry. Subsequent measurements indicated that the loading of the first critical "clean" core, with no control rod followers, would be 1847 fuel rods.

Initial investigations were confined to the determination of the power level of the assembly and the control rod worths. Preliminary values of the temperature coefficients of the assembly were also obtained. These measurements were necessary so that the safety aspects of the experiments could be assured.

The present lattice has a fuel rod pitch of 0.435 in. (square) giving a water/uranium ratio of 3:1 in which the water is assumed to be at room temperature. The uranium volume is obtained by taking the weight of contained uranium (U in UO_2) and dividing by the density of uranium metal, 18.7 gm/cm³. The water/uranium volume ratio is helpful in comparing the Yankee assembly with earlier water-metal lattices. The nominal height of fuel column is 48"; the fuel density is 10.18 gm/cm³. The fuel pellet diameter is 0.300"; the cladding thickness, 0.016". The cladding material is AISI 304 stainless steel (18% Cr- 8% Ni).

Fig. 20 CRX 3:1 Cylindrically Loaded Core With Water Slots and Detector

Flux plots for a cylindrically loaded 3:1 lattice were obtained from which preliminary values of the buckling of the lattice and the reflector savings have been determined. The material buckling is approximately $60 \times 10^{-4} \text{ cm}^{-2}$ resulting in inferred axial and radial reflector savings of 4.0 and 6.8 cm., respectively. The value of axial reflector savings is affected by the stainless steel end plugs in the fuel rods and the aluminum lattice plates which exclude water in the top and bottom reflector regions. The aluminum plates will not be present in the Yankee reactor.

Average Reactivity Worth of Peripheral Fuel Rods

Measurements have been made of the average reactivity worth of peripheral rods in critical assemblies of various sizes. The reactivity values can be accurately measured since control rod positions are repeated and the change in reactivity introduced by the addition of a given number of rods is measured by change in reactor period. For a cylindrical core having a diameter of approximately 22 in. (1880 fuel rods), the reactivity worth of each additional peripheral fuel rod is $7 \times 10^{-5} \Delta k/k$. The core was essentially "clean"-i.e., the assembly was critical with the control rods almost fully withdrawn from the core.

Temperature Coefficient of Reactivity Measurements

Measurements of the temperature coefficients of reactivity have been made for the 3:1 lattice in the range of 12°C to 75°C. Average values of temperature coefficient of reactivity have been obtained by measuring control rod position for the "just critical" condition at two temperatures. The total change in reactivity was then determined from the rod calibration. Measurements have also been made of the temperature coefficient at a particular temperature by allowing the moderator to cool while observing the change in reactor period. At 70°C. the temperature coefficient as measured by the latter method is $-1 \times 10^{-4} \Delta k/k\text{-}^\circ\text{C.}$ to $-2 \times 10^{-4} \Delta k/k\text{-}^\circ\text{C.}$

Control Rod Reactivity Worth Measurements

The critical assembly also was used to obtain reactivity worths of control rods. As mentioned above, the initial measurements were made with the "just critical" assembly to be certain of compliance with a self-imposed requirement that a 2% shutdown with control rods be in effect. From the standpoint of design of the reactor core, larger more heavily loaded cores with 9 control rods (the maximum number for the critical experiments) would yield more significant reactivity worth data. Two methods will be added shortly. The first method used is the classical "rod drop" method which has been found to be more practical with lower worth rods (0.5% to 5%) than with higher worth rods (in the neighborhood of 8%). Somewhat different results are obtained from detectors located at various points surrounding the assembly. Typical values from a single rod drop experiment are 3.5% where the detector is located at the top of the assembly and 3% using the detector on the horizontal center line. (The control rods drop into the assembly under the action of gravity in a "rod drop" experiment).

Control rod worths have also been evaluated by determining the incremental worth and summing over the length of the rod. Incremental worth is obtained by measuring the reactor period as a function of control rod position. It has been found practical to obtain as many as 3 or 4 points for the central control rod, with a given loading of fuel rods. To obtain another series of points it is, of course, necessary to shim the core by inserting (or withdrawing) additional fuel rods. The third method of evaluating control rod worths is by means of neutron poison addition to the moderator. The position of the central control rod, for example, will be determined as a function of the concentration of boron in the moderator. From these measurements the equivalent rod worth will be determined.

Values of control rod worth vary from a maximum of approximately 7.3% for the central rod in the "just critical" cylindrically loaded core to 3% for the central control rod with the heaviest loading so far employed, 3636 fuel rods. All control rods have a span of 7.8 in. (each "wing" has a length of 3.9 in.) while each "wing" or vane has a thickness of 0.285 in.

The neutron absorbing material in the control rods for the critical experiments is an alloy of silver and cadmium (70%-30%). As has been noted above, the nominal material for the control rods for the Yankee reactor is an alloy of silver, indium, and cadmium (80%-15%-5%). (The tertiary alloy has superior corrosion resistance and improved mechanical properties.) It was, of course, important to know the relative worths of the two alloys, and so a single cruciform control rod of silver-indium-cadmium was obtained. Measurements in central and peripheral positions in two lattices showed no substantial difference in reactivity worth between the two rods. The worth ratio of Ag-In-Cd to Ag-Cd was found to be 1.02 to 1.

Effect of Stainless Steel on Epithermal Neutron Flux

One further experiment using control rods is in progress. The object of the experiment is to determine the epithermal effects introduced by the stainless steel in the Yankee core system. The experimental arrangement uses a single control rod consisting of two test sections joined by an aluminum section. One test section consists of stainless steel of the same composition as the fuel cladding; the other test section is made of lithium-aluminum alloy of such composition that the macroscopic thermal absorption cross section to thermal neutrons is the same as the macroscopic thermal absorption cross section of the stainless steel. The change in the period of the reactor is obtained when the stainless steel is removed by withdrawing the control rod and thereby replacing the stainless steel with the lithium-aluminum cruciform. The position of the regular control rods are fixed for this experiment. For one particular core configuration the reactor period is 185 seconds (negative period) with the stainless steel in the core. With the lithium-aluminum alloy in the core, the period of the reactor was 29 seconds (positive). From this the reactivity difference is inferred to be .22%. From a comparison with Ag-Cd control rods it was concluded that the total worth of the stainless steel section was 1.0%.

11.0 RADIATION DAMAGE EXPERIMENTS

Mechanical and Thermal Design Section: E. Frisch, Consultant

This project involves the design and procurement by WAPD, 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 IN-PILE TEST LOOP

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

Stability Studies

The WCAP-4 loop heat regulating system was analyzed for stability and control response in the WAPD Computer Laboratory (Fig. 21). The solution provides for the simulation of "dead" time delays.

The loop was simulated in standard analog computer fashion with the major analytical effort devoted to the non-linear representation of the heat exchanger.

Standard commercial components were used as a basis for the temperature regulator simulation. The control functions were broken down into servo expressions, and the equations were entered into the analog computer. By doing this, it was possible to correlate the computer scaling factors and the knob settings on the controller. Thus, an examination of the response over the entire operating range of the regulator could be made.

A series of runs were made on the analog computer to show the response of the loop control system to a variety of disturbances which might be encountered during MTR operation. In each case, it was desired to regulate Tout to a constant value of 575°F. The first nine runs were made to determine the optimum settings of controller variables. In most cases, the disturbance to the system was assumed to be a change in reactor heat input. A step change, an example of the most difficult case, was used to exaggerate any control shortcomings.

Fig. 21 © APD Computer Laboratory

Figure 22 illustrates the "Loop Control System Response to Step Changes of Heat Input". The results show that it is theoretically possible to achieve satisfactory control response without endangering loop stability.

In-pile Loop Design and Procurement

Component designs and specifications for the loop were completed and submitted to a number of potential suppliers for fabrication bids.

On February 13, a conference was held to discuss the loop design and establish fabricator policies regarding administrative and technical responsibility. Attending were representatives of the potential fabricators, Yankee Atomic Electric Company, United States Atomic Energy Commission, and WAPD.

Loop design improvements of a minor nature continued. These changes will be incorporated in the drawings being sent to the fabricator.

The dimensional relationships of the in-pile pressure tube were obtained mathematically and the design finalized accordingly. A trip was made to the Materials Testing Reactor in Idaho to obtain project engineering approval of the design of the WCAP-4 in-pile test loop. As a result of recommendations made by Phillips personnel, some minor design changes were made. Figure 23 shows the in-pile pressure tube with the proposed test specimens.

On March 31, 1958, a contract was awarded to the Engineering Development Center of the Lummus Company in Newark, New Jersey, for the fabrication of the in-pile loop.

11.2 Performance of Radiation Damage Experiments

Process Water Irradiation Program Status

WCAP-1-3 and WCAP-2-1 were discharged from the Materials Testing Reactor December 9, 1957, and were shipped to KAPL on December 13, 1957. WCAP-1-3 had received a total of 2.06×10^{20} unperturbed nvt. WCAP-2-1 had received a total of 0.62×10^{20} unperturbed nvt. WCAP-1-1 and WCAP-1-5 were discharged from the reactor on December 30, 1957, and shipped to KAPL in January, 1958. WCAP-1-1 received a total of 0.59×10^{20} unperturbed nvt. WCAP-1-5 received a total of 1.18×10^{20} unperturbed nvt.

The remaining eight WCAP-1 and WCAP-2 specimens continue to be irradiated in the MTR. At the end of cycle 100, which terminated March 3, 1958, these capsules had received the following unperturbed nvt.

Fig. 22 WCAP-4 Loop Control System Response to Step Changes of Heat Input

TOP VIEW

SECTION B-B

SECTION A-A

RECOMMENDED ASSEMBLY PROCEDURE

1. WELD ITEMS 12 AND 13 TOGETHER
2. WELD ITEM 9 TO 12 AND 13
3. INSERT ITEMS 10, 11 AND 14 INTO ITEM 9 EXERCISING CARE TO LOCATE ENDS OF ITEMS 10 AND 14 INTO SLOTS IN ITEM 12
4. INSERT ITEM 8 INTO ITEM 9
5. WELD ITEMS 4 AND 6 TO ITEM 9
6. INSERT ASSEMBLED PARTS INTO ITEM 1 FROM BOTTOM
7. WELD ITEM 7 TO ITEM 2 AND ITEM 5 TO ITEM 2
8. WELD ITEMS ASSEMBLED IN STEP 7 TO ITEM 1
9. WELD ITEMS 4 AND 6 TO ITEM 2
10. WELD ITEMS 5 AND 7 TO ITEM 2

TITLE: WCAP-4 TEST LOOP TEST SAMPLE BLOCK ASSEMBLY				GROUP NO. & GROUP STYLE NO.			
DWG: 646J521				SUB: YXZ			
ITEM NO.	DESCRIPTION & MATERIAL	PART NO.	REV.	ITEM NO.	DESCRIPTION & MATERIAL	PART NO.	REV.
1	TEST SAMPLE BLOCK IT.1 414C486			10	TEST SAMPLE	GR.2 414C450	1
2	TOP PLATE IT.1 296B177			11	ADAPTER ASSY.	IT.1 195A258	3
3	BOTTOM PLATE IT.1 296B176			12	ELBOW ASSY.	GR.1 296B174	2
4	INLET PRESSURE TUBE GR.1 414C486			13	ELBOW ASSY.	GR.1 296B175	1
5	INLET JACKET ASSY GR.1 548D882			14	TEST SAMPLE	GR.1 414C480	1
6	OUTLET PRESSURE TUBE GR.1 414C495			15	TEST SAMPLE	GR.3 414C450	1
7	OUTLET JACKET ASSY GR.1 548D883			16	TEST SAMPLE	GR.4 414C450	1
8	SPRING IT.8 195A003			17	TEST SAMPLE	GR.1 548D046	1
9	TUBE IT.1 296B195			18	GUIDE PLATE	GR.1 296B664	1
				19	LIFTING PIN	IT.1 296B222	1

A. SUPPLIED BY WESTINGHOUSE

CONTINUED ON SHEET

PRESSURE TUBE ASSY. (ITEMS 4-6-8-9-10-11-12-13-14-15-16-17)
 DESIGN PRESSURE - 2500 P.S.I.
 DESIGN TEMP. - 650°F
 HYDROSTATIC TEST ITS 4,6,8,12,13 TO 4500 P.S.I. BEFORE INSERTING ITS 10,14,15,16,4,17

VESSEL TO BE MADE IN ACCORDANCE WITH THE A.S.M.E. CODE FOR UNFIRED PRESSURE VESSELS.
 PENETRANT DYE TEST PROC. SPEC. AP292187
 HYDROSTATIC TEST PROC. SPEC. AP291995-7
 PARTIAL RADIOGRAPH PROC. SPEC. AP292253
 CLEANING PROC. SPEC. AP291995-9
 WELDING PROC. SPEC. AP292252

JACKET ASSEMBLY (ITEMS 1-2-3-5-7-18-19)
 DESIGN PRESSURE - 100 P.S.I.
 DESIGN TEMP. - 150°F
 TEST PRESSURE - 150 P.S.I.

1. 646J521
 2. 646J521
 3. 646J521
 4. 646J521
 5. 646J521
 6. 646J521
 7. 646J521
 8. 646J521
 9. 646J521
 10. 646J521
 11. 646J521
 12. 646J521
 13. 646J521
 14. 646J521
 15. 646J521
 16. 646J521
 17. 646J521
 18. 646J521
 19. 646J521

FIGURE 23

WESTINGHOUSE ELECTRIC CORPORATION	
TITLE: WCAP-4 TEST LOOP TEST SAMPLE BLOCK ASSEMBLY	
DATE: 1/15/58	APPROVED: [Signature]
CHECKED: [Signature]	646J521
DIV. & PLANT LOCATION: ATOMIC POWER DEPT.	

<u>Capsule</u>	<u>Accumulated Unperturbed nvt</u>
1-2	4.9×10^{20}
1-4	4.9×10^{20}
2-2	2.6×10^{20}
2-3	1.9×10^{20}
2-4	2.3×10^{20}
2-5	2.3×10^{20}
2-6	1.6×10^{20}
2-7	2.3×10^{20}

Post-irradiation Examinations

The following is a summary of information obtained from observations of WCAP-1 and WCAP-2 specimens examined at KAPL on February 26 and 27.

The general appearance of the fuel tubes was good with no apparent gross distortion or cracking. Nothing unusual was observed at the interfaces of the pellets, regardless of the pre-irradiation pellet-to-clad annular clearance. The maximum designed cold annular clearance was 7.5 mils.

Most of the UO_2 pellets were cracked, but this was expected since under irradiation and/or thermal cycling, this condition has been invariably observed experimentally by others. Considerable care was taken not to crack or to aggravate any cracks in the pellets during sectioning. The crack pattern observed on the pellet interfaces was generally radial. In a few cases, the pellets appeared to be uncracked. However, none of the pellets could be removed from the cladding. Some of the difficulty in trying to remove the pellets was caused by a burr that formed on the inner-diameter of the stainless steel tube during sectioning. Attempts are being made to deburr some of the tubes in an effort to slide out the pellets. Molten zones and solidification craters were not observed in the fuel interfaces. Center melting of the fuel was not anticipated since the samples received a fraction of the neutron flux originally anticipated.

Specific remarks on the individual sectioned tubes are as follows:

1) Tube G-6 (capsule 1-3)

All but one of the pellets was cracked in this tube which had a 5 mil pellet-to-fuel annular clearance. The pellets were held firmly in the tube and could not be moved. The final parting of each section of this tube was accomplished by axially pulling the cut tube wall apart.

2) Tube 3-27 (capsule 1-1)

This particular tube has a 2.5 mil annular clearance. All the pellets were nominally in one piece, except for some chipping and slight cracking. One of the pellets which had

a 5 mil gap was observed to freely slide within the tube. It was not possible to remove this pellet, however, due to burring.

3) Tube A-3 (capsule 2-1)

This tube was designed with a 5 to 7.5 mil annular cold clearance. All but one of the pellets was cracked and two of the pellets were so badly fragmented that large pieces of UO_2 fell out after sectioning.

4) Capsule 1-5

Some time was spent in trying to identify the three tubes in this fuel bundle. The tube identifications appeared covered by a film or deposit.

Nuclear Science and Engineering Company has sent the results of the flux monitors for WCAP-1-3, 2-1, 1-5, and 1-1 capsules. The results are shown in the following table.

TABLE XIX

Results of Flux Monitors for WCAP-1-1, 1-3, 1-5, and 2-1 Capsules

<u>Capsule</u>	<u>Monitor</u>	<u>Enrichment</u>	<u>MWD/T</u>	<u>Atom Percent Fissioned</u>
1-1 (top)	A-8	2.7%	4.46×10^2	0.054
(bottom)	A-2	2.7%	1.19×10^3	0.144
1-3 (top)	A-1	2.7%	4.19×10^3	0.508
(bottom)	A-6	2.7%	5.28×10^3	0.640
1-5 (top)	C-4	5.4%	4.02×10^3	0.487
(bottom)	B-7	5.4%	5.93×10^3	0.718
2-1 (top)	D-1	5.4%	2.66×10^3	0.322
(bottom)	C-5	5.4%	2.85×10^3	0.345

The additional MTR process water experiments designated WCAP-1-6 and WCAP-1-7, discussed in the previous report, have been expanded to a larger experimental program to supplement the results obtained on the WCAP-1 and WCAP-2 experiments. A total of six capsules will be made for this new experiment. The post-irradiation now being done on the WCAP-1 and WCAP-2 experiments will be modified since these specimens received less than the unperturbed flux desired, and these samples no longer represent the reference design. Plans are to move the remaining capsules in the MTR into higher flux holes to obtain the requested flux as dictated by design requirements.

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 first quarter of 1958.

APPENDIX

Section A - Design Data

Yankee Atomic Electric Company Reactor

I. Mechanical and Thermal

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

	<u>First Core</u>
Total Heat Output, Btu/hr	1338×10^6
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	2000
System Pressure, Maximum	2150
System Pressure, Minimum	1850
System Pressure, Design	2500
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	1.47×10^6
(Jens & Lottes Correlation)	
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	4200
Outlet of Hot Channel	603
△ T _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.

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

2.4 liter treatment.

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.	1000

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

II. Nuclear

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

First Core

Core materials (Between Fuel Ends)

	<u>Volume, in³</u>	<u>Wt., lbs.</u>
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)	<u>4,900</u>	<u>-</u>
TOTAL	398,900	71,690

B ² , geometric, cm ⁻² (Assumes Reflector Savings of 7.5 cm)	7.128 x 10 ⁻⁴
Volume Ratio, H ₂ O/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,	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 contents (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 B/liter of H ₂ O	1.2

Section B - Information Availability

Westinghouse Atomic Power Department

During the previous periods of the R&D Program, the availability to (W) 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 by 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.

The availability of reports and documents during the first quarter of 1958 has continued to improve. The report situation is now very satisfactory. 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 received by (W) APD as a result of being on the "Standard" and "Positive" distribution lists during the first quarter of 1958 were:

<u>Month 1958</u>	<u>Unclassified</u>	<u>Classified</u>
January	609	43
February	688	66
March	812	41

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

<u>Month 1958</u>	<u>Ordered</u>	<u>Received</u>
January	0	3
February	13	7
March	14	9

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

7 - Pertains to Naval Reactors and not released to Civilian Application Program.

8 - Report has not been made available to TISE.

8 - Not in TISE files; are checking authors.

Yankee Atomic Electric Company

The large proportion of reports requested by the Yankee Electric Company, which have been received in the past quarter, has been encouraging and is now 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 - to date - 3/31/58

Total Documents Received	323
Total Documents Retained	153
Total Documents Returned	170
Total Documents Requested	3

Summary of Requested Unclassified Documents - to date- 3/31/58

Total Documents Requested	288
Reports Received and Retained	252
Reports Received and Returned (on loan)	6
Reports Not Yet Received (Includes 16 reports for which some notification was received i.e. not at TISE, no copies available, etc)	25
Reports denied	5

Summary of Unrequested Unclassified Reports Received on Standard Distribution

Total Reports Received	125
Reports Retained	70
Reports Returned	55

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: Knolls Atomic Power Laboratory, Schenectady, N. Y. (January 13, 1958)
Visit made by:

R. Winchell - (W) APD

Activation, grain growth, sintering and plastic flow of UO_2 powder were discussed. Information was obtained on KAPL's experience with extruding UO_2 and UO_2 mixtures.

To: Revere Copper and Brass Company, Detroit, Michigan (January 14, 15, 16, 1958)
Visit made by:

E. S. Foster - (W) APD

Extrusion of the silver-indium-cadmium billet received from the American Smelting and Refining Company into a cruciform shape for use as a test control rod was observed. The problems connected with the draw-forming of the cruciform to final thickness were discussed.

To: Bettis Plant, Pittsburgh, Pa. (February 7, 1958)
Visit made by:

R. M. Watkins - (W) APD

Information was obtained on the use of decontamination solutions in the SIW reactor and the Bettis Plant 29 Loop in the MTR.

To: C. L. Gougler Machine Company, Kent, Ohio (February 7, 1958)
Visit made by:

E. S. Foster - (W) APD

W. J. Hartnett (W) APD

Manufacturing problems that could delay delivery of the two CRX test control rods (epithermal and absorption combination, silver-indium-cadmium) were resolved. Fabrication of the two control rods was discussed.

To: Combustion Engineering, Inc. Windsor Locks, Connecticut (February 10, 1958).
Visit made by:

P. W. Davison - W APD

T. J. Thompson - M.I.T

J. E. Howard - YAEC

L. E. Minnick - YAEC

The combustion Engineering critical facilities were inspected and critical experiment methods were discussed. The facilities include a polyethylene moderated assembly and a water moderated assembly. Flux plotting techniques and control rod worth measurements were discussed as well as their experimental results with silver-indium-cadmium control rods. Critical experiments were started in June, 1956 and are continuing.

To: Materials Testing Reactor, Idaho Falls, Idaho (February 12, 1958)
Visit made by:

M. A. Vogel - (W) APD

Design changes to be made to the present MTR balcony drawings to suit WCAP-4 test loop equipment cubicle designs, installation costs, the primary cubicle design, specifications and drawings for the test loop were discussed.

To: Materials Testing Reactor, Idaho Falls, Idaho (February 13, 1958)
Visit made by:

R. P. Goldstein - (W) APD

Instrumentation and control specifications, the primary detector cubicle design, and wiring diagrams for the WCAP-4 In-Pile Test Loop in the MTR were discussed.

To: Bettis Plant, Pittsburgh, Pa. (February 14, 1958)
Visit made by:

G. H. Minton - (W) APD

Information concerning the availability of the costing rate for, and the application procedure to be followed to utilize, the Bettis Plant IBM 704 Computer was obtained.

To: Knolls Atomic Power Laboratory, Schenectady, N. Y. (February 25, 26, 1958)
Visit made by:

J. J. Lombardo - (W) APD

R. K. McGeary - W APD

L. E. Minnick - YAEC

Information was presented on the design, irradiation and fabrication history of the WCAP-1 and WCAP-2 capsules. The relative advantages of using flux wires, end cap fission monitors or isotopic analyses of the fuel itself to determine the amount of burn-up of the sample, the examination and evaluation of cracked ferrule brazes, the advantages of post-irradiation testing of defective fuel tubes and types of fission gas measurements on future specimens were discussed. The results of the Hot Laboratory examination of the two capsules were obtained.

To: The Ferrotherm Company, Cleveland, Ohio (March 10, 1958)
Visit made by:

P. P. King - (W)APD

The assembly and brazing of the 36-tube, 95 inch long, ferrule spaced, fuel tube bundle sub-assembly was observed.