PROGRESS REPORTS

By
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Westinghouse Electric Corporation
Pittsburgh, Pennsylvania**

UNITED STATES ATOMIC ENERGY COMMISSION
Technical Information Service Extension, Oak Ridge, Tennessee

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Yankee Atomic Electric Company
Research And Development Program

YAEC-97

QUARTERLY PROGRESS REPORT

for the period

July 1 to September 30, 1958

by

R. W. Garbe
H. E. Walchli

Large Plant Engineering

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

November 14, 1958

APPROVED:

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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 July 1 to September 30, 1958 (under the Research and Development Program as presented in YAE Development Program Report YAE-61 Revision-1 for the second half of 1958.) An evaluation and the resulting conclusions of work performed are given for each project in which definitive progress has been made.

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

1. Obtaining promising fuel fabrication results from production scale investigations of simplified UO_2 preparations methods, presintering operation elimination, increasing pellet densities and reductions of UO_2 losses.
2. Successfully brazing a ninety-five inch long experimental fuel subassembly in which all joints, including outside ferrules and control rod rubbing strips, had brazed fillets.
3. Completing the fabrication of the last group of MTR process water irradiation samples and all of the in-pile test loop samples except for the specimens containing 27% enriched pellets.
4. Performing a nuclear analysis of various fuel assembly designs as part of an overall evaluation of fuel assembly bowing in the Yankee reactor.
5. Completing corrosion studies of primary plant materials in static autoclaves to aid in the selection of a pH control agent.
6. Developing a "Compromise Design" for the fuel assembly which incorporates a series of small design changes to eliminate the possibility of restricting control rod motion by interference due to a superposition of bowing of the fuel assembly and an adverse accumulation of mechanical tolerances.
7. Completing the calculation of the following first core kinetic parameters: moderator temperature coefficient, the Doppler temperature coefficient, the void coefficient, neutron lifetime and delayed neutron fraction.
8. Obtaining flux profiles, flux peaking, flux spectrum, peripheral fuel rod worth, void coefficient, temperature coefficient and control rod worth data from the Critical Reactor Experiments with a 3:1 water-to-uranium metal ratio core at the Westinghouse Reactor Evaluation Center.
9. Continuing instrumentation and individual component operational testing of the in-pile test loop.
10. Analyzing the results of the post-irradiation examination of the second group of process water samples and delivery of the last group of samples to the MTR.

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Section-A of the Appendix contains the latest nuclear, mechanical and thermal design data for the first core. The studies and calculations performed under Projects 2.0, 4.0, and 5.0 during the third quarter of 1958 were responsible for the changes made in these data from the data presented in Quarterly Progress Report) YAEC-87 covering the second quarter of 1958.

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

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Section-C contains abstracts of trip reports written as a result of visits made by Westinghouse APD and YAEC personnel during the third 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-61, Revision 1, outlines the Research and Development Program for the period from July 1 to December 31, 1958. Quarterly Progress Reports, YAEC-7, YAEC-13 (Revision 1), YAEC-20, YAEC-35, YAEC-44, YAEC-52, YAEC-65 and YAEC-87 describe and evaluate the work accomplished from the beginning of the program, June 6, 1956 to June 30, 1958.

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

1. To carry out the development of stainless steel UO_2 fuel rods to a point where successful operation of experimental elements in a test reactor has been demonstrated.
2. To improve the process of manufacturing sintered UO_2 fuel pellets for the rod elements and, to reduce fuel preparation costs.
3. To select a proven material or to demonstrate the feasibility of a material suitable for use in control rods for a long-lifetime pressurized water reactor core.
4. To obtain through analysis and experiments a thorough understanding of the nuclear physics aspects of the proposed stainless steel UO_2 pressurized water reactor.
5. To obtain an optimized preliminary design for a nuclear reactor and associated auxiliaries for a turbine electric power plant based upon the use of a stainless steel clad long-life core.
6. To determine the proper primary system materials, the required efficiency of the water purification system, the selection and use of a soluble neutron absorber and the minimizing of crud deposition through water chemistry analyses.

"Information Availability Report", YAEC-30 for the period from June 6, 1956 to September 30, 1957, and Quarterly Progress Reports, YAEC-44, YAEC-52, YAEC-65 and YAEC-87 for the third and fourth quarters of 1957 and the first and second quarters of 1958, covered the availability of information to, and the difficulties encountered by the Westinghouse Atomic Power Department and the Yankee Atomic Electric Company in their efforts to obtain documents and reports to further the success of the YAEC Research and Development Program. The five reports also included a section containing abstracts of trip reports made under the YAEC Research and Development Program.

A cut-away drawing of the positive engagement type control rod drive mechanism being procured under subproject 4.2 is shown on the following page.

LATCH TYPE MAGNETIC JACK
CONTROL ROD DRIVE MECHANISM

1.0 FUEL ELEMENT DEVELOPMENT

Chemistry 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

Simplified Fuel Fabrication Procedures

The subcontract to investigate the following fabrication procedures on a production scale with production equipment at the Westinghouse Materials Manufacturing Department is approximately half completed. These fabrication procedures were first found practicable at (W)APD on a laboratory scale:

1. The simplification of UO_2 preparation for pelletization
2. The elimination of the slow and costly presintering operation
3. The increasing of pellet densities to improve dimensional control
4. The reduction of UO_2 losses through improved dimensional control

Results reported by the (W)MMD on the subcontract thus far include:

1. The development of a simpler, less expensive, method for preparing UO_2 for pressing.

Formerly the UO_2 powder required all of the operations listed in Column A in its preparation for pressing. On the two materials (i.e., two different samples of UO_2) tried so far, every trial by the procedure indicated in Column B produced satisfactory pellets.

A (old)	B (new)
1. Ball mill 2 hrs., wet	1. Ball mill 2 hrs., wet, with 1% PVA + 2% Carbowax
2. Dry	2. Dry
3. Granulate	3. Granulate to - 20Mesh
4. Mix in PVA	
5. In another mixer, add H_2O	
6. Dry partially	
7. Granulate to - 8Mesh	
8. Dry thoroughly	
9. Granulate to - 20Mesh	
10. Add lubricant	

2. Promising indications that the presintering operation can be eliminated.

On the two types of UO_2 utilized on this subcontract thus far, elimination of presintering had no deleterious effect - in fact, hour glassing was less in non-presintered pellets. Larger scale trials - on pellets produced from the shorter material preparation procedure above - are to be conducted.

3. Data confirming that high sintering temperatures are much more effective than long sintering times in increasing pellet density.

Comparison of Figure-1 with Figure-2 shows that increasing the sintering temperature is more effective in increasing pellet density than is increasing the time at temperature. Note that the reduced rate of change of density with time after about three hours implies that long hold times will reduce pellet density variations - the major cause of dimensional control problems. Note further that each powder has, for practical purposes, a maximum density it will achieve at each specific temperature - suggesting that density (and, therefore, dimensional) control can be achieved by employing the sintering temperature at which the specific powder supplied will be asymptotic within the density range specified for pellets.

Investigation indicates that the pellet density specification could be increased above $93 \pm 1.5\%$ of the theoretical density, if only fresh UO_2 were employed. However, economical recovery of scrap pellets requires their oxidation to U_3O_8 powder and subsequent reduction to UO_2 powder, a procedure which thus far does not yield powders which produce high density pellets.

4. Powder differences from lot-to-lot manifest themselves in shrinkage variations which have prevented attempts to improve dimensional control, and thereby reduce UO_2 losses.

The problem of UO_2 powder variability is being investigated as described under Process Improvement Studies.

The remainder of the subcontract will concentrate on confirming the promising indications that raw material preparation and pellet sintering can be greatly simplified.

Process Improvement Studies

1. Non-Uniformity of UO_2 Powder

Powder variations from vendor to vendor and even lot-to-lot in a single series from a single vendor have impeded development; as each lot must be pilot run to determine whether previously established fabrication parameters apply.

FIGURE 1- UO_2 PELLET DENSITY VS. SINTERING TIME @ 1500°C

FIGURE 2 -
 UO_2 PELLET DENSITY VS. SINTERING TIME @ 1600°C

The presently existing powder specification is being refined and negotiations are in progress with powder suppliers to persuade them to guarantee reasonable uniformity by means of a performance requirement. Each supplier is asked to agree that his powder will always press to the green density the supplier stipulates (pressure may be varied), when prepared in some standard manner, and that these pellets when sintered under agreed upon conditions will always fall within a certain range of densities. Each supplier may choose his own conditions, but must meet those conditions with every lot, to assure the pelletizer of uniform raw material. In general, the suppliers are agreeable, but the specific values and procedures have not been developed.

2. Atmospheres and Additives to Enhance UO_2 Pellet Densification

In efforts to reduce sintering costs and improve density and dimensional control by lowering sintering temperatures, work was begun on steam sintering (replacing the normal furnace atmosphere of H_2 with steam, above 1200°C . and the use of minor additives in improving densification. Steam sintering shows promise, but results are erratic, being not duplicable and sometimes showing decreased density with increased time or temperature. Similarly, the results with additives are inconsistent, and the pellets show a tendency to crack and to hour-glass to an unusual degree. At present, it has not been determined whether the cracking and hour-glassing are mechanical (a result of material preparation or pressing) or compositional (related to the additives). The Spencer Chemical Company has supplied samples which are guaranteed to be duplicable and to sinter to 10.2 g/cc (about 93% T.D.) at only 1250°C . in 3 hours in hydrogen. Since the work with atmospheres and additives cannot be completed in time to reduce pellet costs for the Yankee First Core, and the Spencer UO_2 may accomplish the purpose, the work will be discontinued for this project, except for the testing of Spencer UO_2 powders.

a. Steam Sintering

Typical results of steam sintering are shown in Table I and Figure 3. The reduction of powder batch HLOI density when sintering time is increased from two to three hours is attributed to experimental error.

b. Additives

Four different UO_2 powders were ball-milled, each with an addition of 0.179% TiO_2 and 0.321% CaO , pelletized, and sintered, with the results shown in Table II.

3. "Hour-Glassing" Studies

An important reason for centerless grinding of pellets is the occurrence of "hour-glassing" (smaller diameter at the middle of the cylinder than at the ends) in unground pellets. To the unaided eye,

TABLE I
RESULTS OF STEAM SINTERING UO₂ PELLETS

Powder 01B

<u>Green Density</u> <u>g/cc</u>	<u>Temperature</u> <u>°C.</u>	<u>Time</u> <u>(hrs.)</u>	<u>Steam Sintered Density</u> <u>g/cc</u>	<u>% T.D.</u>
6.24	1200	3	9.67	88.3
6.35	1250	3	10.1	92.2
6.35	1300	3	10.51	96.0
6.36	1400	3	10.51	96.0

Hydrogen Sintered Density

6.35	1500	4	10.02	91.5
------	------	---	-------	------

Powder HL01

<u>Green Density</u> <u>g/cc</u>	<u>Temperature</u> <u>°C.</u>	<u>Time</u> <u>(hrs.)</u>	<u>Steam Sintered Density</u> <u>g/cc</u>	<u>% T.D.</u>
6.65	1200	3	8.29	75.7
6.62	1250	3	8.56	78.2
6.68	1300	3	8.89 *	90.3 *
6.65	1350	3	9.48 **	86.6 **
6.67	1400	3	9.55	87.2

Hydrogen Sintered Density

6.65	1700	3	9.65	88.1
------	------	---	------	------

* Only one pellet

** Only 2-1/2 hrs. (furnace failed)

Figure 3 Effect of Furnace Atmosphere Upon UO_2 Pellet Densification

TABLE II

RESULTS OF SINTERED DENSITY WITH AND WITHOUT ADDITIVES

Powder	Green Density g/cc	Temp. °C.	Time hrs.	Sintered Density, g/cc	
				With Additive	Without Additive
O2B	6.63	1500	4	10.22 - 10.31 ^A	10.35 ^B
	6.63	1400	4	10.12 - 10.17 ^A	E
O1B	6.56	1500	4	9.95 - 10.27 ^B	10.21 ^B
RX05	6.59	1500	4	10.37 - 10.36 ^B	9.8 ^B
	6.59	1400	4	9.30 - 9.37 ^B	E
SG06B	6.52	1500	4	10.23 - 10.25 ^C	8.3 ^F
	6.27	1400	4	10.01 ^D	E
	6.59	1400	4	9.43 ^D	E

A. Pellets cracked. Density determined by water immersion

B. Good pellets. Density determined by measuring and weighing

C. Pellets evidenced slight lamination. Density, determined by measuring and weighing, is probably higher

D. One pellet

E. No data yet

F. Extrapolated

many hour-glassed pellets appear to be generally straight, except for a short length of narrower "waist". Figure-4 shows the true shape of hour-glassing in a UO_2 pellet.

Pellets of two different types of UO_2 powder were produced with L/D ratios of 1, 1.5 and 2 (Tables III, IV, and V) at forming pressures of 20, 40, 61, and 81 TSI, with 0.2%, 0.6%, and 1.0% Sterotex Z, using the unabbreviated UO_2 preparation process, to determine the effects of various factors so that hour-glassing may be eliminated or minimized.

1. As expected, hour-glassing increases as the L/D ratio increases, and differs from powder to powder.

<u>Powder</u>	<u>Average Mils of Hour-Glassing for L/D Ratios of</u>		
	<u>1</u>	<u>1.5</u>	<u>2</u>
HL01	1.0	1.9	2.5
BY01B	0.7	0.9	1.0

2. In general, hour-glassing is reduced by increasing the amount of lubricant - this effect being much more definite and consistent with the HL01 powder than with the BY01B powder.

<u>Powder</u>	<u>Average Mils of Hour-Glassing in 36 Pellets with Sterotex Z Percentages of</u>		
	<u>0.2</u>	<u>0.6</u>	<u>1.0</u>
HL01	2.4	1.9	1.2
BY01B	1.0	0.8	0.8

3. Increased time of application of forming pressure appeared to increase hour-glassing of the HL01 powder and to be inconsistent with the hour-glassing of the BY01B powder (even the minimum time employed in these trials is longer than the maximum time in production pressing).
4. Hour-glassing appears to be less for both powders when pressed to an intermediate pressure, i.e. 40 TSI, rather than at higher or lower pressures.

<u>Powder</u>	<u>Average Mils of Hour-Glassing at Forming Pressures of</u>			
	<u>20 TSI</u>	<u>40 TSI</u>	<u>60 TSI</u>	<u>80 TSI</u>
HL01	1.86	1.57	1.69	2.08
BY01B	0.91	0.74	0.80	1.23

The final draft of topical report YAEC-84 "Progress in the Development of a Process for Producing UO_2 Pellets" is being prepared covering the progress made under this subproject as of July 1, 1958.

Pellet A Powder: BYOLB (MCW)
Pressed at 10.4 tsi, to 5.2 g/cc
Sintered at 1750° C./4 hrs. to 10.32 g/cc

Figure 4 Hour Glassing of UO_2 Pellet

TABLE III

MILS OF HOUR-GLASSING** - L/D = 1.0

Powder	Pressing Time (Seconds)	Pressure (TSI)				Additive
		20	40	60	81	
HL01	1	1.8	0.9	1.1	1.3	0.2% Stero-tex
"	7	1.6	1.0	1.0	1.4	"
"	20	1.8	1.5	1.2	1.8	"
"	1	1.7	0.5	0.7	0.5	0.6% Stero-tex
"	7	0.7	0.7	1.2	0.8	"
"	20	2.0	1.6	1.3	1.1	"
BY01B	1	1.1	0.5	0.5	0.7	0.2% Stero-tex
"	7	0.4	0.3	0.2	0.6	"
"	20	0.3	0.2	0.2	0.6	"
"	1	2.0	1.5	0.5	*	0.6% Stero-tex
"	7	0.5	0.6	1.6	*	"
"	20	0.4	0.6	0.5	*	"
"	1	1.3	0.7	1.1	*	1.0% Stero-tex
"	7	1.2	0.5	0.7	*	"
"	20	0.7	0.6	0.5	*	"

* Could not press - laminated

** Difference in mils between maximum diameter found on end measurements minus minimum diameter found from center measurement.

TABLE IV

MILS OF HOUR-GLASSING - L/D = 1.5

<u>Powder</u>	<u>Pressing Time (Seconds)</u>	<u>Pressure(TSI)</u>				<u>Additive</u>
		<u>20</u>	<u>40</u>	<u>61</u>	<u>81</u>	
HL01	1	2.6	1.9	2.3	3.0	0.2% Stero-tex
"	7	2.4	2.1	2.8	3.3	"
"	20	1.9	2.2	2.6	3.8	"
"	1	2.4	1.2	1.7	2.0	0.6% Stero-tex
"	7	2.2	1.3	1.5	2.6	"
"	20	2.1	2.1	2.2	2.7	"
"	1	1.2	1.3	1.3	1.4	1.0% Stero-tex
"	7	1.4	1.1	0.9	1.3	"
"	20	1.5	1.0	1.0	1.4	"
BY01B	1	0.3	0.4	1.2	2.0	0.2% Stero-tex
"	7	0.8	0.1	1.0	0.7	"
"	20	1.2	0.8	1.4	0.9	"
"	1	1.2	0.8	1.0	*	0.6% Stero-tex
"	7	0.4	0.9	0.8	*	"
"	20	0.8	0.6	0.7	*	"
"	1	1.1	1.5	0.5	*	1.0% Stero-tex
"	7	1.1	0.7	0.4	*	"
"	20	1.8	0.5	0.2	*	"

* Could not press - laminated

TABLE V

MILS OF HOUR-GLASSING: L/D - 2.0

<u>Powder</u>	<u>Pressing Time (Seconds)</u>	<u>Pressure (TSI)</u>				<u>Additive</u>
		<u>20</u>	<u>40</u>	<u>61</u>	<u>81</u>	
HL01	1	2.7	2.2	2.8	3.7	0.2% Stero-tex
"	7	2.6	2.8	3.5	4.5	"
"	20	2.4	3.3	4.2	4.6	"
"	1	2.6	2.3	2.4	2.7	0.6% Stero-tex
"	7	2.4	2.1	2.6	2.7	"
"	20	2.4	2.1	3.0	2.5	"
"	1	2.1	1.3	1.8	2.1	1.0% Stero-tex
"	7	1.7	1.9	1.6	2.0	"
"	20	1.6	1.7	1.1	2.4	"
BY01B	1	0.5	0.9	1.2	2.2	0.2% Stero-tex
"	7	0.7	0.4	1.2	2.0	"
"	20	1.1	1.0	1.7	1.4	"
"	1	0.8	1.3	1.0	*	0.6% Stero-tex
"	7	0.9	1.0	1.0	*	"
"	20	0.8	1.0	0.9	*	"
"	1	1.3	1.0	0.7	*	1.0% Stero-tex
"	7	0.7	1.1	0.6	*	"
"	20	1.2	0.5	0.4	*	"

* Could not press - laminated

1.3.1 End Closure of Fuel Rods

G. G. Lessmann

Over 100 welds were made in checking parameters and consistency of welding with the AIRCO automatic welding head. Excellent metallographic results that have been consistently obtained are attributed primarily to the control of welding voltage permitted through use of the automatic head. Fit-up of the end plug to the tube is not as critical a problem with the new head because close control of arc voltage permits welding even through open plug-tube interfaces. Numerous samples considered to have poor fit-up have been successfully welded without discs brazed in place near the end plugs. Considerable difficulty, however, was encountered as a result of high air pressure build-up in the end plug-disc compartment. During welding, the pressure in the small volume becomes heated because it is confined, forcing air out through the molten weld causing blowholes to form. Blowholes, due to this pressure build-up, have been practically eliminated by a multiple end plug pressing-repressing operation which limits the interference fit enough to eliminate the pressure build-up.

1.3.2 Joining Fuel Bundles into Assemblies

P.P. King

J. R. Dazen

M. D'Amore

Brazed Joint Studies

A 95 inch experimental Yankee type subassembly was successfully brazed at $1890 \pm 10^\circ\text{F}$. for three hours, then furnace cooled. The maximum bow along the length was 0.034" and the maximum spread in cross section was 0.014". All of the joints, outside ferrules, and control rod rubbing strips showed brazed fillets. An overall view of the subassembly is shown in Figure-5. A close-up view of the external bumper ferrules and the rubbing strips is shown in Figure-6. Table VI shows a compilation of data for 14 brazed bundles made to date. These data indicate for the last several bundles a close control of dimensions and practically no unbrazed joints. Jigging and fixturing methods are still being altered for each run in order to improve fit-up of parts which has been found to be the major factor determining final dimensions and integrity brazed joints.

Dimensional surveys and joint-strength studies indicate that two brazing techniques appear feasible for subassemblies. One technique is chemically plating 0.0005 inch of Ni-P on the ferrules backed up with Microbraz 50 slurry and the other technique involves plating 0.001 inch of Ni-P on the ferrules. Bundles X-4 and X-5 using each technique with variations in ferrule arrays have been inspected and tested with results shown in Table VII.

Figure 5 Yankee Fuel Element Sub-Assembly (95") Showing Rubbing Straps and Outside Ferrules

Figure 6 Close-up View of Yankee Fuel Element Sub-Assembly

TABLE VI
QUALITY SURVEY OF BRAZED FUEL SUBASSEMBLIES

Braze Location	Bundle Length	Fixture Designer	Braze Alloy and Cycle	Dimensional Survey of Subassemblies					Fracture Load Data					Remarks
				Max E/w	Max Cross Sect.	Min Cross Sect.	Spread	Nom Design	No Jnts Pulled	Avg Fracture Load lbs.	% Jnts. be- low 200#	% Jnts. Not Braze	% Jnts. Above 200#	
2/57 Wall Colmonoy Corp. Detroit, Mich.	18"	Wall Colmonoy	Microbraz 45 30 min 2100°F		2.492	2.410	0.082	2.450						
2/57 Wall Colmonoy Corp.	18"	Wall Colmonoy	Microbraz 45 30 min 2100°F		2.499	2.409	0.090	2.450						
3/57 Wall Colmonoy Corp.	52"	Wall Colmonoy	Microbraz 45 30 min 2150°F	0.070	2.251	2.496	0.025	2.495						
3/58 Wall Colmonoy Corp. 548D849	56"	W APD	Microbraz 50 1 hr 1850°F	0.034	2.555	2.527	0.028	2.537	20	380	0	0	100	Fracture load data on samples subassembly used for display.
3/58 Wall Colmonoy Corp. 548D850*	56"	W APD	Microbraz 50 1 hr 1850°F	0.048	2.546	2.528	0.017	2.537	12	410	0	0	100	Subassembly used for display.
3/58 Wall Colmonoy Corp. 548D851*	56"	W APD	Microbraz 50 1 hr 1850°F	0.054	2.546	2.529	0.017	2.537	139	360	28	2.2	73	Actual bundle destructively tested after 28 thermal cycles in an autoclave heated to 650°F then cooled to room temperature.
3/58 Wall Colmonoy Corp. 548D852*	56"	W APD	Microbraz 50 1 hr 1850°F	0.034	2.546	2.530	0.016	2.537	12 (Samples)	300	0	0	100	Qualification pieces tested. Sub-assembly used for a display bundle
12/57 Ferrotherm Company Yankee #1	95"	Ferrotherm	Microbraz 50 35 min 1840°F	0.038	2.503	2.412	0.091	2.462	160	178	72.5	2	27.5	Horizontally brazed
3/58 Ferrotherm Company Yankee #2	95"	Ferrotherm	Microbraz 50 1 hr 1850°F	0.055	2.465	2.412	0.053	2.462	24 (Samples)	100	0	2.4	0	Qualification pieces tested. Assembly still intact. Poor hydrogen supply caused surface oxidation of bundle and caused poor joints.
3/58 Westinghouse X-1	56"	W APD	NIP (0.0005") + Microbraz 50 1 hr 1850°F	0.015	2.546	2.528	0.018	2.538	284	320	45	0.2	55	
6/58 Westinghouse X-2	56"	W APD	NIP (0.0005") + Microbraz 50 3 hrs 1875°F	0.017	2.757	2.742	0.015	2.752	162	412	19	0	81	
8/58 Westinghouse X-3	56"	W APD	NIP (0.0005") + Microbraz 50 3 hrs 1875°F	0.026	2.759	2.743	0.016	2.752	Strength data not taken because of improper parts fit-up in brazing fixture.					
8/58 Westinghouse X-4	56"	W APD	NIP (0.0005") + Microbraz 50 3 hrs 1875°F	0.015	2.752	2.743	0.009	2.752	222	512	14	0	86	
8/58 Westinghouse Yankee #3	95"	W APD	NIP (0.0005") + Microbraz 50 3 hrs 1875°F	0.034	2.555	2.541	0.014	2.542						Used for deflection studies by Mechanical Design Group.

* 548D850, 548D851, and 548D852 were originally brazed in November of 1957 but had over 5/32" of bow along the length, due to fixture design. An attempt was made to straighten the subassemblies and they were then used as a display fuel bundle. Substitutes of these bundles were then made using fixtures designed by APD and they are listed above.

TABLE VII
COMPARISON OF BRAZING TECHNIQUES

Both brazed at 1875°F for 3 hours		X-4 0.0005" NiP + NB50	X-5 0.001" NiP
Alternating layers of arrangements A and B.	Layers 1, 2, 3 with ferrules in all spaces. Layers 4 & 6 per arrangement A; Layers 5 & 7 per arrangement B.
<div style="display: flex; justify-content: space-around; width: 100%;"> A B </div>			
<u>Maximum bow along length</u>		Approx. 0.013"	Approx. 0.014"
<u>Design Size</u>		2.752 x 2.752	2.752 x 2.752
Maximum Size		2.752	2.757
Minimum Size		2.743	2.742
Spread		0.009	0.015
<u>Design Spacings, Tube-to-Tube</u>		0.136	0.136
Maximum Spacing		0.154	0.147
Minimum Spacing		0.117	0.127
<u>Spread</u>		0.037	0.020
<u>Spread on Layers, 1, 2, 3 (X-5)</u>		--	0.010
<u>Number of Unbrazed Joints</u>		0	0
<u>Percentage of Joints below 200 lbs</u>		16.9%	14%
<u>Average Fracture Load</u>		485 lbs	360 lbs
<u>Percentage of Zero Load Joints</u>		3.95%	3.30%

Although the Ni-P + Microbraz 50 joints gave a higher average fracture load, the plated ferrule joints gave a higher percentage of joints with fracture loads above 200 pounds. In addition, the plated ferrule bundles were cleaner, less subject to base metal erosion, and free of brazing to fixture members because of run-off of excess braze material. Plated ferrules required about one-half the assembly time compared to the slurry method. None of the ferrules slipped out of position significantly when vertically brazed.

Conclusions drawn from the brazed joint studies are as follows:

1. Longitudinal bowing and cross-sectional dimensions are not affected by the brazing technique.
2. Tube-to-tube spacing variations can be cut in half if ferrules are placed between each fuel tube.
3. Joints brazed with 0.0005 inch of plated Ni-P and Microbraz 50 gave an average fracture load 125 pounds higher than with the 0.001 inch plated ferrule technique although the latter can probably be increased by increased time at temperature.
4. Cleaner, more consistent fuel elements can be made faster by using 0.001 inch of Ni-P.
5. Corrosion studies indicate that the corrosion resistance of Ni-P is equivalent to Microbraz 50 and AISI-304 stainless steel.

Stretch Forming Studies

Experimental loading studies were continued with the use of eight full length Yankee-type fuel tubes. It was found that vacuum drawing of the pellets into the tube was required since a column of pellets longer than 5 to 6 feet produced excessive friction. Pellet chipping tendencies and loading friction have been reduced by the use of the vacuum technique in the loading operation. Planning has been directed toward incorporating the loading operation in the same machine that will stretch the tubes.

Much of the detailed statistical data obtained for the Belgian Thermal Reactor stretch-forming operation have been analyzed and found to be applicable to the Yankee fuel element. From these data, Tables VIII and IX have been obtained showing a tentative stretching procedure for 15 and 21 mil wall tubing depending on the cold tube-pellet gap and axial clearance desired.

Corrosion Coupons

A total of 175 stainless steel AISI-304 corrosion coupons (2-1/8" x 3/4" x 1/8") were prepared. Of these, 133 coupons were given the following heat treatment prior to corrosion testing in dynamic loops.

TABLE VIII

YANKEE STRETCH-FORMED FUEL ROD

Tentative Stretching Procedures for 0.015" Wall Thickness Tubing

Dimensions and tolerance ranges of components and the required stretching methods to obtain three different pellet-tube clearance ranges after stretch are itemized below for a compartmented fuel rod. The pellet-tube ranges listed below are:

- 0 clearance: Tube and pellets are in contact.
- 0-2 mils clearance: Tube is contacting largest pellets only.
- 0.9 - 2.9 mils clearance: Tube is not in contact with any pellet.

Pellet-Tube Clearance (After Stretch)	0 mils	0-2 mils	0.9 - 2.9 mils
<u>Components</u>			
Tube I.D. required (Before Stretch)	0.299 - 0.300"	0.299 - 0.300"	0.2995 - 0.3005"
Pellet Diameter	0.2935 - 0.2945"	0.2935 - 0.2945"	0.2935 - 0.2945"
Disc Diameter Required	0.2958 - 0.2962"	0.2968 - 0.2972"	0.2973 - 0.2977"
<u>Stretching Method</u>			
Percent Stretch with Ram	1.8% (2.5 mils)*	1.10% (1.5 mils)	No ram needed
Percent Stretch without Ram	2.60 (3.56 mils)	2.15% (3.0 mils)	3.15% (4.1 mils)
Total	4.40	3.25	3.15
<u>Stretch Tube Characteristics</u>			
Maximum Tube Diameter	Pellet-Tube	0.2955	0.2964
Minimum Tube Diameter	Contact Achieved	0.2945	0.2954
Pellet Expansion Space (length)	2.45%	2%	3%

* Figures in parentheses indicate the amount tube will reduce for the listed percentage of stretch.

TABLE IX

YANKEE STRETCH-FORMED FUEL ROD

Tentative Stretching Procedures for 0.021" Wall Thickness Tubing

Pellet-Tube Clearance (After Stretch)	0 mils	0-2 mils	1-3 mils
<u>Components</u>			
Tube I.D. (Before Stretch)	0.2990" - 0.300"	0.299" - 0.300"	0.2995" - 0.3005"
Pellet Diameter	0.2935 - 0.2945	0.2935 - 0.2945	0.2935 - 0.2945
Disc Diameter	0.2958 - 0.2962	0.2968 - 0.2972	0.2973 - 0.2977
<u>Stretching Method</u>			
With Pellet Retaining Ram	1.75% (2.45 mils)*	1.10% (1.54 mils)	No ram needed
Without Ram	2.50% (3.55 mils)	2.15% (3.01 mils)	2.85% (4.0 mils)
Total	4.25	3.25	2.85
<u>Stretched Tube</u>			
Expansion Space for Pellets in Stretched Compartments.	2.35%	2.0%	2.7%
Max. Tube I.D.	0.2945	0.29545	0.2965
Min. Tube I.D.	Pellet Tube Contact	0.29445	0.2955

* Figures in parentheses indicate amount tube I.D. will reduce for the listed percentage stretch.

No. of CouponsCondition or Heat Treatment

25	as-received
25	polished
25	annealed at 1850°F.
25	50% weld coated and annealed at 1850°F.
33	sensitized

Results of the corrosion tests are not yet available.

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

J. J. Lombardo

Inspection of the 24-inch and 30-inch long MTR samples after brazing indicated a condition of general bowing in both samples. These samples were brazed in a vertical-position furnace and were straight in the jig when the jig was removed from the furnace. However, when the brazed bundle was removed from the jig, it sprang into a bowed shape. When the same jig was used for brazing in a horizontal-position furnace, no such bowing was encountered. The explanation of this behavior is believed to be as follows: during vertical brazing, pellets fell to the bottom of the tubes; during cooling from the brazing temperature, the greater contraction of the tubing over the solid pellet column produces elastic stresses which are not equally distributed; when removed from the straight jig, stresses are released resulting in bowing of the bundle. This same behavior can not occur during horizontal brazing because a solid pellet column does not form.

The following pellets were prepared for use in the MTR in-pile loop tests:

U-235 EnrichmentNo. of Pellets

Depleted	1200
2.7	50
5.4	175
6.8	50
10	160
27	130

The 27% enriched pellets were subsequently found to chip and fracture upon fuel rod stretch-forming. After reprocessing and repressing, the new pellets were pressed and underwent fuel rod stretch-forming satisfactorily.

All of the in-pile test loop and process water samples are now complete except for the in-pile loop specimens containing the 27% enriched pellets. The 27% enriched fuel specimen is ready to be brazed into a fuel bundle. A thorough examination was made and photographs were taken of all specimens. Photographs of the brazed bundles and control rod specimens are shown in Figures-7, 8, and 9 for the process water and in-pile loop tests.

WCAP-2 PROCESS WATER SPECIMENS

Figure 7 Fuel Bundle Specimens to be Irradiated in Process Water in MTR
(WCAP-2)

WCAP-4 LOOP TEST SPECIMENS

4-1

4-2

4-3

4-4

Figure 8 Fuel Bundle Specimen to be Installed in the WCAP-4 In-Pile Test Loop. Specimens 4-1 and 4-3 will be in-pile; Specimens 4-2 and 4-4 will be out-of-pile.

WCAP-4 LOOP TEST SPECIMENS

4-5

4-6

Figure 9 Arrangement of Control Rod Materials to be Installed in the WCAP-4 In-Pile Test Loop. Specimen 4-5 will be in-pile; Specimen 4-6 will be out-of-pile.

2.0 NUCLEAR DESIGN AND REACTOR PHYSICS

Nuclear Design Section:
Computer Analysis Group:
Reactor Physics Group:

H. W. Graves, Jr., Manager
G. H. Minton, Supervisor
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 third quarter of 1958.

2.2 Core Steady State Analysis

R. E. Wolf

R. G. St. Paul

R. J. French

Fuel Assembly Design

A nuclear analysis of various fuel assembly designs was made as part of an overall evaluation of fuel assembly bowing in the Yankee reactor. Two factors of importance from the nuclear standpoint were emphasized: flux peaking around control rod followers, and U-235 loading requirements. The results obtained to date are preliminary but do indicate trends for the several suggested designs. The following designs have been studied:

- (1) A shrouded assembly design in which either the control rod or fuel assembly is surrounded by a thin steel "box".

This design is acceptable from the nuclear standpoint if the dimensions in the drawings are maintained. The additional steel due to the shroud is equivalent to a 3 mil increase in clad thickness, but if clad thickness is reduced to 15 mils, there would be a net reduction in steel from the definitive design. The effect on fuel cost of the changes in amount of steel in the core can be inferred from previous studies, which indicate that a 6 mil increase in clad thickness causes a 19% increase in the required U-235 loading and an increase in the nuclear fuel cost of 0.13 mills/KWH (for a 10,000 hour life, \$27/gm plutonium credit, and 4% investment charge).

- (2) A "3-bundle" fuel assembly design in which the rods are divided into three axial sections.

Axial flux peaking can be minimized in a 3-bundle design, but only at the expense of U-235 loading. Two cases with a one inch gap 30 inches from each end of a 90 inch long core were considered. In the first case, the water-to-solid ratio was nearly that of the active core (UO_2 replaced by stainless steel), while the second case had a considerably higher fraction of water. The first case requires roughly a 5% increase in U-235 loading to offset the poison effects of

stainless steel, although it does not have too great a flux peak because of the increased absorption due to the steel in the gap. The second case results in an increase in the axial peak/average power density of 65%. For a given lifetime, slightly higher reactivity would be required in this core due to the decrease in epithermal conversion of U-238. Both of these designs would have a more positive temperature coefficient.

- (3) An "hour-glass" fuel assembly design in which the fuel rod spacing is closer at the midplane of the core than at the top and bottom plates.

Two sets of calculations were made to analyze the effect of lateral movement of a subassembly toward a zirconium control rod follower. Some lateral movement actually decreases the flux peaking since the "water holes" are more uniformly distributed when the movement of the fuel subassembly decreases the follower gap and creates an internal gap in the assembly. However, the internal gap may cause a higher peak than the follower gap. The reactivity effect due to the lateral movement is very small, indicating that reactivity transients due to mechanical movement would be negligible.

Fuel Bearing Control Rod Followers

A general study was also performed on the Yankee core in an attempt to estimate the effect of various design modifications on flux peaking in the fuel adjacent to a control rod follower. The study included the dependence of flux peaking on water gap thickness, the use of steel plates between the followers and fuel and the effect of using fuel-bearing followers. The results of this study indicate that a significant reduction in power peaking can be achieved with fuel bearing followers. If a U-235-Zircaloy-2 alloy is used, the power output of the core can be increased by approximately 10%, whereas if low enrichment UO_2 - stainless steel followers are used, an increase in core rating of over 20% appears possible.

The flux peaking which exists when a Zircaloy-2 control rod follower is in the active region of the core is due to the low absorption of Zircaloy-2 relative to that of the fuel and to the increased thermalization of neutrons in the adjacent water over that in the neighboring fuel regions. Previous studies indicate that a Zircaloy-2 follower and adjacent water channel give rise to a power peak in the neighboring fuel of the order to 20 to 30%. This, correspondingly, reduces the maximum output of the core by 20 to 30% and aggravates the thermal bowing of fuel rods by increasing the radial temperature gradient near control rod channels. This peaking can be eliminated by increasing the neutron absorption properties of the control rod follower. If steel or another non-productive type of absorption is used, however, the loading requirements are increased and the control rod worth is reduced by a factor of almost two. A more practical solution to the problem is to use fissionable material in the control rod follower.

The cost of fuel-bearing control rod followers will be greater than for Zircaloy-2, particularly since replacement will be required during the life of the plant. This increased cost must be balanced by the definite increase in maximum power capability, the reduced thermal gradient around control rod slots, and the improved utilization of primary coolant in the core.

The main nuclear disadvantage resulting from the use of highly enriched material in control rod followers is the different burnup characteristics of high and low enrichment material. The effect of this variation of power peaking is shown in Table X.

TABLE X

Effect of Fuel Burnup on Local Power Peaking for a
U-235 - Zircaloy-2 Control Rod Follower

<u>Fuel Burnup - MWD/Ton</u>	<u>$P_{\text{edge}}/P_{\text{avg}}$</u>
0	0.99
5,000	1.08
10,000	1.14
20,000	1.22

(The initial fuel loading in the follower is 8.5 weight percent U-235)

In the calculations of Table X it was assumed that both the stationary fuel and the fuel contained in the follower were exposed to the same integrated flux. The 20,000 MWD/Ton burnup value is realistic for the peak burnup in a core which has an average fuel burnup of 8,200 MWD/Ton. The table shows that most of the improvement in power peaking is lost when the core is exposed to a high fuel burnup.

A comparable study of the burnup characteristics of control rod followers composed of low enrichment UO_2 in stainless steel rods indicates that a variation in flux peaking no greater than 6% would occur for burnups similar to those considered in the above table. Thus, it seems more desirable to use a follower consisting of two perpendicular rows of stainless steel tubes filled with low enrichment UO_2 . This type of cruciform fuel-bearing follower would be preferable to a similarly shaped alloy of Zircaloy-2 and highly enriched UO_2 .

Flux Peaking Analysis

An experimental and theoretical study to determine the accuracy of various methods of calculating thermal neutron flux peaking in water channels and at the core-reflector interface was begun. Experiments were made to measure the flux in a central water slot and at the core edge of a critical assembly using both foils and fuel. A theoretical investigation involving the examination of various calculating procedures to obtain the desired flux plots was started. In performing these calculations for a few-group model, the thermal and fast group constants must be obtained and then used in one of the operational multi-region flux distribution computer codes. Of the many approximations available in such a calculation, only those considered most theoretically sound are being investigated. It is hoped that one of these will agree well enough with experiment to justify its use as a standard scheme to replace currently used approximation methods.

For the thermal constants, the microscopic cross-sections may be averaged over a Maxwellian or a Wigner-Wilkins spectrum. Both of these methods, obtainable from the SOFOCATE code for the IBM-704 computer are being considered.

For the fast constants, the MUFT-IV code for the IBM-704 computer is being examined. Four approximations which are an intrinsic part of the MUFT code itself involve a choice between the B_1 and P_1 approximations; the Selengut-Goertzel approximation is also available.¹ All four possibilities are being applied to several water/metal lattices for which experimental data are available to determine the value of the second moment of the slowing down distribution at thermal energies given in the code. The results of these comparisons will determine the choice of procedure for calculation of fast group constants.

Another problem associated with the use of the MUFT-IV code is the choice of a source spectrum for the non-fuel region. In fuel regions the normal fission spectrum of neutron flux and current within the core should be considered. These spectra are obtained as the output of a MUFT calculation of fast group constants. The three spectra are shown in Figure-10 for a typical Yankee First Core configuration. The normalization is such that the area under each curve is unity. Theoretical considerations indicate that the most applicable spectrum for use in a calculation of reflector constants is probably the spectrum of neutron current within the core, whereas, for the non-multiplying regions within the core, the spectrum of core neutron flux should be used.

After the fast and thermal constants have been determined, the neutron flux predicted by diffusion theory may be calculated by means of the WANDA code, or transport theory may be applied through use of the SNG code. Both of these will be employed, using both 2 and 4 group theory in one dimension. The two dimensional diffusion theory program PDQ, will also be applied to the calculation of neutron flux distribution. The correlation of results obtained with diffusion theory and transport theory in one dimension will provide information required to evaluate the results obtained with PDQ.

Figure 10 Neutron Spectra in the Yankee First Core Obtained by Use of the MUFT IV Program
on the IBM-704 Computer

Fuel Depletion Analysis

A detailed study of the advantages and limitations of various digital computer codes for the calculation of reactor fuel depletion is in progress. The object of the study is the establishment of final specifications in order to revise the CAP-1 code which will include the calculation of thermal neutron cross sections from a Wigner-Wilkins spectrum; it will also incorporate an epithermal fission factor (β) as well as the fast fission factor (ϵ) in the modified one group equation describing fuel depletion, plutonium buildup and reactivity.

Results obtained in this study indicate that when all differences in basic nuclear data are eliminated between the CANDLE and MERLIN codes, very good agreement can be expected in the calculation of reactor lifetimes for uniform cores. A principal source of discrepancy which remains is due to epithermal fission in U-235. This discrepancy will be eliminated in the revised version of the CAP-1 code which can then be introduced into the MERLIN code. With regard to the organization of the revised CAP-1 code, it has been established that the calculation of changes in number density is relatively insensitive to the changes in thermal flux spectrum during life, so that frequent re-runs of the SOFOCATE code during the calculation of depletion will be unnecessary.

Loading Configurations

A preliminary study of loading configurations for the first Yankee core was completed. A comparison of uniformly loaded, two-region cycled, and two enrichment cores, all designed for a thermal output of 482 MW was made. The conclusion is that, ignoring the effects of control rod programming, the two enrichment core is superior from the standpoint of allowable power output. Although the preliminary control rod programming study performed on the uniform core indicates this type of core is subject to power flattening throughout life using control rods, the analytical procedures used in the study have not yet been experimentally verified. A study of rod programming in the two enrichment core must also be made before definite conclusions can be drawn.

2.3 Core Kinetic Analysis

H. C. Hecker

T. Gogniat

Kinetic Parameters

Calculations of various kinetic parameters of the Yankee First Core have been completed. A summary of the results is given in Table XI. The methods used and a more detailed description of results will be presented in a future Yankee topical report entitled: "Kinetic Parameters of the Yankee First Core". The methods employed have not yet been adequately checked against experimental values.

TABLE XI

Kinetic Parameters of the Yankee First Core

	<u>68°F</u>	<u>516°F</u>
Moderator Temp. Coefficient ($\frac{1}{k} \frac{\partial k}{\partial T_m} \times 10^4 / ^\circ F$)	+0.1 to -0.65	-0.8 to -2.6
Doppler Temp. Coefficient ($\frac{1}{k} \frac{\partial k}{\partial T_f} \times 10^4 / ^\circ F$)	-0.13 to -0.42	-0.15 to -0.50
Void Coefficient (% Δ % void)	--	-0.33
Neutron Lifetime (ℓ^* - sec.)	-1.5 - 2.4×10^{-5}	1.4 - 2.5×10^{-5}
Delayed Neutron Fraction (β_{eff})	0.0047 - 0.0070	0.0047 - 0.0070

The wide range of values given for the moderator temperature coefficient is due primarily to variations in the method of control and to computational uncertainty. At the reactor operating temperature, a small part of the spread is due to changes during core life. The uncertainty in the Doppler temperature coefficient is due to the spread in the available measured values of the logarithmic derivative of the U-238 resonance integral. A nominal value of the void coefficient has been presented. The variations in neutron lifetime and delayed fraction are expected to occur as the core changes during life, although the lower value for β is somewhat extreme.

The range in the moderator coefficient under various possible types of control is shown in Figure 11. The upper curve represents the maximum coefficient since boron poison-alone was assumed for purposes of control. The lower curve shows the most negative coefficient since a minimum boron concentration (control rods fully inserted) was assumed for control. Thus, these assumptions represent the extreme limits, and any other combination of control should fall within the curves shown.

Reactor Transient Analysis

The analysis of an uncontrolled rod withdrawal accident during reactor startup requires a representation of the average temperature of the fuel $\bar{T}_f$. The importance of $\bar{T}_f$ arises from its effect on reactivity through the

Figure 11 Possible Range of Yankee First Core Moderator Temperature Coefficient Due to Various Methods of Control

Dopper coefficient. When using the analog computer in the analysis, various models can be used to represent the transient behavior of $\bar{T}_f$. Three models were investigated to determine how each affected certain plant variables during an uncontrolled rod withdrawal accident. These models were:

- (1) Sectionalized fuel element.
- (2) First moment approximation.
- (3) Second moment approximation.

The procedure used in the computer analysis was as follows:

- (1) A somewhat simplified Yankee plant was represented.
- (2) $\bar{T}_f$ was derived by sectionalizing the fuel element into 5 radial regions of equal thickness.
- (3) The rods were withdrawn at some given ramp rate.
- (4) Recordings were made of plant variables such as heat generated, $\bar{T}_f$ and $\bar{T}_{ss}$ (average steel temperature).
- (5) The procedure was repeated, except that the first moment approximation model was substituted for the sectionalized fuel element model.
- (6) Similarly, the second moment approximation was substituted for the first moment approximation.

Results obtained from a typical run for the core power and the average fuel temperature are shown in Figures-12 and 13. The results indicate fairly good agreement between the second moment approximation and the sectionalized model with a clear indication of the discrepancies to be expected from the use of the first moment approximation. Experience with the sectionalized model and the second moment approximation have led to the conclusion that from the standpoints of convenience in use and economy of equipment, the sectionalized model is to be preferred. This model will provide the basis for development of a new analog computer representation of the Yankee plant for use in start-up transient analysis.

2.4 Control Rod and Chemical Poison Analysis

T. Williams

W. J. Eich

Control Rod Heat Generation

In an attempt to prevent excessive corrosion of the Yankee control rods (80% silver, 15% indium, and 5% cadmium), a suitable protective cladding may be desirable. In designing such a clad for these cruciform rods, it is necessary to know the maximum induced gamma heating in a typical rod. Assuming a hardened gamma spectrum over the core and an overall core ϕ peak/ ϕ avg. of 4 (not including local flux peaking), a maximum gamma heating of 46.2 watts/cc was calculated.

Figure 12 Transient Behavior of the Yankee First Core Power for Three Analog Computer Representations of the Fuel Rod Heat Transfer

Figure 13 Transient Behavior of the Average Fuel Temperature for Three Analog Computer Representations of the Fuel Rod Heat Transfer

Control Rod Programming

In a preliminary analysis of the reference control rod program for the Yankee reactor, the flux synthesis technique was used to obtain r-z power distributions. The accuracy of one flux distribution obtained by the synthesis technique is being checked by the two-dimensional PDQ code. This PDQ problem is also being used to determine whether an axial xenon tilt can be detected by the monitors in the neutron tank.

2.5 Critical Experiment Design and Analysis

R. Wolf
W. Eich

J. Jedruch

V. Grob

Power Distribution Comparisons

A power distribution has been calculated for a cylindrical loading of the Yankee 2.23:1 water/uranium metal ratio critical experiment core containing 3476 fuel rods and five aluminum alloy control rod followers. The power plot is shown in Figure-14 and includes the experimental points previously obtained. All points were normalized to an average power density of unity. The calculated power distribution was found using the PDQ code which utilizes 2-dimensional diffusion theory with two groups of neutrons to obtain neutron flux distribution and multiplication factors for a multi-region reactor core. Agreement between the calculated power distribution and the experimental points is good. The discrepancy at the edge of the core may be due to the necessity of homogenizing the fuel region in the PDQ code.

Measurements have been made at the Westinghouse Reactor Evaluation Center of flux shapes from a 2.23:1 critical assembly containing a central control rod. A calculation is being performed on the same configuration to check the simulation of control rods in a core region by a uniformly distributed poison. This work is required to evaluate the analytical methods used in the control rod programming analysis. More accurate values for the macroscopic poison cross section equivalent to a control rod will also result from the analysis of these data.

An additional experimental confirmation has been obtained for the assumption of separation of variables in the solution of the diffusion equation. A series of measurements of flux in the axial direction has been made at various radial distances from the axis of symmetry of the 2.23:1 water/uranium metal ratio CRX core, and the data have been fitted by means of the COFIT code. The following results have been obtained:

Figure 14 Comparison of Calculated and Experimental Power Distributions in the 2.23:1 W/U Ratio CRX Core

TABLE XII

Axial Buckling at Various Radii of the 2.23:1 W/U Ratio Yankee CRX Core

Distance from Center r, cm	Axial Buckling $B_H^2, \text{ cm}^{-2}$	Axial Reflector Savings (one side) $\lambda_H, \text{ cm}$
4.36	0.000543	6.44 ± 0.40
8.73	0.000541	6.58 ± 0.22
13.09	0.000545	6.34 ± 0.36
17.45	0.000549	6.08 ± 0.29
21.82	0.000534	7.02 ± 0.36

These results show agreement to within 3% and justify the assumption of separability.

Following the achievement of criticality with the 2.23:1 water-to-uranium ratio CRX core, a re-evaluation of the procedures for calculating the critical mass was initiated. The new procedure of calculation accounts for such additional effects as the epithermal fission in U-235, resonance absorptions in U-235 and stainless steel, and new flux disadvantage factors calculated from integral transport theory. The use of a U-238 resonance integral fitted to give the correct critical masses of the 3:1 (1862 rods) and 2.23:1 cores (3048 rods) led to a prediction of 1323 rods for the critical mass of the 4:1 core. This compares very well with the experimentally obtained value of 1314 rods.

Rod Worth Studies

Evaluation of the measurements performed on the 3:1 water/uranium metal ratio CRX core continued with the calculation of the total worth of a bank of nine control rods moved as a unit. The calculation was based on the measurement of flux shape in the axial direction in the core, where the flux was measured by irradiation of gold and copper foils. A part of the flux traverse corresponding to the upper region of the core, which contained the banked rods, has been fitted to obtain the best argument of an exponential function by means of the EXFIT code - a least squares curve fitting program for the IBM-704 computer. The axial buckling was

thus obtained for various core sizes and positions of the bank of nine control rods and was then used to calculate the total worth they would have if fully inserted. The results of these studies are as follows:

TABLE XIII

Total Control Rod Worth of the 3:1 W/U Ratio CRX Core
Nine Control Rods Moved in a Bank

<u>No. of Fuel Rods</u>	<u>Depth of Insertion</u>	<u>Total Rod Worth</u>
3244	80 cm	20.4 %
3244	80	19.0
3536	82	18.6
3536	82	19.7
2064	42	20.3

These results are reasonably consistent with each other, although somewhat higher than expected from measurements of single rod worths.

A study is being conducted of the experimental control rod worth data from the 3:1 W/U ratio Yankee critical core. The experimental data were obtained by fully inserting various control rod combinations and determining the amount of poison (boric acid) needed to make $k_{eff} = 1$. The rods were then withdrawn and the amount of additional poison needed to make $k_{eff} = 1$ was obtained. Thus, the boron equivalent of each rod or group of rods was obtained. The core parameters were found using the SOFOCATE code which determines the thermal constants averaged over a Wigner-Wilkins flux spectrum. The experimental rod worths were then found from the following equation:

$$\Delta k/k = \frac{f_1 - f_2}{f_1} + p = \frac{\left(\frac{\sum au}{\sum a}\right)^1 - \left(\frac{\sum au}{\sum a}\right)^2}{\left(\frac{\sum au}{\sum a}\right)^1}$$

where $\left(\frac{\Sigma_{au}}{\Sigma_a}\right)^1 = f_1$ = the fraction of all thermal neutrons that are absorbed in uranium with the control rod fully inserted

$\left(\frac{\Sigma_{au}}{\Sigma_a}\right)^2 = f_2$ = the fraction of all thermal neutrons that are absorbed in uranium with the control rod fully out

p = period correction, caused by failure to adjust boron concentrations to exact values required for criticality

The experimental rod worth $(\Delta k/k)_{exp}$ for the central control rod was then compared to a $(\Delta k/k)_{theory}$ for the central control rod, obtained by using perturbation theory. The theoretical value was 24% below that obtained by this experiment. This difference can be attributed to resonance absorption in the control rod, which the theoretical calculation does not take into consideration.

Temperature Coefficient Comparisons

Computations of the moderator temperature coefficient of the CRX core, at moderator temperatures below 80°C., were continued. Extensive work was performed on the 3:1 core, but agreement with experimental results has not yet been attained.

Data Reduction

A description of the new JOFIT code is given in topical report YAEC-86, "JOFIT - A Bessel J Least Squares Fitting Program for the IBM-704 Computer" issued in October 1958. The code is designed to fit the reduced results of foil irradiations in a neutron flux to the best argument of the Bessel J_0 function. Prior to final release, the JOFIT code was thoroughly tested and its performance found to be satisfactory.

2.6 Irradiation Experiment Design and Analysis

W. J. Eich

Piping near the purification cubicle of the MTR in-pile test loop is located close to radiation-sensitive instrumentation. It was determined that three inches of lead shielding around the pipe will reduce the level of radiation from fission products in the coolant to a safe level for the instrumentation.

Nuclear design data were calculated for the MTR process water irradiation experiments for a pellet having a 0.294 inch diameter within either a 15 or 21 mil stainless steel clad. Table XIV summarizes the calculations.

TABLE XIV

"Unperturbed" Fluxes (10^{13} n cm⁻² sec⁻¹) Required for Clad Surface
Heat Fluxes of 4.5×10^5 Btu/ft²-hr

<u>Enrichment (%)</u>	<u>15 Mil Clad</u>	<u>21 Mil Clad</u>
2.7	8.9	9.2
5.4	4.6	4.9
6.8	3.8	4.0

2.7 Shielding Analysis

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

2.8 Startup Experiment Assistance

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

3.0 CHEMISTRY

Chemistry and Ceramics Section:

R. F. Sterling, Manager

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

3.1 Properties and Removal of Chemical Neutron Absorbers

N. Michael

Ion Exchange of Radioisotopes

Mixed Fission Products

Decontamination factors were reported in YAEC-87, Quarterly Progress Report for the April to June, 1958 period for three candidate ion exchange resins - Rohm & Haas XE-150, -154 and -170 - using a synthetically prepared solution of mixed corrosion products. In order to learn the degree of selectivity in pick-up of the different elements, experiments were performed with separate corrosion elements - chromium, cobalt, and iron. Since fresh Oak Ridge National Laboratory radioisotopes were used in these experiments, re-runs were made with mixed corrosion products for more accurate comparison of decontamination factors. Each of the three resins was tested separately with four solutions: 1) iron 0.06 ppm, 2) chromium 0.0007 ppm, 3) cobalt 0.012 ppm, and 4) corrosion product mixture (i.e., a mixture of Fe, Cr, and Co in above concentrations). In testing XE-150 and XE-170 resins, solution pH was adjusted to 8.5 with NH_4OH ; in the case of XE-154, solution pH was adjusted to 8.5 with LiOH . A 1000 ml solution sample at 130°F. was passed through 100 ml of resin in each test. The mean decontamination factors of the three resins are compared below:

Decontamination Factors

<u>Resin</u>	<u>Iron</u>	<u>Chromium</u>	<u>Cobalt</u>	<u>Mixed Corrosion Products</u>
XE-150	273	180	4000	1157
XE-154	264	14	155	64
XE-170	86	44	138	95

The values given above for XE-150 and XE-170 tested with mixed corrosion products are somewhat higher than those previously reported in Quarterly Progress Report YAEC-87 - 335 and 7.7, respectively. The reason for this difference is twofold:

1. The material tested is from a different isotope shipment and the chemical properties of the two possibly differ, e. g., oxidation states and colloidal form may not be alike.
2. Column throughput in these tests was 1000 ml compared to 1500 ml previously.

Boric Acid Removal Studies

Studies have been conducted to compare the capacity of several candidate ion exchange resins for use in removing boric acid from the Yankee primary coolant system during reactor startup. The experiments were concluded, and Rohm and Haas XE-78 anion resin has been selected as the reference resin for this application. A 40 ppm boron solution and Nalcite SBR and Rohm and Haas XE-78 anion resin were used. Before addition of the acid, water pH was adjusted to 10 with LiOH. Flow rate during the experiment was 25 ml/min (1.9 GPM/ft³) at a temperature of 130°F. Spent resin was eluted with NaOH. The eluent was analyzed to determine total boric acid capacity of each resin. The XE-78 has a capacity, under the conditions outlined, of 3.9 lbs/ft³ and the SBR a capacity of 3.3 lbs/ft³. Capacity results reported in the YAEC-87 Quarterly Progress Report should be revised as a result of corrections submitted by the (W) APD Analytical Laboratory. The conditions of the test were as follows: Boric acid was added to LiOH solution to give a total boron concentration of 94.5 ppm boron. The revised boric acid capacity of XE-78 with this test solution is 5.6 lbs/ft³, with SBR it is 3.7 lbs/ft³.

Pressure Drop and Filtration Studies

Hydraulic characteristics of XE 150 and "Neva-Clog" filter medium - the respective mixed-bed reference resin and under-drain filter for the Yankee Reactor - were investigated. Both filter and resin data showed a straight-line relationship for pressure drop vs. flow rate on logarithm coordinates. Pressure drop for the "Neva-Clog" increased from 0.2 to 4 psi on increasing the flow rate from 2 to 8 GPM/ft² - area. The pressure drop for the resin over the same flow range is 0.05 to 0.3 psi per foot of bed depth. During the study of hydraulic characteristics, approximately 13 gallons of demineralized water was sent through the column. The effluent was collected and filtered with Whatman No. 40 paper. The resin particles on the filter were counted and the size of the largest particle measured. A total of 150 particles was found with sizes up to a maximum of 50 microns. The preceding results are well within the guaranteed retentivity of the "Neva-Clog" filter. Information obtained from the use of this type filter in other reactors indicates that it is a most reliable resin bed support medium.

3.3 Corrosion of Materials of Construction

D. D. Whyte

A. Krieg

Corrosion of AISI 304 Stainless Steel in High Temperature Borated Water

Revaluation of the results of corrosion tests of AISI 304 stainless steel in high temperature borated water is in progress. Specimens originally electrolytically descaled have been descaled chemically using the basic permanganate-ammonium citrate procedure. The results of Test No. MEDA-1 (Boron - as boric acid = 39 ppm, pH = 9.8 with lithium hydroxide) are complete and are shown graphically in Figure-15. No significant change from the previous results obtained with electrolytic descaling has been observed. The chemical descaling treatment was particularly effective in removing scale easily and completely. This treatment produced some attack on the base metal (1.1 mg/dm² in 5 minutes). The results shown in Figure-15 are corrected for base metal attack. Preliminary indications are that the residual scale (after electrolytic descaling) was more difficult to remove chemically in all of the remaining tests (MEDA-2, MEDB-1 and MEDB-2). The descaled weight changes after chemical descaling are substantially higher for the three remaining tests than those previously obtained by electrolytic descaling. The final draft of Topical Report YAEC-68 covering the revaluation of the four tests in the (W) Materials Engineering Department test loops is being prepared.

Dynamic Screening Tests of Yankee Primary System Materials

Screening tests of proposed Yankee Primary System materials have been continued under high flow conditions at various levels of boron concentration in the presence of lithium hydroxide. The results of tests performed in the absence of lithium hydroxide are shown in Table XV for comparison with the results of the pH adjusted tests shown in Table XVI. The effect of pH adjustment has been to decrease the extent of gross corrosion of AISI 304 stainless steel, AISI 410 stainless steel and Inconel. The effect on Ag-In-Cd corrosion is at present uncertain. The gross corrosion of Zircaloy-2 appears to be independent of pH adjustment.

(W) APD loop "A" Test No. 7 (Table XVI) was performed under conditions which were intended to investigate the feasibility of a chemical stability study in the WCAP-4 in-pile loop test at MTR. The detailed water chemistry data for Test No. 7 are summarized in Table XVII. The results indicate that the analytical procedures used are sufficiently precise to account for boron concentration changes as a function of dilution by make-up water addition. No evidence for boron "hide-out" within the accuracy of the analytical procedure was detected. Lithium concentrations were more difficult to follow at the low concentration range used. In general, the feasibility of a chemical stability in-pile experiment was indicated.

FIGURE 15- WEIGHT CHANGES OF 304 STAINLESS STEEL IN WATER AS A FUNCTION OF EXPOSURE TIME (TEST MEDA-1)

TABLE XV

Results of Screening Corrosion Test Numbers 3, 4 and 5 of Yankee Primary System Materials for One Month in High Temperature Water

Test No.	3	4	5			
Test Conditions						
Temp., °F	598	600	600			
Pressure, psig	1792	1823	1831			
Flow, fps	39.4	38.7	37.7			
Duration, hr.	715	759	707			
H ₂ , ml(STP)/kg.	27	25-35	31.2			
B (as boric acid),ppm	1590	cycled (1600 to <5)⊗ 2 cycles	3			
pH	5.0-5.2	cycled ⊗*	6.1			
Specimen Results	No. of Specimens	Avg.Wt.Change, mg/dm ² Exposed* Descaled**	No. of Specimens	Avg.Wt.Change, mg/dm ² Exposed* Descaled**	No. of Specimens	Avg.Wt.Change, mg/dm ² Exposed* Descaled**
<u>AISI 304 s/s</u>						
a. quenched	2	-21.4 -46.7	1	-13.1 -33.3	1	-16.9 -46.7
b. heat sensitized	3	-17.3 -34.3	2	-12.5 -33.3	2	-20.1 -52.3
c. as-received	0	- -	2	-13.7 -32.7	1	-21.7 -62.2
d. polished	0	- -	2	-11.3 -33.6	2	-13.8 -42.2
e. weld coated, 50%	1	-14.9 -40.2	1	-10.1 -27.0	1	-14.9 -40.2
f. with Microbrazed-50	1	-31.5 -55.9	0	- -	1	-14.9 -
Ag-In-Cd	3	+56.5 -	3	+22.6 -	3	+154.5 -
410 s/s	2	-65.5 -121.4	2	-75.0 -142.7	2	-31.8 -84.6
Inconel	2	-77.4 -140.9	2	-41.1 -80.9	2	-13.4 -42.6
Zr-2	2	+16.7 +15.2	2	+13.7 -	2	+24.2 -
Hafnium	2	+ 4.2 + 2.4	0	- -	0	- -

* Weight after exposure minus original weight.

** Original weight minus weight after chemical descaling (corrected for attack of descaling solution on base metal).

⊗ High boron- average ppm boron= 1630; low boron- average ppm boron= <5

⊗* High boron- average pH = 5.7; low boron- average pH = 6.0

TABLE XVI

Results of Screening Corrosion Test Numbers 6, 7 and 8 of Yankee Primary System Materials for One Month in High Temperature Water

Test No.	6	7	8
Test Conditions			
Temp., °F	600	600	591
Pressure, psig	1828	1816	1808
Flow, fps	38.8	38.6	38.6
Duration, hr.	714	712	691
H ₂ , ml (STP)/kg	27.1	34.9	30.5
B (as boric acid), ppm	1561	189	Nominal 1600 to <5 (2 cycles)*
pH (Li, ppm)	5.9 - 6.3 (0.98)	7.4 (2.0)	2 cycles**
O ₂ , ppm	<0.1	<0.1	0.3

Specimen Results	No. of Specimens	Avg.Wt.Change, mg/dm ²		No. of Specimens	Avg.Wt.Change, mg/dm ²		No. of Specimens	Avg.Wt.Change, mg/dm ²	
		Exposed*	Descalcd**		Exposed*	Descalcd**		Exposed*	Descalcd**
<u>AISI 304 s/s</u>									
a. quenched	1	-10.4	-23.6	1	-6.0	-22.5	1	-22.4	-35.5
b. heat senstitized	2	-8.4	-23.4	2	-6.6	-25.3	2	-10.3	-24.6
c. polished	1	-7.1	-22.7	1	-6.0	-19.5	1	-10.6	-23.8
d. weld coated, 50%	1	-3.9	-26.0	1	-7.6	-27.1	1	-10.0	-22.5
e. with Nicrobraz-50	1	-13.4	-	1	-6.6	-	1	-11.8	-
<u>Ag-In-Cd</u>	3	+70.8	-	3	+47.0	-	2	-20.6	-
<u>AISI 410 s/s</u>	2	-33.4	-62.5	2	-36.2	-104	2	-26.1	-72.8
<u>Inconel</u>	2	-27.2	-70.5	2	+8.7	-18.5	2	-15.9	-34.6
<u>Inconel-X</u>	2	-24.5	-71.4	2	+5.7	-22.4	2	-16.5	-38.7
<u>Zr-2</u>	2	+19.1	-	2	+22.2	-	2	+16.0	-

* Weight after exposure minus original weight.

** Original weight minus weight after chemical descaling (corrected for attack of descaling solution on base metal).

⊛ High boron- average ppm boron= 1790; low boron- average ppm boron= 11.6.

⊞ High boron- average pH =6.1 (lithium = 1.28 ppm); low boron - average pH = 8.6 (lithium = 0.8 ppm).

TABLE XVII

MTR "WATER CHEMISTRY STABILITY" EXPERIMENT SIMULATION

PERIOD NO.	TIME, HR.	AVER. CONC.* (DAILY SAMPLES)		% DECREASE FROM INITIAL CONC.**		TOTAL WATER LOST	WATER, ML (Rm. Temp.)		WATER ADDED	LEAKAGE RATE, ml/hr.	% DILUTION (TOTAL)
		B, ppm	Li, ppm	B	Li		WATER RE- MOVED FOR ANALYSIS	LEAKAGE			
1	97	216	2.88	2.3	3.0	2231	1672	559	0	5.8	0
2	99	206	2.39	6.8	19.6	2400	1437	963	1847	9.7	4.1
3	170	193	1.98	12.7	33.4	2592	1490	1102	2832	6.5	10.4
4	142	186	1.80	15.8	39.4	2448	1889	559	2256	3.9	15.6
5	119	176	1.95	20.4	34.4	2880	1600	1280	2448	10.1	20.8
6	84	158	1.64	28.5	44.9	1200	513	687	3660	8.2	29.1
over-all											
= 7.2											

Test Conditions:

T = 600°F
 P = 1816 psig
 Flow = 33.6 feet per second
 Duration = 712 hours
 H₂, ml (STP) per kg. = 34.9
 B (as boric acid), ppm = 189
 Li as (LiOH), ppm = 2.0
 pH = 7.4
 O₂, ppm = 0.03
 Cl⁻, ppm = 0.11

* Up to but not including water addition at end of period.

** Initial conc.: B = 221 ppm; Li = 2.97 ppm.

pH Control Agent Selection

Corrosion studies of Yankee primary system materials undertaken in static autoclaves to aid in the determination of a pH control agent have been completed. Two tests were conducted simultaneously under the following conditions:

Flow - static

Pressure - 1525 psig

Temperature - 600°F.

Water Chemistry - 1600 ppm B (added as H_3BO_3)

pH - adjusted to 10 ± 0.5 before addition of H_3BO_3
(controlled with KOH or LiOH) 30 cc $H_2/Kg H_2O$ (STP)

$O_2 = < 0.1$ ppm

Conclusions from the corrosion studies are as follows:

1. The corrosion rate of 80Ag-15In-5Cd and Microbraz-50 is much greater in water containing KOH than in water containing LiOH.
2. The corrosion of all other materials was slight and little difference was observed for corrosion in water containing either LiOH or KOH.

Dynamic Tests of 80/Ag-15/In-5/Cd Alloy in Boric Acid Solution

The effect of boric acid on the corrosion behavior of unplated 80/Ag-15/In-5/Cd alloy was investigated at 600°F. under semi-dynamic conditions (0.1 fps) in water containing H_3BO_3 , and either LiOH or KOH. The results are that the presence of 9145 ppm H_3BO_3 (1600 ppm B) in the LiOH solution decreased the weight gain of Ag-In-Cd from 107 mg/dm²/mo (without H_3BO_3) to 34 mg/dm²/mo. The presence of 9145 ppm H_3BO_3 (1600 ppm B) in the KOH solution decreased the corrosion rate of Ag-In-Cd from 76 mg/dm²/mo (without H_3BO_3) to 21 mg/dm²/mo.

Simulated Reactor Startup and Shutdown Tests

Two dynamic autoclave corrosion tests are being performed to determine the effect of concentration cycling of boric acid on corrosion of Yankee primary system materials. One autoclave contains 1 ppm of Li as LiOH and the other contains no lithium.

Flow - 0.1 fps
 Temperature - 600 °F
 Pressure - 1525 psig
 Water Chemistry - pH - test 1 - 5.8 (controlled with 3.4 ppm LiOH)
 pH - test 2 - 4.8
 30 cc H₂/Kg H₂O (STP); O₂ = < 0.1 ppm
 Boron - 1600 ppm to 5 ppm cycled up and down every 14 days.

Reaction of Oxygen with Hydrazine in Borated Water

<u>Time (min.)</u>	<u>Temp. (°F.)</u>	<u>O₂ (ppm)</u>	<u>N₂H₄ (ppm)</u>	<u>pH</u>
0	71	7.0	0	4.8
N ₂ H ₄ added	--	--	7.0	5.9
30	122	3.0	3.6	5.6
45	173	1.2	1.9	5.4
60	211	1.0	1.6	5.4
75	254	< 0.02	1.5	5.4
90	303	< 0.02	1.4	5.4
105	351	--	1.3	5.4
120	402	--	1.2	5.4
135	424	--	1.1	5.4

TABLE XVIII

SIMULATED REACTOR STARTUP AND SHUTDOWN TEST RESULTS

<u>Spec. Comp.</u>	<u>Exposed Weight Change (mg/dm²)</u>				<u>pH Control</u>
	<u>1 day</u>	<u>3 day</u>	<u>7 day</u>	<u>14 day</u>	
Microbraz-50	-2.0	-1.6	-0.6		LiOH
	-1.8	-3.3	-0.9	-3.6	none
AISI 304 S/S Sensitized	-0.3	-0.2	-1.4		LiOH
	-2.7	-4.8	-1.4	-2.9	none
Zircaloy-2	6.1	6.9	8.0		LiOH
	9.9	10.1	12.7	14.3	none
Inconel	-0.2	-0.2	0.3		LiOH
	-1.2	-2.3	-0.8	-2.0	none
Ag-In-Cd	3.7	16.1	35.4		LiOH
	7.8	12.9	25.7	36.1	none
AISI 304 S/S Quenched	0.7	2.5	2.7		LiOH
AISI 410 Stainless Steel	-4.0	-4.1	-4.2		LiOH
	-8.7	-11.0	-12.3	-16.8	none
AISI 304 S/S As-received	5.5	6.0	7.5		LiOH
	-1.7	-3.4	-1.5	-2.0	none
Zircaloy-2 vs. Ag-In-Cd	4.8	12.9	24.7		LiOH
	8.4	12.3	18.5	24.5	none
Zircaloy-2 vs AISI 304 S/S Quenched	6.6	13.1	10.3		LiOH
	-7.4	-5.6	-3.6	-5.6	none
AISI 304 S/S Quenched vs. Ag-In-Cd	8.2	12.7	18.7		LiOH
	5.2	8.8	12.3	20.7	none

The results indicate that hydrazine is effective in removing oxygen at a temperature below 254° F. The further reduction in hydrazine content above this temperature was caused by the effect of elevated temperature on the hydrazine. The fact that there was not an equimolar reaction between hydrazine and oxygen is indicative of oxygen removal by autoclave materials. The next series of tests will determine the time required for hydrazine to remove essentially all oxygen from the water under the same conditions as the preceding test except that constant reaction temperatures of approximately 200° F and 250° F will be maintained.

Silver - Indium - Cadmium Studies - Crud Transport Problems

Silver-indium-cadmium crud transport problems were evaluated by WAPD Reactor Engineering Department personnel with the following results:

1. Forty mg/dm²/mo of crud from corrosion of the control rods would not present shielding difficulties in the Yankee reactor. Four hundred plus mg/dm²/mo corrosion would begin to present problems.
2. Dissolution of unplated Ag-In-Cd samples in the in-pile test loop (100% dissolution of all specimens postulated for calculation purposes to occur at 10 months exposure) would present no shielding problems.

3.4 Interactions Between Chemical Absorber, Corrosion Products and Fission Products

R. F. Sterling

C. C. Thomas

The study of the effects of Van de Graaff electron beam irradiation on crud deposition, at both neutral and elevated pH was concluded and the data were analyzed. The final draft of topical report YAE-93, "Corrosion Product Transport and Deposition Under Ionizing Radiation" covering the investigation is being prepared. Conclusions which were drawn include the following:

1. The only deposits attributable to radiation-induced deposition were found on the directly irradiated surfaces (Zircaloy-2 and AISI 304 stainless steel) rather than on the specimens.
2. No significant difference was noted in the amount of iron deposited under radiation at pH 7 and at pH 10 on AISI 304 stainless steel.
3. The deposition of iron on Zircaloy-2 at pH 10 is greater than on AISI 304 stainless steel at pH 10.
4. In future studies, the use of test specimens that do not contain elements in common with the crud would facilitate the evaluation of the results obtained in the experiments.

5. The size of the crud particles appear to increase with pH.
6. Comparison of these results with previous experimenter's data shows considerable difference in the apparent crud release rates found at pH 7 and in the location of the crud deposits at either pH. The need for additional crud deposition studies is indicated to determine the cause of such differences.

3.5 Decontamination and Waste Disposal Studies

R M. Watkins

The best method of measuring the effectiveness of a proposed decontamination procedure is to actually determine its ability to remove a deposited radioactive oxide film from a sample of material similar to that from which the plant system is to be fabricated. In an effort to do this, contaminated specimens of austenitic stainless steel tubing (3/8 inch in diameter) from the original SLW steam generator have been obtained. The necessary equipment for handling the radioactive pipe specimens was procured, and after receiving (W) APD Health Physics and Isotope Committee approval for the proposed tests, experimentation was begun.

The basic permanganate-citrate cleanup procedure which appears to be most promising for Yankee reactor application was tested first. The experimental results obtained were most encouraging as D. F. values in excess of 1500 (99.93% activity removal) were achieved in static tests. A series of cleanup trials indicated that the temperature of the basic permanganate reagent is a significant variable in determining the success of the decontamination operation and that the exposure time to the ammonium citrate solution should be increased beyond the one hour period originally recommended for optimum decontamination efficiency. The results of an exposure time dependence study show that the D. F. values rise linearly with time during the first few hours of contact with the ammonium citrate reagent. (See Figure-16)

Metallographic examination was performed on several specimens of the SLW steam generator tubing after treatment with the decontaminants under study. No significant damage to the base metal was found. It was noted, however, that in the treatment of austenitic stainless steel, low pH complexing agents such as oxalic acid are more corrosive than ammonium citrate. The first signs of mild intergranular attack were noted in the examination of tubing specimens treated at 100 C. for two hours with a 10% oxalic acid solution (after the standard preliminary treatment with the basic permanganate reagent as described in Quarterly Progress Report YAE-52). No such attack was evident in the case of a similar treatment with the 10% ammonium citrate solution suggested for application in Yankee reactor decontamination (See Figures-17a and 17b).

Figure 16 Relation of Exposure Time in Ammonium Citrate Solution to Decontamination Factor Achieved in Basic Permanganate Citrate Decontamination

Figure 17a Photomicrograph (1000X) of Edge of Original
SLW Steam Generator Tubing Section After
Basic Permanganate Decontamination
Followed by 10% Oxalic Acid Treatment

Figure 17b Photomicrograph (1000X) of Edge of Original
SLW Steam Generator Tubing Section After
Basic Permanganate Decontamination Followed
By 10% Ammonium Citrate Treatment

Information was obtained on the decontamination procedure for stainless steel reactor components developed by Turco Products Inc. of Los Angeles, California at a meeting held with Turco Products personnel at (W) APD. The Turco "4501 Process" for radioactive decontamination was discussed in detail and samples of the reagents used were provided for use in decontamination studies at (W) APD.

The Turco process involves six steps and makes use of three separate reagents in contrast to the four steps and two reagents employed in the basic permanganate-citrate procedure described in topical report YAEC-90, "A Study of Decontamination Agents for Use in the Yankee Reactor" to be issued in December, 1958. Basically, the Turco "4501 Process" consists of the following:

1. Immersion in Turco 4501 (an amber colored highly alkaline liquid containing potassium hydroxide, potassium salts of organic acids, aliphatic amines, and a small amount of phenolic material).
2. High pressure water rinse.
3. Immersion in Turco 4502 (a highly alkaline solution containing up to 3.8 pounds per gallon solids. These include potassium permanganate, alkaline potassium salts, small amounts of fluoride salts and inhibitors).
4. High pressure water rinse.
5. Immersion in Turco 4512 (a highly inhibited phosphoric acid containing a detergent).
6. High pressure water rinse.

This procedure was applied in the cleanup of contaminated specimens cut from original SLW steam generator tubing. For comparison purposes specimens from the same tube section were decontaminated by use of the basic permanganate-citrate procedure. The experimental conditions and D. F. values obtained in these tests are given in Table XIX.

TABLE XIX

SLW Steam Generator Tubing Decontamination Test Results

<u>Reagent Designation</u>	<u>Time (min)</u>	<u>Temperature °C</u>	<u>Total D. F.</u>
Turco 4501	90	133	1.07
Turco 4502	90	106	2.94
Turco 4512	30	71	222
Basic Permanganate	90	110	3.85
Ammonium Citrate	120	100	>1500

It appears that the basic permanganate citrate procedure is not only more simple in its application, but is also capable of producing substantially better decontamination results than those which can be achieved by the Turco 4501 process.

3.6 Crud Inhibition, Suspension, and Removal

D. G. Sammarone

Calculations were made to determine the feasibility of running crud deposition experiments in (W) APD Loop "A". These experiments were intended to determine whether crud would preferentially deposit on cooled surfaces having a heat flux of 30,000 - 50,000 BTU/Hr-ft² under high (10.5) or neutral (6.5) pH conditions to simulate conditions in the Yankee steam generator.

It was recommended that experiments should not be run because:

1. Carbon steel wool would have to be used to produce detectable amounts of crud. This crud would not be a close enough simulation to crud found in the steam generator.
2. It would require a "hot" facility to be able to run experiments which relate activity build up with crud deposition. Such facilities are not available at (W) APD.

3. The size of the equipment required to simulate heat flux would be bulky and could not be conveniently accommodated in the present (W) APD Loop "A" area.
4. A considerable amount of uncertainty exists as to whether crud build-up is a function of the heat flux that exists across the surface on which the deposition occurs.

It was therefore decided not to assemble and install the equipment considered for this experiment on (W) APD Loop "A", but rather to find out whether preferential crud deposition exists at all on cooled surfaces. This will be performed by sectioning portions of the Van de Graaff crud deposition loop from (a) the heat exchanger and (b) thermally insulated piping immediately adjacent to the heat exchanger, and using the alkaline permanganate-citrate decontamination procedure which has proven so effective on stainless steel in removing crud and corrosion products. It will be then determined whether any significant difference in total crud and corrosion product exists between the two situations.

4.0 MECHANICAL DESIGN

Mechanical and Thermal Section:

A. G. Thorp, Manager

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

4.1 Fuel Assemblies and Control Rod Design

L. Balog

C. Johnson

Control Rod Design

Tests completed on the spring-loaded control rod joint indicated satisfactory performance. The joint design was incorporated in the coupling section between the control rod and follower for the channel mock-up for fuel handling tests. The design of the coupling section and the socket for adapting the control rod shaft were proportioned for adaption to the new Universal Handling Tool, and the test model was altered accordingly. The control rod is now complete and ready for test in the Yankee channel mock-up program.

A fuel bearing control rod follower consisting of fuel rods containing UO_2 pellets was designed for use in a comparison study with a solid Zircaloy-2 follower. The fuel-containing follower would produce a substantially lower flux peak in the fuel adjacent to the follower in the reactor core. An analysis of the mechanical design of the fuel rod follower indicated that the mechanical performance should be more than adequate providing that fabrication techniques can be developed for maintaining accuracy of construction.

Fuel Assembly Design

Analysis of the problem of fuel assembly bowing was continued and included an investigation of a number of design alternatives. The studies resulted in the tentative selection of a "Compromise Design" for use in the Yankee reactor. The "Compromise Design" is based upon the use of the brazed ferrule fuel subassembly design concept, but incorporates a series of small changes in the design to increase clearances adjacent to the control rods to reduce thermal bowing of the complete assembly. A cross section of the "Compromise Design" is shown in Figure-18. The design consists of eight subassemblies which are free to float axially and which surround a ninth rigid center subassembly fixed rigidly to a new type of handling socket. The eight outer assemblies are guided on each end in the handling sockets and are tied together at various axial stations by thin straps in such a way that all subassemblies must bow the same amount.

Figure 18 Yankee Fuel Assembly - Compromise Design

The result of this arrangement is that the subassemblies which do not experience the high thermal gradients will act to restrain the remaining subassemblies and thus minimize the thermal bowing. A comparison of the results of the thermal bowing studies and the significant design dimensions are given in Table XX.

The analytical study to determine the ability of the fuel tubes to withstand a cold hydrostatic test at 3750 psi was continued. Analysis indicated that the fuel cladding will withstand these test conditions satisfactorily and without collapsing. A series of tests on stretch-formed tubes will be conducted to confirm the analysis.

TABLE XX

Fuel Assembly Design Data and Results of Thermal Bowing Studies

<u>Design Parameters</u>	<u>Definitive Design</u>	<u>Compromise Design</u>	
		<u>First Iteration</u>	<u>Second Iteration</u>
Subassembly Size	9 x 9	6 x 6	6 x 6
Fuel Rod Pitch	0.425 in.	0.418 in.	0.422 in.
Control Rod Thickness	0.285	0.245	0.265
Nominal Clearance	0.119 (Strap)	0.165 (Strap to Control Rod)	0.123 (Strap to Control Rod)
Guide Clearance	0.062	0.095	0.095
Fabrication Bowing	0.060	0.060	0.060
Socket Tolerance	0.020	0.020	0.020
Thermal Bowing	0.146	0.095	0.106
Control Rod Bowing	0.015	0.015	0.015
Additional Tolerance	0.017	0.017	0.017
Net Clearance	-0.077	+0.053	0.000

4.2 Control Rod Drive Mechanism

L. Balog

R. Goldstein

A. Bishop

Progress at the (W) Atomic Equipment Department on the fabrication of the positive engagement type control rod drive mechanism indicates that the mechanism will be completed in December. A cut away drawing of the mechanism is shown in the illustration on Page 10.

The design and construction of a test stand for the control rod drive mechanism prototype was continued. The test stand will serve as a facility for cold tests where loads can be readily varied and maintenance and inspection of the mechanism can be performed with a minimum of effort. After completion of the cold tests and any necessary development modifications, pressure and temperature tests will be conducted with the mechanism mounted on an autoclave.

A DC power supply was designed for supplying power to the control rod drive mechanism. Quotations were requested for the detailed design and construction of the power supply, and an order was placed with a supplier.

Analysis of the cooling requirements for the control rod drive mechanisms was completed. Experimental data obtained for the friction grip type mechanism were utilized as a basis for computing the cooling requirements for the positive engagement type control rod drive mechanism. Under operating conditions the analysis indicated a heat load for cooling of 1.3 KW per mechanism.

4.3 Design of Core Support Structure and Fuel Handling Tools

L. Baloy

G. Eng

Core Support Structure

Topical report YAEC-77, "Deflection and Stress Analysis of the Yankee Core Support Structure" was issued in October, 1958. The report describes the experimental investigation on plastic models to determine the various design parameters and the structural rigidity of the assembly. The report suggests a rational method of semi-theoretical analysis of the problem based on Reissner's small deflection theory. This takes into account the effect of transverse shear and normal stress deformation.

Core Baffle

The initial stress and deflection studies on the one-quarter scale core baffle model were conducted over a wide range of pressures. By means of a dimensional analysis study it was found that the deflections at several points in the baffle were somewhat greater than desired. The baffle model was returned to the (W) APD Model Shop where additional

ribs were welded to the baffle structure. By the addition of these added supports, the stiffness of the plates which had undergone the largest deflections was increased substantially. After these changes were incorporated, deflections were again measured. Calculations will now be performed to determine the extent of deflections in a full size baffle. From analyses conducted to date, it appears that the largest deflection in the full size baffle will not exceed 0.045 inch.

Fuel Handling Tool

The basic design of a hydraulically actuated Universal Handling Tool, including the hydraulic system, was completed. The tool is capable of handling the control rod, control rod follower, control rod drive shaft, fuel assembly and guide tube. The total weight of the tool is approximately 305 pounds, and its overall length is 72 inches. The maximum cross sectional dimension of the tool, taken across a set of hydraulic cylinders, is approximately 11 inches. The tool was designed for a static load of 6000 pounds and to operate at a hydraulic pressure of 100 psi.

Yankee Channel Mock-Up

The (W) APD Model Shop continued work on the Yankee Channel Mock-Up. At present, the design changes made necessary by the Universal Handling Tool are being incorporated into the handling sockets of the Guide Tube Adaptor and the Fuel Assembly Nozzles. The majority of the large tower components, including the Drive Shaft Assembly and Control Rod Stand, were completed.

4.4 Design for Critical Experiment and Irradiation Tests

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

5.0 THERMAL AND HYDRAULIC DESIGN

Thermal and Hydraulic Design Section:

A. G. Thorp, Manager

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

5.1 Thermal Design

C. Johnson

A. Bournia

E. McCabe

A. Petrovas

Fuel Temperature Study

The maximum temperature at the center of the fuel, the average temperature of the hottest pellet, and the average temperature of the fuel in the core were calculated for various thermal outputs of the Yankee first core, based on a UO_2 conductivity of $1.0 \text{ BTU/hr-ft-}^\circ\text{F}$. In this study the 0.294 inch diameter pellet was considered. The average temperature of the coolant was kept constant at 514°F . Also, in this study, it was assumed that the gap, between pellet and tube, is filled with 85% xenon and 15% krypton. It was further assumed, whenever the pellet expanded more than the tube, that the minimum effective gap was 0.0002 inch. An average constant coefficient of expansion for UO_2 of $6.2 \times 10^{-6} \text{ in/in - }^\circ\text{F}$. was used in the study. The results are shown in Figure 19.

Loss of Flow Accident

Programming of the loss of flow accident for the IBM-704 computer was begun and is now being checked. A problem model is being used that is similar to the one used in the hand calculations described in topical report YAE-72, "Studies of Thermal Behavior Under Loss of Pump Power Transient Conditions", i.e. the pressure drop of the average channel of infinite radial extent is compared to the hot channel. Most of the program development has been completed.

In-Pile Test Loop Loss of Flow Accidents

A hazards analysis was made on loss of flow accidents in the in-pile test loop. Using flow rates that have been obtained experimentally from the actual loop, steady state analyses were performed to determine the final conditions for two pump operation and single pump operation for an inlet temperature assumed to be constant. An analysis was also made for the transient condition during the coastdown of two pumps, with or without

Figure 19 Power Output Versus Fuel Temperature for the Yankee First Core

scramming, the coastdown from two to one pump operation without scramming, and the coastdown of one pump without scramming. For the transient calculations, the limiting condition was assumed to be loop outlet temperature reaching saturation temperature at 2000 psia. Thermal time constants are such that no detrimental condition is anticipated for any of the accidents. A topical report is being written on this study.

Internal Pressure of Fuel Rod

The internal pressure of the Yankee first core hot channel fuel rod was calculated based upon theoretical fission gas release rates. The calculations were for the end of core life, i.e. 10,000 hours of operation, and considered the initial radial and axial void as well as the radial elastic deformation of the clad. Using these pressures, the tangential stress on the inside surface of the clad was calculated to be 7,000 psi for the non-compartmented and 27,800 psi for the compartmented fuel rod.

In the study of the compartmented rods, fuel rods having a disc spacing of 9 inches and 8 inches were considered. For the 9 inch compartment, the axial gap was assumed to be 30 mils in the hot condition. In the 8 inch compartment the hot gap was assumed to be 58 mils. The dimensions used are as follows:

	<u>Non-Compartment Design</u>	<u>Compartment Design</u>	
		<u>Nine Inch Spacing</u>	<u>Eight Inch Spacing</u>
Pellet Diameter, inch	0.294	0.294	0.294
Outside tube diameter, inch	0.337	0.337	0.337
Clad thickness, mils	21.0	21.0	21.0
Tube length, inches	91.375	--	--
Pellet length, total, inches	90.0	--	--
Compartment length, inches	--	9.14	7.956
Pellet length per compartment, inches	--	9.0	7.8

The pressures and stresses calculated for the above cases are:

	<u>Non-Compartment Design</u>	<u>Compartment Design</u>	
		<u>Nine Inch Spacing</u>	<u>Eight Inch Spacing</u>
Internal pressure, psi	3,180	7,780	5,890
Tangential stress, psi	7,000	42,300	27,800

If the assumption and procedure used in this analysis are valid then the design of a compartmented fuel rod requires some modification. By increasing the initial void, the final pressures will be low and thus create tangential stresses below the elastic limit of stainless steel, approximately 17,000 psi for AISI 304 stainless steel at 600°F. The non-compartmented fuel rod has sufficient safety in the design that no additional modification is necessary.

Power Ratio Variation

A study was conducted to determine the axial position in the hot channel which will have the highest "power ratio" over the core life. The power ratio is defined as the instantaneous power density divided by the overall average power density. Information obtained from a study of control rod insertion in the first core was used as the basis for the power density distributions during core life. Axial power density distributions were calculated for various control rod positions. The study was predicated on control rods being inserted 77% into the core when k_{eff} became 1.0, with the assumption then being made that the control rods would move at a constant rate from 77% inserted at beginning of core life to 0% inserted at end of core life. Also, it was assumed that all the control rods move in a blanket. Calculations showed that a point 2 feet from the bottom of the core had the highest power ratio of 1.84. This can be compared to the overall maximum power ratio of 2.0 assumed heretofore.

Thermal Bowing

A summary of thermal bowing results for the definitive design and compromise design is included in Table XX under Fuel Assembly Design.

A program was initiated for a laboratory test to measure bowing of a subassembly from linear and non-linear temperature gradients. The experimental program is to be expanded in the future to include investigations of thermal bowing of a complete fuel assembly.

5.2 Hydraulic Design

A. Bishop

R. Berringer

Reactor Vessel Pressure Drop and Flow Studies

Topical Report YAEC-74, "Pressure Drop and Visual Flow Studies for a Heterogeneous Reactor Model" will be withheld pending the completion of flow distribution studies in the lower plenum of the pressure vessel model.

The final draft of topical report YAEC-76, "The Effect of Local Boiling on Pressure Drop and Flow Distribution in the Yankee Reactor Core" was completed. The report will be issued in December, 1958.

Fuel Rod Assembly Pressure Drop Studies

Pressure drop calculations of the flow and pressure relationships through a fuel rod assembly were investigated in light of experimental data obtained on a fuel rod assembly model. The standard analytical formulae based on the equivalent diameter concept predicted the pressure drop within +15% of the experimental data. The final draft of topical report YAEC-75, "Model Study of the Pressure Drop Relationships in a Typical Fuel Rod Assembly" will be completed in December, 1958, and will incorporate the results of experimental studies on the ferrule type fuel assembly in addition to the results obtained from the strap type fuel assembly study.

Control Rod Hydraulic Shock Absorber Dashpot Tests

Drop tests were conducted on the hydraulic shock absorber prototype. Drag forces acting on the simulated control rod assembly were calculated. The actual weight of the simulated assembly was 350 pounds. The effective weight was approximately 250 pounds after including the effect of drag. High-speed motion pictures indicated the lightest spring (initial load - 17.5 pounds, spring rate - 15 pounds per inch) allowed the velocity pressure of the water on the bottom of the dashpot to reduce the available dashpot action by approximately 40%. Additional tests were conducted with a heavier spring. The test results indicate that the heavy spring alleviated the difficulty of pre-compression, but a new piston taper will be necessary to provide optimum deceleration rates. New tapers will be machined and tested before the final design is selected. A topical report will be prepared covering the design and testing of the control rod shock absorber.

6.0 CONTROL ROD DEVELOPMENT

Metallurgy Section:
Chemistry and Ceramics Section:

R. K. McGeary, Manager
R. F. Sterling, Manager

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

J. R. Schettig

Dynamic Corrosion Testing of Silver-Indium Cadmium Alloy

Corrosion specimens of silver-indium-cadmium were exposed to borated water with lithium hydroxide and with potassium hydroxide to determine the effect of boric acid on the corrosion behavior of the alloy. Results are given in subproject 3.3 under the article entitled: "Dynamic Effects of Ag-In-Cd Alloy in Boric Acid Solution".

Nickel Plating of Silver-Indium-Cadmium

A test program to evaluate nickel plating and copper with nickel plating on Ag-In-Cd was continued. The copper transitional layer was used to provide a metallurgical bond through diffusion between the nickel plate and the base metal. Varying amounts of copper (from zero to 1/2 mil) were electroplated and diffused into the base metal and the nickel plate by heating the specimens to 550 and 600°C. for various times. Data obtained from the metallographic examination of the specimens are being processed.

Cross-sectioned specimens of each type of plated specimen are to be exposed to simulated Yankee operating water conditions to determine the corrosion resistance of the diffusion zone. Test conditions in (W) APD Loop "A" are to be 600°F. temperature, 1823 psig pressure, 38.7 fps flow, 25-35 ml/kg (STP) of hydrogen, less than 0.1 ppm O₂, pH of 10 by the addition of LiOH (no boron added) for a one month period.

Considerable difficulty has been experienced by (W) APD in obtaining a set of Ag-In-Cd specimens having a satisfactory nickel plate for use in the in-pile test loop experiments. The plating of two Ag-In-Cd control rods by the Bart Manufacturing Company for the FWR Project at the Bettis Plant appeared to be satisfactory. A final set of specimens will be nickel plated by the Bart Manufacturing Company.

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.

S. Ehrenpreis

J. Gallagher

J. Cunningham

D. Rathbone

Analysis of the Transient Behavior of the Steam Pressurizer

Effort continued on the development of a digital computer program for analysis of the transient behavior of a steam pressurizer. This program will be utilized in the development of an analog model for the steam pressurizer for use in the analysis of the transient response of the controlled Yankee plant. Programming of the equations describing the pressurizer has been completed and initial test runs of the program are in progress.

8.0 PLANT SYSTEMS DEVELOPMENT

Reactor Systems Design Group:
Plant Systems Design Group:

P. B. Haga, Supervisor
H. A. Smith, Supervisor

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

8.5 Chemical Handling and Control Systems

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

8.11 Reactor Handling Tools

E. V. Anderson

A. F. Fritz

H. S. Kresny

Head Gasket and Seal Ring Handling Fixture

Several improvements were made in the design of the head gasket and seal ring handling fixture. Previously it was necessary to actuate approximately six mechanisms individually to release or pick up the seal ring. In addition, the gaskets were released automatically when the fixture contacted the vessel flange. In the new conceptual design, all of the seal ring fixtures are actuated with one vertical motion by means of a simple linkage scheme connection. The gasket handling mechanisms are now tied in with the linkage employed for the seal ring in order to have control over the release of the gaskets. Consequently, all mechanisms are now actuated with one vertical motion.

The actuating device for the mechanisms will be similar to that contemplated for the plate and barrel handling fixture. A mock-up of the barrel handling fixture is now being fabricated and will be tested in the (W) APD High Bay Building Deep Pit. The test results should be applicable to both types of handling fixtures.

9.0 PLANT SAFETY ANALYSIS

Reactor Engineering Department:
Large Plants Engineering:

W. E. Abbott, Manager
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 third quarter of 1958.

10.0 CRITICALITY EXPERIMENTS

Reactor Evaluation Section:

P. W. Davison, Manager

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

S. S. Berg

W. H. Bergman

D. F. Hanlen

Critical Experiments

Core Configuration Changes

On July 15, 1958, the CRX 3:1 water/uranium metal ratio experimental program was completed. After removing the fuel, the core tank and spacer plates for the 3:1 core were removed and the counterparts for the 2.23:1 core were installed. The critical experiments on the new water/metal ratio core were completed on September 21.

On September 25, replacement of the 2.23:1 core accessories by those for the 4:1 water/uranium ratio was completed and the new loading to critical was begun.

Critical Loadings

The 2.23:1 ratio core was loaded to critical in eight steps to a critical loading of 3016 fuel units in a cylindrical configuration. After loading adjustments were made, the final "clean-just-critical" loading, without control rods, was determined to be 3046 fuel rods at 18.2°C.

Loading the 4:1 core to critical was accomplished in five steps and was concluded with 1317 fuel rods and four control rods in a cylindrical configuration in the core. The "clean" loading was subsequently determined to be 1300 fuel rods at 18°C.

Flux Profiles for 2.23:1 W/U Ratio CRX Core

Flux profiles were obtained by two general methods, fuel rod scanning and gold foils.

The foils are normally inserted 1-1/2 unit cells (one fuel rod and associated moderator, or one-quarter of each of four fuel rods with enclosed moderator) from the midline and from 2 to 4 inches above the bottom lucite spacer plate (see Figure-20.) Figure-21 shows a radial profile from Run No. 17 made using gold foil techniques in the clean core. The values for reflector savings and buckling were determined by the JOFIT code to be 8.52 cm and $34.6 \times 10^{-4} \text{ cm}^{-2}$, respectively. Figure-22 is the axial profile obtained on the same run, again using gold foils. The reflector savings and buckling are being obtained from the COFIT code application of this and similar curves. Figures-23 and 24 are fuel rod scans in the borated core, radial and axial respectively. The values of reflector savings and buckling with a moderator boron concentration of 443 ppm are tabulated in Table XXI.

TABLE XXI

Reflector Savings and Buckling Using Fuel Scanning Techniques for the
2.23:1 W/U Ratio CRX Core; Boron = 443 ppm

	λ	B^2
Axial	$7.08 \pm 0.15 \text{ cm}$	$(5.33 \pm 0.023) \times 10^{-4} \text{ cm}^{-2}$
Radial	$8.11 \pm 0.15 \text{ cm}$	$(24.6 \pm 0.02) \times 10^{-4} \text{ cm}^{-2}$

An axial scan made by using a 1/4 inch fission chamber is shown in Figure-25. A fuel rod was removed from position $x = 0$, $y = 18$, and a stainless steel tube sealed at the lower end was substituted. The detector was attached to a control rod drive shaft, and its position could thus be changed remotely by operation of the drive mechanism from the console. A reflector savings value of approximately 8 cm is indicated by the data obtained.

Figure 20 Yankee Critical Reactor Experiment Core Arrangement

Figure 21 Radial Flux Profile Using Gold Foils for the 2.23:1 W/U Ratio CRX Clean Core

Figure 22 Axial Flux Profile Using Gold Foils for the 2.23:1 W/U Ratio CRX Clean Core

Figure 23 Radial Flux Profile Using Fuel Rod Scanning for the 2.23:1 W/U Ratio
CRX Borated Core

Figure 2h Axial Flux Profile Using Fuel Rod Scanning for the 2.23:1 W/U Ratio

CRX Borated Core

Figure 25 Axial Flux Profile Using a One-Quarter Inch Fission Chamber for the
2.23:1 W/U Ratio CRX Clean Core

Flux Peaking in Water Slot for 2.23:1 Core

The curve shown in Figure 26 was obtained with fuel rods; it is superimposed on a profile obtained with gold foils across a water slot, through one radius and across the reflector peak. The water slot was obtained by removing the entire $x = 0$ row of fuel rods. The maximum-to-normal ratio of flux in the water slot is 1.43.

Flux Spectrum Studies for 2.23:1 Core

Disadvantage Factor

The disadvantage factor, the ratio of the average flux in the moderator to the average flux in the fuel, was measured for both the clean and poisoned core. The disadvantage factor, excluding the cadmium ratio correction, was 1.10 ± 0.02 for the clean core. There were 3065 fuel rods in the core, and control rods 2, 3, 4 and 5 were banked at 106 cm.

The disadvantage factor for the poisoned core, excluding the cadmium ratio correction, was 1.12 ± 0.02 . The core was poisoned with a concentration of 400 ppm of boron. There were 4613 fuel rods in the core with control rods 2, 3, 4 and 5 at 110 cm.

It appears that the best value of the disadvantage factor uncorrected for the cadmium ratio of U-235 is 1.11 ± 0.02 . The assigned error is based on the errors associated with weighing and counting plus an estimated error due to other factors. Using the cadmium ratio correction factor of 0.96 ± 0.02 obtained from the cadmium ratio measurement, the corrected disadvantage factor is then:

$$(1.11 \pm 0.02) \times (0.96 \pm 0.02) = 1.07 \pm 0.03$$

Resonance Escape Probability

The experimental determination of the resonance escape probability, p , involves the measurement of the ratio of resonance capture to thermal capture in U-238. This quantity, which is designated by p , is related to P by the expression $p = 1 / (1 + p f F)$, where f is the thermal utilization and F is the ratio of thermal absorption in the U-238 to the total thermal capture in the fuel.

The value of p as determined by gamma ray spectroscopy was 3.25 ± 0.1 . The clean core irradiation run (No. 29) was made on August 1. There were 3065 fuel rods in the core and control rods 2, 3, 4 and 5 were banked at 106 cm. The value of p for the poisoned core (400 ppm of boron) as measured by gamma ray spectroscopy was 2.75 ± 0.1 . There were 4613 fuel rods in the core with control rods 2, 3, 4 and 5 at 100 cm.

Figure 26 Radial Flux Profile Using Gold Foils Through a Water Slot and Reflector Peak
for the 2.23:1 W/U Ratio CRX Clean Core

The value of the cadmium ratio leading to ρ for the poisoned core was $1.365 \pm 1\%$. Because the parameter of interest is the inverse of the cadmium ratio minus one, this 1% error manifests itself as a 3% error in ρ .

Cadmium Ratio of U-235

An aluminum U-235 alloy with the same macroscopic cross section as the Yankee CRX fuel element was used in this measurement. The cadmium ratio was found to be 4.00 ± 0.04 in the fuel rods and 4.49 ± 0.04 in the moderator. These measurements were made with a clean moderator.

Peripheral Fuel Rod Worth in the 2.23:1 Core

The change in reactivity associated with the addition or removal of peripheral fuel rods was determined by measuring the reactor period before and after the fuel was removed. To accomplish this, peripheral fuel rods were connected by cords through a system of pulleys to the control room. After the stable reactor period had been determined, the fuel was selectively lifted out of the core by pulling on the appropriate cord. After the resultant period was measured, another group of fuel rods was removed. This procedure was continued until a maximum of 13 fuel rods had been extracted.

The peripheral fuel worth for the clean core containing 3190 fuel rods is $3.5 \times 10^{-5} \Delta k/k$ per fuel rod. In the core of 4745 fuel rods with borated moderator, $2.1 \times 10^{-5} \Delta k/k$ per fuel rod was obtained.

Void Coefficient of Reactivity in the 2.23:1 Core

A procedure similar to that used to determine the peripheral fuel rod worth was used to evaluate the effect of voids in the core. Two effects were measured by the following steps: 1) Thin walled aluminum tubes were installed in the core between adjacent fuel rods; the bottom ends of the tubes were sealed, and thus the tubes were filled with air; reactivity measurements were thus determined for an air void plus aluminum 2) Reactivity measurements were made on the same aluminum tubes filled with water 3) The effect of the voids alone were determined by subtracting the measurements of Step-2 from Step-1. The coefficients determined are shown in Table XXII.

TABLE XXII

Void Coefficients for the 2.23:1 W/U Ratio CRX Core

	<u>% $\Delta k/k$ per % void</u>		
	<u>Al</u>	<u>Al-air</u>	<u>Air</u>
Borated moderator	-0.40	-0.37	-0.31
Clean moderator	-0.50	-0.46	-0.35

These values are for voids distributed along radii of the core and do not represent the average void coefficient. It should be mentioned that the borated core had a volume approximately 1.6 times that of the clean core.

Temperature Coefficient in the 3:1 and 2.23:1 Cores

The temperature coefficient of reactivity was studied in the 3:1 W/U ratio CRX borated core and in the 2.23:1 W/U ratio CRX borated and clean core. Analysis of these experiments is not yet complete.

Control Rod Worth

A variety of control rod worth studies were made in both the clean and borated core. The central control rod worth was established in the clean core of 3030 fuel rods by the differential technique, i.e., plotting the differential reactivity at a number of points along the length of the control rod and integrating the area under the differential curve thus obtained. A total worth of 6.1% $\Delta k/k$ was determined.

The central control rod and various rod groups were evaluated by adding increments of boron to the moderator and determining the total worth. The core loading for these measurements was 4704 fuel rods; Figure-20 indicates the location of the control rods. A total worth of 4.1% $\Delta k/k$ in this core was indicated. Figure-27 is the rod worth curve of control rod No. 1.

Other measurements included comparisons of the worth of a Ag-In-Cd control rod with the worth of a Ag-Cd rod in the central position (Rod No. 1). The Ag-In-Cd rod was found to have a worth 1.04 times that of the Ag-Cd rod. This measurement was made with a clean moderator. A set of control rods was obtained from the FWR Group at the Bettis Plant enabling a comparison to be made between the worths of a Ag-In-Cd and a hafnium control rod. Measurements were made with 180 ppm boron in the moderator indicate that the hafnium rod worth was 1.01 times that of the Ag-In-Cd rod. A comparison of Li-Al and stainless steel was also made; the data that were obtained are now being analyzed.

Figure 27 Central-Silver Cadmium Control Rod Worth in a Clean Core Containing 3030 Fuel Rods

11.0 RADIATION DAMAGE EXPERIMENTS

Mechanical and Thermal Design Section:
Metallurgy Section:

A. G. Thorp, Manager
R. K. McGeary, Manager

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

11.1 Design and Fabrication the In-Pile Test Loop

A. Bishop

C. Dishman

The loop fabrication and component testing continued on schedule except for late deliveries of the main flow control valve and Honeywell and Bristol instrumentation. The late deliveries affected loop testing procedures due to adjustments and calibrations that were necessary after the components were installed. Instrumentation and individual component operational checks were continued. Component replacement was required in several locations.

Tests conducted at the Westinghouse Atomic Equipment Department indicated that the loop pumps would not rotate when idle at flow rates through the pump in the normal direction of up to 25 gpm.

Tests conducted at the Lummus Company indicated flow rates through the loop at room temperature for one, two, and three pumps to be 6.1 gpm, 8.95 gpm, and 11.1 gpm, respectively. The predicted flow extrapolation to operating conditions of 550°F and 2000 psi for two pumps indicated an average velocity of 10.3 ft/sec past the fuel samples. This corresponds to a maximum outlet velocity past the samples of 11.1 ft/sec at 600°F.

A complete hydraulic analysis of the loop was conducted using the latest available information. The analytical results as well as the latest experimental data are shown in Figure-28. Figure-29 shows the testing arrangement of the primary cubicle and components, primary detector cubicle control panel internals, and a section of the purification system. Figure-30 shows the purification system and data logger.

Lummus Company technicians were trained in the analytical procedures to be used in the performance of the final operational test with conditioned water and in the vacuum filling procedure to be used in loop filling. Various leaks were detected and the cold hydrostatic leak rate was reduced from 100 psi/hr to 40 psi/hr. Further progress is expected in reducing

Figure 28 WCAP-4 In-Pile Test Loop Primary System Head Curves

Figure 29 WCAP-L In-Pile Test Loop (Primary Cubicle at Left)

Figure 30 WCAP-4 In-Pile Test Loop - Data Logger and Purification System

the leak rate. Storage vessels and the pressurizer were volume calibrated. During hot operation with several known leaks, the leak rate was found to be approximately 200 ml of water per hour over a 24 hour period. Metering of make-up water was found to be inadequate and alternate procedures are being investigated.

11.2 Performance of Radiation Damage Experiments

J. J. Lombardo

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

Results obtained from the post-irradiation examination of WCAP-1-2, 1-4, 2-3, and 2-6 process water capsules by KAPL are as follows:

1. Fine hairline cracks were observed in the ferrule brazes at 4 to 6X magnification that were undetected by the unaided eye.
2. Helium in appreciable amounts was identified in the gas sample after burn-ups of approximately 10,000 MWD/T.
3. A high percentage of water vapor was determined by analysis, presumably resulting from insufficient drying of fuel pellets or insufficient protection from moisture after drying.
4. Xenon and krypton release was a small proportion of the gas sample.
5. No grain growth within pellets was observed.
6. No significant fuel tube diameter or length changes were observed.
7. No bowing or bulging of the fuel tubes was observed.

Final results of the post-irradiation examination of the four capsules have not been received from KAPL.

Post-Irradiation examination of the WCAP-2-4 process water capsule received by KAPL has been delayed because of repair and decontamination of their disassembly hot cell facility. The latest information from KAPL indicates that the hot cell facility will not be available for use until the middle of November.

WCAP-2-8, 2-9, 2-10, 2-11, 2-12, and 2-13 Capsules

WCAP-2-8 through 2-13 process water capsules were sent to the MTR during the third quarter of 1958. A description of the six capsules is given in Table-XXIII.

TABLE XXIII

Description of WCAP-2-8 through 13 Process Water Specimens

<u>Capsule No.</u>	<u>Tube No.</u>	<u>Configuration</u>	<u>Clad Thickness</u>
2-8	A-1, A-2, A-3	Disc in center & ends	21 mils
2-9	A-4, A-5, A-6	"	21 mils
2-10	A-7, A-8, A-9	"	21 mils
2-11	B-1	Disc at one end	21 mils
	B-2	No discs	15 mils
	B-3	Disc at both ends	21 mils
2-12	B-4	Disc at one end	21 mils
	B-5	No discs	15 mils
	B-6	Disc at both ends	21 mils
2-13	C-1	Compartmented	21 mils
	C-2	Compartmented	21 mils
	C-3	Non-compartmented	15 mils

1. Cladding consists of AISI 304 stretch-formed stainless steel.
2. Tubes in Capsules 2-8 through 2-12 are 5.1 inches in length.
3. Tubes in Capsules 2-13 are 30.78 inches in length.
4. Tubes in Capsules 2-8 through 2-12 contain UO_2 pellets enriched to 5.4% in U-235.
5. Each tube in Capsule 2-13 contains 39 UO_2 pellets enriched to 2.7% in U-235, 2 pellets enriched to 5.4% and 6 pellets enriched to 6.8%.

WCAP-2-13 capsule was inserted at the beginning of Cycle 113 in the MTR. The remaining capsules will be inserted at the beginning of the next cycle.

WCAP-2-2, 2-5, and 2-7 Capsules

The three remaining process water capsules being irradiated at the MTR have received the following accumulated unperturbed thermal nvt at the end of Cycle 111:

<u>Capsule Number</u>	<u>nvt</u>
WCAP-2-2	9.8×10^{20}
WCAP-2-5	10.0×10^{20}
WCAP-2-7	9.4×10^{20}

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	$1,388 \times 10^6$
Total Heat Output, MW	392*
Coolant Flow:	
Total Rate, lbs/hr	37.8×10^6
Heat Transfer Rate, lbs/hr	34.0×10^6
Flow Area in Fuel Rod Cross Section, ft ²	15.4
Velocity Along Fuel Rods, ft/sec	14.0
Pressure, psi	
System Pressure, Nominal	2,000
System Pressure, Maximum	2,150
System Pressure, Minimum	1,850
System Pressure, Design	2,500
Total Drop Across Vessel	29.7
Drop Across Core	13.7
Heat Transfer	
"Active" Surface, ft ²	15,400
Average Flux, Btu/hr-ft ²	87,000
Maximum Flux, Btu/hr-ft ²	450,000
Average Film Coefficient, Btu/hr-ft ² -°F	6,050
Burnout Flux Ratio	
$(Q/A)_{BO}/(Q/A)_{max}$ POB**	2.19
Bettis Plant Correlation, Btu/hr-ft ²	9.80×10^5
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	14.4
Film Drop, Average	651
Center of Fuel, Maximum	4,700
Outlet of Hot Channel	603
ΔT_m at Heat Exchanger	37
Steam Temperature	475
Steam Pressure, psia	540
Coolant at Inlet to Vessel	499

* If center fuel temperature of 4500°F is used, the total heat output can be raised to 428 MW. Lifetime is reduced proportionately.

** POB - Point of Burnout.

Yankee Atomic Electric Company Reactor (Continued)

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

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

	<u>First Core</u>
Fuel Rod	
Outside Diameter, in.	0.339
Rod Lattice, in.	0.427

Yankee Atomic Electric Company Reactor (Continued)

II. Nuclear

		<u>First Core</u>
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), inches		90
Core Materials (Between Fuel Ends)	Volume, in. ³	Wt., lbs.
Fuel (10.07 gm/cm ³)	138,000	50,200
Water (0.7814 gm/cm ³)	198,800	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 and Pellet)	4,900	--
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, ε		1.043
Age, τ, cm ²		59.5
Typical performance data and nuclear parameters based on 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 Content (Atom Percent)		9.14
Fraction of Energy from:		
U-235		0.728
U-238		0.067
Pu		0.205
Nominal Shutdown Concentration of Chemical Poison, Gms. Natural Boron/Liter of H ₂ O		1.2
Moderator Temperature Coefficient, $\frac{1}{k} \frac{\partial k}{\partial T}/^{\circ}\text{F}$ at 516°F		-2 to -4 x 10 ⁻⁴
Fuel Temperature Coefficient, $\frac{1}{k} \frac{\partial k}{\partial T}/^{\circ}\text{F}$ at 516°F		-1 to -4 x 10 ⁻⁵

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.
4. YAEC-65, Quarterly Progress Report, advised of the availability of documents and reports during the January 1 to March 31, 1958 quarter.
5. YAEC-87, Quarterly Progress Report, advised of the availability of documents and reports during the April 1 to June 30, 1958 quarter.

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

The following quantities and types of reports and microcards were received by (W) APD as a result of being on the "Standard" and "Positive" distribution lists during the third quarter of 1958:

<u>Month 1958</u>	<u>Unclassified</u>	<u>Classified</u>
July	842	7
August	988	16
September	857	16

The following quantities of documents not on the "Standard" or "Positive" distribution lists were requisitioned and received during the third quarter of 1958:

<u>Month 1958</u>	<u>Ordered</u>	<u>Received</u>
July	23	11
August	12	10
September	7	7

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

	July	August	September
"Denied or not available."	8	1	7
"Attempts will be made to obtain the report."	4	5	8

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

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

Yankee Atomic Electric Company

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

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

Summary of Classified Documents Received - As of September 30, 1958

Total Documents Received	390
Total Documents Retained	169
Total Documents Returned	221

Summary of Requested Unclassified Documents - As of September 30, 1958

Total Documents Requested	338
Documents Received and Retained	283
Documents Received and Returned (on loan)	5
Documents Not Yet Received (Include documents for which some notification was received i.e. not at TISE, no copies available, etc.)	36
Documents Denied	14

Summary of Unrequested Unclassified Reports Received on Standard Distribution - As of September 30, 1958

Total Documents Received	617
Documents Retained	174
Documents Returned	443

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 YAEK Research and Development Program have been included.

To: Knolls Atomic Power Laboratory, Schenectady, New York (July 1-2, 1958)
Visit made by:

J. J. Lombardo - (W) APD

The results of the post-irradiation examination of WCAP-1-2, 1-4, 2-3 and 2-6 process water capsules were obtained. Methods of continuing the cooperation between (W) APD and KAPL personnel during the performance of the post-irradiation program were explored.

To: Argonne National Laboratory, Lemont, Illinois (July 2, 1958)
Visit made by:

A. Bournia - (W) APD

Data were obtained on pressure drop runs in which air and water were used to simulate the steam voids in a two phase flow at atmospheric pressure. The problem of predicting pressure drops across sudden expansion and contraction of flow areas in a two phase flow system was discussed. The condition of fuel elements removed from the EBWR exposed to high and low heat fluxes was observed and reasons for the condition were given.

To: The Lummus Company, Newark, New Jersey (July 30-31, August 1, 1958)
Visit made by:

C. D. Dishman - (W) APD A. G. Thorp - (W) APD

Removal of air from the primary loop section of the in-pile test loop was accomplished after several cycles of pressurizing and circulating loop water. Attempts were made to obtain flow data by operating the main flow pumps and recording the readings from the flow meter. A program was developed in order to obtain accurate flow data.

To: The Lummus Company, Newark, New Jersey (August 19-20, 1958)
Visit made by:

C. D. Dishman - (W) APD R. P. Goldstein - (W) APD M. A. Vogel - (W) APD
J. J. Lombardo - (W) APD L. E. Minnick - YAEK

Abstracts of Trip Reports (Continued)

Loop operation flow data were established with various combinations of pump operation. Some instrument calibration and settings were made. Hydrostatic testing of the purification system was witnessed.

To: Westinghouse Atomic Equipment Department, Cheswick, Pa. (September 3, 1958)
Visit made by:

E. Frisch - (W) APD
G. Landau - (W) APD

L. Balog - (W) APD

P. B. Haga - (W) APD

The need for enlarging the inside diameter of the mechanism port in the reactor vessel head to accommodate the latch type control rod drive mechanism was presented. A delivery schedule and dimension requirements for machined lower mechanism housing adapters were discussed. Control rod drive mechanism specifications and a detail drawing of the mechanism port were provided by (W) APD personnel.

To: Bettis Plant, Pittsburgh, Pa.; PWR, Shippingport, Pa. (September 3, 1958)

W. E. Abbott - (W) APD

G. Reed - YAEC

Bettis Plant: The purpose of and results obtained from PWR flow measuring instruments and fuel and coolant thermocouples were discussed. Information was given on other instrumentation such as local boiling sonic detectors, heat flux and coolant measuring devices, fuseable alloys to determine fuel assembly maximum temperatures and axial scanning flux wire thimbles.

PWR: Data obtained from PWR full power core instrumentation and experimental information instrumentation were discussed. Time, manpower and computer analysis requirements to obtain core instrumentation data were reviewed.

To: The Lummus Company, Neward, New Jersey (September 3, 1958)
Visit made by:

G. L. Duke - (W) APD

A. G. Thorp - (W) APD

L. E. Minnick - YAEC

The price changes resulting from changes to in-pile test loop equipment were reviewed. The procedure for submittal of increases for approval was agreed upon. A new date for the delivery of the test loop to the MTR was scheduled. The advisability of furnishing isometric drawings and photographs of the test loop to the MTR was discussed. A general discussion was held on the subject of (W) APD personnel working at the MTR.

To: The Lummus Company, Neward, New Jersey (September 3-4, 1958)
Visit made by:

C. D. Dishman - (W) APD

Abstracts of Trip Reports (Continued)

Pump coast-down data for the in-pile test loop were obtained with all combinations of pump operation. Several loop components which require modifications were discussed. A program was established to incorporate the modifications prior to the operation test.

To: The Naval Reactor Testing Station, Scoville, Idaho (September 15, 16, & 17, 1958)
Visit made by:

A. E. Voysey - (W) APD

W. J. Miller - YAE

Operating experience with the SLW reactor was discussed with SLW personnel including primary system long-lived activities, core instrumentation, boiling tests, and maintenance difficulties and their prevention. Information was obtained on industrial hygiene methods and operator training.

To: The Lummus Company, Newark, New Jersey (September 17, 18, & 19, 1958)
Visit made by:

C. D. Dishman - (W) APD

R. P. Goldstein - (W) APD

L. E. Minnick - YAE

Improper operation of the in-pile test loop three-way valve was observed during heating and pressurizing of the loop. An adjustment was made in pressurizer safety relief valve. Melting of the cast aluminum material surrounding the loop heater element occurred after three hours of operation of the loop heater. New heaters are to be provided using lower watt density heaters and aluminum-bronze cast material.

To: The Lummus Company, Newark, New Jersey (September 29-30, October 1, 1958)
Visit made by:

J. J. Lombardo - (W) APD

A. Krieger - (W) APD

C. D. Dishman - (W) APD

The 120 hour operational test of the in-pile test loop was observed. During the test a complete check was made of the components and of the functional operation of the loop. From this check and previous negotiations, a list of items was presented to the Lummus Company for correction and action.